Vermont Yankee License Renewal Project

Evaluation of Aging Management Programs

4.19: Reactor Vessel Surveillance Program

4.19 Reactor Vessel Surveillance Program

A. Program Description

The Reactor Vessel Surveillance Program complies with the guidelines for an acceptable Integrated Surveillance Program described in NUREG-1801, Section XI.M31, Reactor Vessel Surveillance. This program manages reduction in fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor pressure vessel is maintained for the period of extended operation.

Dave Mannai

VYNPS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as approved by License Amendment 218. This program monitors changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. As BWRVIP-ISP capsule test reports become available for RPV materials representative of VYNPS, the actual shift in the reference temperature for nil-ductility transition of the vessel material may be updated. In accordance with 10CFR50 Appendix H, VYNPS reviews relevant test reports to assure compliance with fracture toughness requirements and P-T limits.
(Ref. Bases Section 3/4.6.A, VYNPS Technical Specifications)

BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," describes the design and implementation of the ISP during the period of extended operation. BWRVIP-116 identifies additional capsules, their withdrawal schedule, and contingencies to ensure that the requirements of 10CFR50 Appendix H are met for the period of extended operation. schedule?
(Ref. Commitment Report BWRVIP-116-01)

This program is credited in the following.

- AMRM-31, Reactor Pressure Vessel

B. Evaluation

1. Scope of Program

The Reactor Vessel Surveillance Program includes all reactor vessel beltline materials as defined by 10 CFR 50 Appendix G, Section II.F.

2. Preventive Actions

No actions are taken as part of this program to prevent aging effects or mitigate aging degradation.

3. **Parameters Monitored/Inspected**

The Reactor Vessel Surveillance Program monitors reduction of fracture toughness of reactor vessel beltline materials due to neutron irradiation embrittlement. The BWRVIP ISP uses existing BWRVIP-ISP surveillance capsules in BWR plants, as well as supplemental capsules irradiated in host plants, to provide data which bounds all operating plants. The capsules in the VYNPS vessel are spares, not currently scheduled for withdrawal. VYNPS plate and weld metal is represented by the surveillance capsule in Susquehanna Unit 1.
(Ref. Section 3.1, BVY 03-29)

4. **Detection of Aging Effects**

The Reactor Vessel Surveillance Program detects the effects of reduction of fracture toughness prior to loss of the reactor vessel intended function in accordance with the information provided in Monitoring and Trending.

5. **Monitoring and Trending**

The Reactor Vessel Surveillance Program uses existing BWRVIP-ISP surveillance capsules in BWR plants, as well as supplemental capsules irradiated in host plants, to provide data which bounds all operating plants. The capsules in the VYNPS vessel are spares, not currently scheduled for withdrawal. VYNPS plate and weld metal is represented by the surveillance capsule in Susquehanna Unit 1. Present trend?
(Ref. Section 3.1, BVY 03-29)

Representative capsule data will be evaluated using the methods in Regulatory Guide 1.99 in accordance with Appendix G to 10CFR50 for the determination of the actual shift in the reference temperature for nil-ductility transition (RT_{NDT}) of the vessel material. Charpy shift results will be used to reevaluate embrittlement projections for vessel beltline materials represented by materials in the capsule. If changes to pressure-temperature limits are required due to a reassessment of limiting RT_{NDT} values, changes to the licensing basis will be requested.
(Ref. Section 3.1, BVY 03-29)

Enhancement: The Reactor Vessel Surveillance Program will be enhanced to proceduralize (in PP 7027 or a new procedure) the data analysis, acceptance criteria, and corrective actions described in this program description. Schedule?

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 as a contingency.
(Ref. Section 3.1, BVY 03-29)

6. Acceptance Criteria

VYNPS embrittlement projections will comply with 10CFR50 Appendix G limits for the period of extended operation.

RT_{NDT} for material in the beltline will remain below screening criterion using end of life fluence.

Enhancement: The Reactor Vessel Surveillance Program will be enhanced to proceduralize (in PP 7027 or a new procedure) the data analysis, acceptance criteria, and corrective actions described in this program description.

Acceptable pressure-temperature curves for heatup and cooldown of the unit will be maintained in Technical Specifications. The operational EFPY shall not exceed the Technical Specification limits for the pressure-temperature curves. ***(Ref. Section 3/4.6, VYNPS Technical Specifications)***

7. Corrective Actions

Specific corrective action and confirmation will be implemented as follows.

If embrittlement projections drop below 50 ft-lbs, the margins of safety against fracture will be demonstrated to be equivalent to those of Appendix G of ASME Section XI. This could be accomplished by demonstrating that the equivalent Margin Analysis documented in BWRVIP-74 represents a bounding evaluation for the VYNPS reactor vessel.

If RT_{NDT} for material in the beltline is projected to exceed the screening criterion using end of life fluence, VYNPS may implement flux reduction programs that are reasonably practicable to avoid exceeding this criterion. If no reasonably practicable flux reduction program will avoid exceeding the screening criteria, VYNPS will submit a safety analysis to determine actions to prevent potential failure of the reactor vessel as a result of postulated events if continued operation beyond the screening criterion is allowed.

Enhancement: The Reactor Vessel Surveillance Program will be enhanced to proceduralize (in PP 7027 or a new procedure) the data analysis, acceptance criteria, and corrective actions described in this program description.

If a capsule is not withdrawn as scheduled by BWRVIP-ISP, the NRC will be notified and the withdrawal schedule will be updated and submitted to the NRC. ***(Ref. Section 5.7, BWRVIP-86-A)***

8. Confirmation Process

This attribute is discussed in Section 2.0, Background.

9. Administrative Controls

This attribute is discussed in Section 2.0, Background.

10. Operating Experience

Operating experience provides assurance that the program will be effective in managing effects of aging so that components crediting this program can perform their intended function consistent with the current licensing basis during the period of extended operation. The fact that VYNPS now participates in the BWRVIP ISP ensures that future operating experience from all participating BWRs will be factored into this program. For more information on applicable operating experience, see VYNPS Report LRPD-05, Operating Experience Review Results.

Where is the evaluation

C. References

10CFR50, Appendix G, Fracture Toughness Requirements, U.S. NRC

10CFR50, Appendix H, Reactor Vessel Material Surveillance Program Requirements, U.S. NRC

BVY 03-29, Technical Specifications Proposed Change No. 258, RPV Fracture Toughness and Material Surveillance Requirements, March 26, 2003

BWRVIP-86-A, BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program (ISP) Implementation Plan, EPRI Report 1003346, October 2002

Commitment Report BWRVIP-116-01, BWRVIP ISI Implementation for License Renewal, 9/17/2003

PP 7027, Rev. 03, LPC 00, Reactor Vessel Internals Management Program

VYNPS Technical Specification, Amendment 228

D. Summary

The Reactor Vessel Surveillance Program ensures that reactor vessel embrittlement is monitored and corrective actions are taken prior to exceeding allowable limits. The Reactor Vessel Surveillance Program provides reasonable assurance that aging effects will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Reactor Vessel Surveillance Program

The following enhancement will be initiated prior to the period of extended operation.

Attributes Affected	Enhancement
5. Monitoring and Trending 6. Acceptance Criteria 7. Corrective Actions	The Reactor Vessel Surveillance Program will be enhanced to proceduralize (in PP 7027 or a new procedure) the data analysis, acceptance criteria, and corrective actions described in this program description.

XI.M31 REACTOR VESSEL SURVEILLANCE

Program Description

The Code of Federal Regulations, 10 CFR Part 50, Appendix H, requires that peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E >1MeV), or that reactor vessel beltline materials be monitored by a surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard. However, the surveillance program in ASTM E 185 is based on plant operation during the current license term, and additional surveillance capsules may be needed for the period of extended operation. Alternatively, an integrated surveillance program for the period of extended operation may be considered for a set of reactors that have similar design and operating features in accordance with 10 CFR Part 50, Appendix H, Paragraph II.C. Additional surveillance capsules may also be needed for the period of extended operation for this alternative.

The existing reactor vessel material surveillance program provides sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the period of extended operation, and to determine the need for operating restrictions on the inlet temperature, neutron spectrum, and neutron flux. If surveillance capsules are not withdrawn during the period of extended operation, operating restrictions are to be established to ensure that the plant is operated under the conditions to which the surveillance capsules were exposed.

All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of ASTM E 185-82, to the extent practicable, for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the Nuclear Regulatory Commission (NRC) prior to implementation. Untested capsules placed in storage must be maintained for future insertion.

An acceptable reactor vessel surveillance program consists of the following:

1. The extent of reactor vessel embrittlement for upper-shelf energy and pressure-temperature limits for 60 years is projected in accordance with the NRC Regulatory Guide (RG) 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials." When using NRC RG 1.99, Rev. 2, an applicant has a choice of the following:

   a. Neutron Embrittlement Using Chemistry Tables

   An applicant may use the tables in NRC RG 1.99, Rev. 2, to project the extent of reactor vessel neutron embrittlement for the period of extended operation based on material chemistry and neutron fluence. This is described as Regulatory Position 1 in the RG.

   b. Neutron Embrittlement Using Surveillance Data

   When credible surveillance data is available, the extent of reactor vessel neutron embrittlement for the period of extended operation may be projected according to Regulatory Position 2 in NRC RG 1.99, Rev. 2, based on best fit of the surveillance data. The credible data could be collected during the current operating term. The applicant may have a plant-specific program or an integrated surveillance program during the period of extended operation to collect additional data.

2. An applicant that determines embrittlement by using the NRC RG 1.99, Rev. 2, tables (see item 1[a], above) uses the applicable limitations in Regulatory Position 1.3 of the RG. The limits are based on material properties, temperature, material chemistry, and fluence.

3. An applicant that determines embrittlement by using surveillance data (see item 1[b], above) defines the applicable bounds of the data, such as cold leg operating temperature and neutron fluence. These bounds are specific for the referenced surveillance data. For example, the plant-specific data could be collected within a smaller temperature range than that in the RG.

4. All pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage. (Note: These specimens are saved for future reconstitution use, in case the surveillance program is reestablished.)

5. If an applicant has a surveillance program that consists of capsules with a projected fluence of less than the 60-year fluence at the end of 40 years, at least one capsule is to remain in the reactor vessel and is tested during the period of extended operation. The applicant may either delay withdrawal of the last capsule or withdraw a standby capsule during the period of extended operation to monitor the effects of long-term exposure to neutron irradiation.

6. If an applicant has a surveillance program that consists of capsules with a projected fluence exceeding the 60-year fluence at the end of 40 years, the applicant withdraws one capsule at an outage in which the capsule receives a neutron fluence equivalent to the 60-year fluence and tests the capsule in accordance with the requirements of ASTM E 185. Any capsules that are left in the reactor vessel provide meaningful metallurgical data (i.e., the capsule fluence does not significantly exceed the vessel fluence at an equivalent of 60 years). For example, in a reactor with a lead factor of three, after 20 years the capsule test specimens would have received a neutron exposure equivalent to what the reactor vessel would see in 60 years; thus, the capsule is to be removed because further exposure would not provide meaningful metallurgical data. Other standby capsules are removed and placed in storage. These standby capsules (and archived test specimens available for reconstitution) would be available for reinsertion into the reactor if additional license renewals are sought (e.g., 80 years of operation). If all surveillance capsules have been removed, operating restrictions are to be established to ensure that the plant is operated under conditions to which the surveillance capsules were exposed. The exposure conditions of the reactor vessel are monitored to ensure that they continue to be consistent with those used to project the effects of embrittlement to the end of license. If the reactor vessel exposure conditions (neutron flux, spectrum, irradiation temperature, etc.) are altered, then the basis for the projection to 60 years is reviewed; and, if deemed appropriate, an active surveillance program is re-instituted. Any changes to the reactor vessel exposure conditions and the potential need to re-institute a vessel surveillance program is discussed with the NRC staff prior to changing the plant's licensing basis.

7. Applicants without in-vessel capsules use alternative dosimetry to monitor neutron fluence during the period of extended operation, as part of the aging management program (AMP) for reactor vessel neutron embrittlement.

8. The applicant may choose to demonstrate that the materials in the inlet, outlet, and safety injection nozzles are not controlling, so that such materials need not be added to the material surveillance program for the license renewal term.

The reactor vessel monitoring program provides that, if future plant operations exceed the limitations or bounds specified in item 2 or 3, above (as applicable), such as operating at a lower cold leg temperature or higher fluence, the impact of plant operation changes on the extent of reactor vessel embrittlement will be evaluated and the NRC will be notified. An applicant without capsules in its reactor vessel is to propose reestablishing the reactor vessel surveillance program to assess the extent of embrittlement. This program will consist of (1) capsules from item 6, above; (2) reconstitution of specimens from item 4, above; and/or (3) capsules made from any available archival materials; or (4) some combination of the three previous options. This program could be a plant-specific program or an integrated surveillance program.

Evaluation and Technical Basis

Reactor vessel surveillance program is plant-specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant submits its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal.

References

10 CFR Part 50, Appendix H, *Reactor Vessel Material Surveillance Program Requirements*, Office of the Federal Register, National Archives and Records Administration, 2005.

ASTM E-185, *Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels*, American Society for Testing Materials, Philadelphia, PA. (Versions of ASTM E-185 to be used for the various aspects of the reactor vessel surveillance program are as specified in 10 CFR Part 50, Appendix.)

NRC Regulatory Guide 1.99, Rev. 2, *Radiation Embrittlement of Reactor Vessel Materials,* U.S. Nuclear Regulatory Commission, May 1988.

A.2.1.26 Reactor Vessel Surveillance Program

VYNPS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as incorporated into the plant Technical Specifications by Amendment 218. The Reactor Vessel Surveillance Program monitors changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. As BWRVIP-ISP capsule test reports become available for RPV materials representative of VYNPS, the actual shift in the reference temperature for nil-ductility transition of the vessel material may be updated. In accordance with Appendix H to 10CFR50, VYNPS reviews relevant test reports to assure compliance with fracture toughness requirements and P-T limits.

BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," describes the design and implementation of the ISP during the period of extended operation. BWRVIP-116 identifies additional capsules, their withdrawal schedule, and contingencies to ensure that the requirements of 10CFR50 Appendix H are met for the period of extended operation.

A.2.1.27 Selective Leaching Program

The Selective Leaching Program ensures the integrity of components made of cast iron, bronze, brass, and other alloys exposed to a raw water, treated water, or groundwater environment that may lead to selective leaching of one of the metal components. The program includes a one-time visual inspection and hardness measurement of selected components that may be susceptible to selective leaching to determine whether loss of material due to selective leaching is occurring, and whether the process will affect the ability of the components to perform their intended function for the period of extended operation.

A.2.1.28 Service Water Integrity Program

The Service Water Integrity Program relies on implementation of the recommendations of GL 89-13 to ensure that the effects of aging on the service water systems (SWS) will be managed for the period of extended operation. The SWS include the service water, residual heat removal service water, and alternate cooling systems. The program includes component inspections for erosion, corrosion, and blockage and performance monitoring to verify the heat transfer capability of the safety-related heat exchangers cooled by SWS. Chemical treatment using biocides and chlorine and periodic cleaning and flushing of redundant or infrequently used loops are the methods used to control or prevent fouling within the heat exchangers and loss of material in SWS components.

B.1.24 REACTOR VESSEL SURVEILLANCE

Program Description

The Reactor Vessel Surveillance Program complies with the guidelines for an acceptable Integrated Surveillance Program described in NUREG-1801, Section XI.M31, Reactor Vessel Surveillance. This program manages reduction in fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor pressure vessel is maintained for the period of extended operation.

VYNPS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as approved by License Amendment 218. This program monitors changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. As BWRVIP-ISP capsule test reports become available for RPV materials representative of VYNPS, the actual shift in the reference temperature for nil-ductility transition of the vessel material may be updated. In accordance with 10CFR50 Appendix H, VYNPS reviews relevant test reports to assure compliance with fracture toughness requirements and P-T limits.

BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," describes the design and implementation of the ISP during the period of extended operation. BWRVIP-116 identifies additional capsules, their withdrawal schedule, and contingencies to ensure that the requirements of 10CFR50 Appendix H are met for the period of extended operation.

NUREG-1801 Consistency

The Reactor Vessel Surveillance Program at VYNPS will be consistent with the program described in NUREG-1801, Section XI.M31, Reactor Vessel Surveillance, with one enhancement.

Exceptions to NUREG-1801

None

Enhancements

The following enhancement will be initiated prior to the period of extended operation.

Attributes Affected	Enhancement
5. Monitoring and Trending Actions 6. Acceptance Criteria 7. Corrective Actions	The Reactor Vessel Surveillance Program will be enhanced to proceduralize the data analysis, acceptance criteria, and corrective actions described in this program description.

Operating Experience

VYNPS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as incorporated into the plant Technical Specifications by Amendment 218. The fact that VYNPS participates in the BWRVIP ISP ensures that future operating experience from all participating BWRs will be factored into this program.

Conclusion

The Reactor Vessel Surveillance Program ensures that reactor vessel degradation is identified and corrective actions are taken prior to exceeding allowable limits. The Reactor Vessel Surveillance Program provides reasonable assurance that aging effects will be managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Section **4.4.20** presents conclusions regarding the effectiveness of this program for managing aging effects.

3.4.21. Reactor Vessel Surveillance Program

The VYNPS Reactor Vessel Surveillance Program is a condition monitoring program, which monitors reduction of fracture toughness of reactor vessel beltline materials due to irradiation embrittlement. The attributes of this program are described in LRPD-02 (**Ref. 5.17**).

The results of OE reviews described in Section 2.0 are as follows.

Item	Issue	OE Evaluation
Action Item / Regulatory Commitment No.BWRVIP-116_01	BWRVIP-116 identifies additional capsules, their withdrawal schedule, and contingencies to ensure that the requirements of 10CFR50 Appendix H are met for the period of extended operation.	VYNPS has committed to participate in the BWRVIP Integrated Surveillance Program (ISP) thereby ensuring that operating experience from all participating BWRs is factored into this program.
VYNPS to NRC Letter BVY-03-029 and NRC to VYNPS Letter NVY-04-027	Plant specific surveillance requirements in technical specifications were replaced with NRC-endorsed BWRVIP integrated surveillance program (ISP) criteria.	Technical specification criteria assure continuing compliance with 10 CFR 50 Appendix H requirements for managing fracture toughness.

Section **4.4.21** presents conclusions regarding the effectiveness of this program for managing aging effects.

3.4.22. Service Water Integrity Program

The VYNPS Service Water (SW) Integrity Program is an inspection, monitoring, and testing program, which manages loss of material, cracking and fouling on service water, residual heat removal service water, and alternate cooling system components and structures and components serviced by the service water systems. The attributes of this program, which relies on implementation of recommendations of NRC Generic Letter (GL) 89-13, are described in LRPD-02 (**Ref. 5.17**).

The results of OE reviews described in Section 2.0 are as follows.

VYNPS program is consistent with the NUREG-1801 program with one exception; when reactor head closure studs are removed for examination, either a surface or volumetric examination is allowed. Since cracking initiates on the outside surfaces of bolts and studs, a qualified surface examination meeting the acceptance standards of IWB-3515 provides at least the sensitivity for flaw detection that an end shot ultrasonic examination provides on bolts or studs. Therefore, the VYNPS program is effective at managing loss of material and cracking for applicable components (**Ref. 5.17**)

The Reactor Head Closure Studs Program has been effective at managing aging effects. The Reactor Head Closure Studs Program provides reasonable assurance that the effects of aging will be managed such that the applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.4.21. Reactor Vessel Surveillance Program

The Reactor Vessel Surveillance Program complies with the guidelines for an acceptable program described in NUREG-1801, Section XI.M31, Reactor Vessel Surveillance. This program manages reduction in fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor pressure vessel is maintained for the period of extended operation. (**Ref. 5.17**)

VYNPS is a participant in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as incorporated into the plant Technical Specifications by Amendment 218. This program monitors changes in the fracture toughness properties of ferritic materials in the reactor pressure vessel (RPV) beltline region. As BWRVIP-ISP capsule test reports become available for RPV materials representative of VYNPS, the actual shift in the reference temperature for nil-ductility transition of the vessel material may be re-established. In accordance with Appendix H to 10CFR50, VYNPS reviews relevant test reports and makes a determination or whether or not a change in Technical specifications is required as a result of the data. (**Ref. 5.17**)

BWRVIP-116, "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Implementation for License Renewal," describes the design and implementation of the ISP during the period of extended operation. BWRVIP-116 identifies additional capsules, their withdrawal schedule, and contingencies to ensure that the requirements of 10CFR50 Appendix H are met for the period of extended operation. (**Ref. 5.17**)

The Reactor Vessel Surveillance Program has been effective at managing aging effects. The Reactor Vessel Surveillance Program provides reasonable assurance that the effects of aging will be managed such that the applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

4.4.22. Service Water Integrity Program

Recent performance test and inspection results (2004) provide evidence that the program is effective for managing aging effects for applicable components. For example, diesel generator service water cooled heat exchanger performance testing revealed no significant performance degradation, RHR heat exchanger inspection revealed no loss of material, cracking or fouling, a

Entergy

Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Operations, Inc.
185 Old Ferry Road
Brattleboro, VT 05302-0500

March 26, 2003
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

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

Vermont Yankee
License Amendment
and
Technical Specification
Proposed Change
List

License Amendments and Changes

The purpose of this note is to assist in clarifying use of the terms **Amendment** (to either the Facility License or the PDAR/FSAR) and **Change** to tech specs; both were issued and/or used by the AEC/NRC and VY in the VY Operating License revision process.

Amendments

Once the Operating License was granted, the AEC, and later the NRC, authorized revisions or modifications to the Facility License by issuing **Amendments.** Initially, the AEC-issued Amendments were used only to transmit approved revisions or modifications to the specific conditions and/or requirements incorporated in the License; e.g., financial qualifications, owners, Commission-approved environmental requirements requested by the states, increases in reactor power level, U-235 possession limits). At the outset, the AEC did not use Amendments for approval of tech spec changes (see footnote 2, below).

The amendment numbers issued by the AEC started with No. 1 (dated April 21, 1972), which authorized possession and use of additional Special Nuclear and Byproduct Materials. Subsequent AEC/NRC-issued Amendment Nos. have been sequential with no break in number continuity. The last Amendment, as of the date of preparation of this note (November 17, 1998), was No. 162, issued on September 1, 1998.

VY also used amendment numbers to amend its License Application, but only for the specific purpose of identifying sequentially issued versions of the PDAR/FSAR(1), which form a part of the License Application. These Amendment Nos. also started with No. 1, thus replicating the first 35 AEC/NRC-issued Amendment Nos., but are unrelated thereto.

Changes

The AEC initially issued their approval of VY-proposed tech spec changes using **Change** Nos., but this procedure changed with time(2).

VY assigned Proposed Change (PC) Nos. to its requests for tech spec revisions; this started with PC No. 1, dated June 16, 1972. However, there were a limited number of earlier requests that were not assigned a PC No.

---

(1) The PDAR, and subsequently the FSAR, were revised by issuing Amendments 1 through 35 thereof. Amendment 10 of the PDAR became the FSAR on December 31, 1969, and subsequent changes were submitted continuing the Amendment No. sequence. Amendment 35 of the FSAR became Rev. 0 of the UFSAR on July 20, 1982, in response to revisions to Part 50.71 dealing with updating of the FSAR. Subsequent issues of the UFSAR have been identified using Revision Nos.

(2) The first 17 Change Nos. used by the AEC in approving tech spec changes proposed by VY were not associated or correlated with Amendment Nos. This practice changed with the issuance of Change No. 18 on June 19, 1974, when the AEC's notification letter specifically stated that Amendment No. 7 to the License was issued "incorporating Change No. 18". (As an additional example, Amendment No. 12, issued on December 3, 1974, "include[d] Change No. 23 to the Technical Specifications".) The practice of using both an amendment number and a change number when approving proposed tech spec changes ceased following the issuance of Change No. 29 on November 12, 1975. Thereafter, the NRC (the name of the agency was changed in January 1975) dropped the use of Change Nos., and all subsequent tech spec changes were approved using Amendment Nos.

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
Not Assigned		1/19/72	Reduction in Secondary Containment Negative Pressure Requirement		
Not Assigned		3/10/72	Tech Spec Change for Operation at 1% Power (Rod Worth vs. Moderate Density)		
[redacted]	[redacted]	[redacted]	1% License: Authorized operation to 1% of rated power (15.9 Mwt)	3/21/72	DPR-28 Issued
[redacted]	[redacted]	[redacted]	Authorization to receive, possess and use additional special nuclear and byproduct materials	4/21/72	A-1
Not Assigned		6/29/72	Clarification of Primary Containment testing requirements; admin changes	8/1/72	C-2
Not Assigned		7/6/72	Bypass RPS scrams with MSIV closed	7/24/72	C-1
[redacted]	[redacted]	[redacted]	Authorized operation to 20% rated power	9/7/72	A-2
[redacted]	[redacted]	[redacted]	Technical Specification Changes Made in Conjunction With Issuance of Temporary Operating License	9/7/72 9/12/72 Correction	C-3
[redacted]	[redacted]	[redacted]	Authorized temporary operation to 100% rated power	10/12/72	A-3
1		6/16/72 6/16/72	Changes to Main Condenser Air Ejector Off-Gas (AOG) System AOG Modification Description and Operation (Attachment A of PC No.1)		
		7/25/72	Supplement No. 1	9/18/72 Accepts Modification	
		7/26/73	Supplement No. 2	8/29/73 Accepts Suppl. 2	
[redacted]	[redacted]	[redacted]	Clarifies responsibility of Advisory Group for Environmental Monitoring Program (from ASLB Hearing Transcript of 10/26)	1/10/73	C-4
2		10/12/72 2/12/73 4/3/73	MSIV Closure in Refuel, Shutdown or Startup Modes When Flow exceeds 50% Rated Supplement No. 1 Completion of modification	3/6/73 4/11/73 C-6 (Authorizes Modification)	
5		12/20/72	Increase in Total Amount of U-235 For Fuel Rod Replacement Program	1/8/73	A-4
6		12/20/72	Higher Peak Noble Gas Release Rate Prior to Fuel Sipping	1/3/73	C-5
[redacted]	[redacted]	[redacted]	Full Power License: Authorized Full Term Operation to 1593 Mwt	2/28/73	A-5
[redacted]	[redacted]	[redacted]	Reduces Respiratory Protection Factors in TS	4/30/73	C-7
7		1/23/73	Changes to Off-Gas System	2/16/73	(Changes involve no USQ)

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
8		2/26/73 3/15/73 4/4/73	VY Interim Off-Gas Modification Supplement No. 1 Supplement No. 2 AEC Comments (5/14/73)	5/11/73 Approves Modification	
9		3/19/73	Increase in Noble Gas Activity Release Rate	4/5/73	Denied
10		4/9/73	Senior Control Room Operator License Requirements	5/3/73 (Modifies and Approves)	C-8
11		5/23/73	Revised Definition for Abnormal Occurrences and Reporting Requirements	6/11/73	C-9
12		5/30/73	Corrective Update	1/17/74 1/28/74 2/14/74 Correction 4/10/74	C-13 C-15 C-17
13		8/15/73	Fuel Densification	8/24/73	C-10
14		9/18/73	First Reload License Submittal	11/16/73	No TS Change required.
		10/4/73	Supplement #1; Response to request for additional information		
		10/19/73	Effects of Inverted Control Blade Absorber Tubes; response to 9/5/73 request		
		11/12/73	Supplement 2; End of Cycle Reactivity Analysis		
		12/18/73	Additional confirmatory calculations		
15		9/25/73	Reduction in MAPLHGR for fuel assemblies with deviant enrichments	10/5/73	C-11
16		11/6/73	Fuel Channel Wear, Investigation and Corrective Actions Taken	11/16/73	C-12
17		12/13/73	Increase in MAPLHGR (based on fuel densification model in GEGAP IIIA)	1/4/74	C-14
18		2/21/74	Relief Valves settings	3/28/74	C-16
		3/19/74	Supplement No. 1		
		3/19/74	Supplement No. 2		
[redacted]			Clarifies condition requiring closed cycle operation	4/4/74	A-6
			Order to Inert Containment Atmosphere	6/19/74	A-7, C-18
19		3/8/74	Technical Specification Subsection 3.3 Control Rod System - Rod Drop Accident		
20		5/21/74	Second Core Reload	12/3/74	A-12, C-23
		WVY 74-17 7/26/74	Supplement No. 1		
		WVY 74-25 8/23/74	Supplement No. 2		
20		WVY 74-33 9/25/74	Request to load fuel pending approval of PC	10/21/74	A-10, C-21

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
21		WVY 74-5 6/25/74	Proposed ECCS Modification	12/3/74	A-11, C-22
		WVY 74-31 9/9/74	Supplement No. 1		
		WVY 74-35 10/1/74	Response to 9/11/74 RAI		
		WVY 74-38 10/7/74	Drawings to accompany response to 9/11/74 RAI		
		WVY 74-40 10/9/74	Testing and Analysis of Recirc System Discharge Valves		
		WVY 74-45 10/29/74	Second set of Independently Interlocked Contactors		
		WVY 74-51 11/13/74	Supplement No. 2		
		WVY 74-61 12/9/74	Correction to Supplement 2		
22		WVY 74-16 7/25/74	Removal of Neutron Flux Dosimeter/During First Refueling Outage	8/23/74	A-8, C-19
23		WVY 74-47 10/31/74	Revised ECCS Evaluation, GETAB and Revised Technical Specification	12/27/74 Commission Order	
		WVY 75-70 7/30/75	Supporting Information for Operation with Bypass Flow Holes Plugged		
24		WVY 74-25 8/23/74	Off-Gas System Isolation Instrumentation/Condenser Low Vacuum Trip Function	10/23/74	A-9, C-20
25		WVY 74-22 8/20/74	Preliminary Evaluation of Core Configuration		
26		WVY 75-17 3/3/75	Full Power with Relief Valve Inoperable	5/21/75	A-13, C-24
27		WVY 74-64 12/16/74	Incorporate Provisions of Reg. Guide 1.16	11/5/75	A-17, C-28
			NRC letter (11/10/75) with enclosures missing from 11/5/75 letter		
			NRC letter (1/28/74) correction to 11/5/73 letter		
28		WVY 75-24 3/20/75	Corrections Necessitated by Previous License Amendments/Organizational Changes	5/21/75 2/14/75 (Notice)	A-14, C-25
29		WVY 75-31 3/31/75	Limit Torus Suppression Pool Temperature (Response to NRC 2/14/75 letter)	10/8/75	A-16, C-27
			NRC Notice of Proposed Issuance of Amendment (7/15/75)		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
30		WVY 75-38 4/14/75	Single Valve Failure/14.4 kW/ft Technical Specification Increase in LHGR	11/12/75	A-18, C-29
		WVY 75-53 5/28/75	Supplement 1		
		WVY 75-61 7/8/75	Supplement 2		
		WVY 75-98 9/15/75	Supplement 3		
		WVY 75-100 9/22/75	Supplement 4		
		WVY 75-104 9/23/75	Withdraws the 14.4kw/ft portion of PC No. 30		
31		WVY 75-47 5/6/75	Standby Gas Treatment System	8/28/75	A-15, C-26
			NRC issued Tech Spec requiring Increased Control Rod Surveillance	3/11/76	A-20
32		WVY 75-64 7/16/75	Byproduct Material License Incorporated into Technical Specifications	2/14/77	A-31
		WVY 76-87 7/15/76	Supplement 1		
33		WVY 75-102 9/22/75	Orderly Shutdown through Use of Open Cycle Upon Loss of Cooling Tower	11/21/77	A-40
		WVY 76-56 5/10/76	Response to RAI of 4/7/76		
		WVY 77-26 3/8/77	Supplemental Information		
		WVY 77-90 9/28/77	State Authorization Letters (VT & NH)		
34		WVY 75-71 7/31/75	APRM Setdown into the Reactor Protection System	3/12/76	A-21
		WVY 75-83 8/28/75	Supplement 1		
35		WVY 75-82 8/27/75	Snubber Surveillance to Protect Primary Coolant System	7/19/76	A-24
		WVY 76-7 1/29/76	Supplement 1		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
36		WVY 75-103 9/23/75	Surveillance Interval Change from 15 to 18 Months	2/25/76	A-19
37		WVY 75-117 12/8/75	Change Table 3.1.1: Operator Response to a Failed Instrument Channel	4/29/76	A-22
38		WVY 76-39 4/7/76	Changes Instrument & Electrical Surveillance Test Interval from a maximum of 18 to 22 months	4/29/76	A-23
39		WVY 76-43 4/23/76	Refueling of Reactor Core/Cycle 4	8/2/76	A-25
		WVY 76-59 5/25/76	Supplement 1	8/23/76 Correction	
		WVY 76-77 6/23/76	Response to RAI of 6/4/76		
		WVY 76-80 7/6/76	Supplement 2		
		WVY 76-90 7/19/76	Response to RAI of 7/12/76		
40		WVY 76-82 7/8/76	Delete High Drywell Pressure Signal from Automatic Isolation for RHR Shutdown Cooling Isolation Valves	Withdrawn	
		WVY 77-11 2/2/77	Response to RAI of 1/3/77		
		WVY 78-39 4/20/78	Withdrawal Letter		
41		WVY 76-88 7/15/76	Operation and Surveillance Requirements for CAD System (see WVY 76-67, dated 6/1/76)	Withdrawn	
		FVY 83-62 6/17/83	Withdrawal Letter		
42		WVY 76-70 6/8/76	Deletion of Testing of Standby Gas Treatment System Gaskets and Doors	1/19/79	A-49
		WVY 78-45 5/11/78	Supplement 1; Testing Requirements for SBGT System		
43		WVY 76-85 7/15/76	Installation of 480 Volt Uninterruptible Power Supply/Emergency Core Cooling Valves	8/2/76	A-26
44		WVY 76-83 7/8/76	Modifies Conditions/Requirements for Discharge of Condenser Cooling from 9/6/75 to 5/31/76	9/6/76	A-28
		WVY 77-29 3/15/77	Status of Appendix B Environmental Monitoring Requirements after Completion of Phase IV Open Cycle Testing		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
45		WVY 76-89 7/15/76	Replacement of Valve Position Limiters with Inline Orifices	8/2/76 8/23/76 Correction	A-27
46		WVY 76-154 12/10/76	Drywell/Suppression Chamber Differential Pressure	1/31/79 2/28/79 Correction	A-50
		WVY 77-42 4/14/77	Supplement 1		
		WVY 78-46 5/16/78	Additional Information		
47		WVY 76-103 8/26/76	Use of Dose Integrating Devices in High Radiation Areas	6/16/77	A-36
		WVY 76-140 11/10/76	Supplement 1		
		WVY 77-34 3/30/77	Modification to prior submittals		
48		WVY 76-102 8/26/76	Change MAPLHGR Curves to Current Design Limits (LHGR/13.4 kw/ft)	2/10/77	A-30
49		WVY 76-101 11/5/76	Spent and New Fuel Storage/Moving Racks in Spent Fuel Pool	9/15/77 (SE sent 6/10/77) 6/20/77 (Suppl 1 To SE)	A-37
		WVY 77-10 2/1/77	Response to RAI of 1/11/77		
			3/30/77 NRC Summary of 3/27/77 Meeting		
		WVY 77-36 4/4/77	Response to RAI of 3/14/77		
		WVY 77-44 4/27/77	Response to RAI of 4/15/77		
		WVY 77-52 5/11/77	Supplemental Dynamic Analysis in response to 3/14/77 Letter		
		WVY 77-59 6/3/77	Modification of Description of supplemental Dynamic Analysis		
50		WVY 76-108 9/2/76	Recirculation Pump Discharge Valve and Bypass Valve Surveillance	3/22/77	A-32

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
51		WVY 76-109 11/10/76	Single Recirculation Loop Settings and Limits	Withdrawn	
		FVY 83-62 6/17/83	Request to place PC on hold		
		FVY 86/22 3/12/86	Withdrawn - See PC 132		
52		WVY 76-136 11/9/76	Pressure-Temperature Limitations/10CFR Part 50, Appendix G	3/23/77	A-33
53		WVY 76-121 10/15/76	Expand Radiological Surveillance Program	4/19/77	A-35
54		WVY 76-133 11/5/76	MAPLHGR Curve Correction to 1/3 Drilled Core	Withdrawn	(see NRC letter of 3/25/77)
		No VY Letter No. 1/19/77	Cycle 4 MAPLHGR Limits		
		WVY 77-15 2/18/77	Response to 2/15/77 telephone request		
		WVY 77-16 2/23/77	Agreement to reanalyze LOCA response		
			NRC letter (3/25/77) concludes no MAPLHGR restrictions required		
		WVY 77-80 9/14/77	Withdrawal of Proposed Change		
55		WVY 76-134 11/8/76	Reactor Building Crane Surveillance Prior to Fuel Cask Handling	1/28/77	A-29
56		WVY 76-151 12/3/76	Exposure Dependent MCPR Operating Limits	4/8/77	A-34
		WVY 77-30 3/17/77	Response to 2/15/77 RAI		
57		WVY 77-27 3/9/77	RWM Operability Power Level Increase	9/30/77	A-39
58					

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
59		WVY 77-31 3/18/77	Fire Protection Systems (see WVY 77-8, dated 1/31/77, and NRC letter dated 12/2/76)	1/13/78	A-43
			6/16/77 NRC Revised sample fire protection Tech Specs		
		WVY 77-64 7/14/77	Supplement 1: Modification of Proposed Change pages		
		WVY 77-73 8/18/77	Restates VY position on Fire Protection Issues		
		WVY 77-105 11/30/77	Supplement 2: Incorporates revisions agreed upon since 6/16/77 NRC letter revising sample fire protection Tech Specs		
		WVY 80-57 4/7/80	Self-Contained Breathing Apparatus for Control Room Personnel		
60		WVY 77-60 6/8/77	Administrative Radiation Protection Controls (Compliance to R.G. 1.8)	9/30/77	A-39
61		WVY 77-69 8/5/77	High Drywell Trip Setpoint Revision from 2.0 to 2.5 psig	2/7/78	A-44
62		WVY 77-62 7/1/77	Cycle 5 Reload	9/30/77	A-39
		WVY 77-71 8/12/77	Supplement 1		
		WVY 77-86 9/16/77	Supplement 2: Response to RAI of 9/1/77		
63		WVY 77-70 8/8/77	Phase 5 Open Cycle Testing	9/30/77	A-38
		WVY 77-77 9/1/77	Supplemental Information		
64		WVY 77-67 8/4/77	Increase Circ Water ph Limit to 8.5	7/31/78	A-46
		WVY 78-7 1/30/78	Supplemental Information: Basis page for change		
		WVY 78-283 3/10/78	Calculations in support of PC (response to verbal request)		
65		WVY 77-84 9/16/77	Inservice Inspection Requirements	9/30/77	A-39
66		WVY 77-94 10/12/77	MAPLHGR Limit Uprate	11/30/77	A-41
67		WVY 77-103 11/23/77	Increase CR Scram Times	Withdrawn	

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
		6/17/83 FVY 83-62	Withdrawal letter		
68		WVY 77-100 11/7/77	Administration Organization, Corporate Organization - Westboro Changes	Denied 8/14/79 Withdrawn	
		WVY 77-116 12/29/77	Supplement1 : Removal of Corporate and Plant Organization Charts		
		WVY 80-18 1/25/80	Supplement 2: Revised Current Organizational Structure		
		WVY 80-61 4/15/80	Withdrawal of Proposed Change		
69		WVY 77-108 11/22/77	Administrative - Monthly Reporting Requirement	12/29/77	A-42
70		WVY 77-115 12/29/77	Coolant Leakage Limit/Augmented ISI Requirement (Outstanding portion cancelled by WVY 82-39, PC No. 77, Supplement 1)	6/20/78	A-45 (only includes Item III)
		FVY 86-2 1/6/86	Withdrawal of PC No. 77, Supplement 1, and unapproved portions of PC No. 70		
71		WVY 78-25 3/17/78	SRV Setpoint Requirements and Corrective Update	Withdrawn	
		FVY 87-45 4/27/87	Withdrawal of PC		
72		WVY 78-14 2/2/78	Safety-Relief Valve Surveillance Requirements	7/16/79 Canceled by NRC	
		WVY 79-107 9/19/79	Deletes PC from NRC Action Item List		
73		WVY 78-59 6/21/78	Reload 5 Licensing Submittal	10/10/78	A-47
		WVY 78-64 7/12/78	Response to RAI of 5/23/78		
		WVY 78-82 8/30/78	Correction to 7/12/78 Letter		
		WVY 78-89 9/20/78	Response to RAI of 8/31/78		
		Telecopy 10/5/78	Responses to RAI of 9/29/79		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
73		WVY 79-30 3/5/79	Supplement 1: MCPR Limits to end of Cycle 6	3/30/79	A-52
		WVY 79-58 5/9/79	APRM Gain adjustment		
[redacted]	[redacted]	[redacted]	Incorporates Physical Security Plan based on WVY 77-57 (5/25/77), WVY 78-100 (12/1/78) and letter dated 2/12/79	2/23/79	A-51
74		WVY 78-77 8/16/78	Complete Appendix B Rewrite - Nonradiological Environmental Technical Specifications	10/13/78 2/22/80	A-48 A-56
		WVY 78-83 8/31/78	Additional Information		
		WVY 79-18 2/23/79	Supplement 1: Resubmittal of Water Quality Limits		
			7/3/79 NRC Rewrite of Appendix B in new format for review		
		WVY 80-03 1/2/80	Documents EPA Acceptance 316 Demonstration Document		
		WVY 80-38 3/13/80	Submittal schedule for rewritten Technical Specification pages (see PC No. 87)		
75			This number never used	NA	NA
76		WVY 79-5 1/9/79	Containment Purging During Normal Plant Operation	Withdrawn	
		WVY 79-6 1/9/79	Additional Support for Purging Justification: Response to NRC letter (11/29/78) to cease purging during operation		
		WVY 79-148 12/27/79	Response to NRC letter 11/9/79		
		WVY 80-15 1/21/80	Supplemental Information to WVY 79-148		
		WVY 80-81 5/28/80	Response to NRC letter 3/12/80		
		WVY 80-139 10/3/80	Supplemental Information to WVY 79-148 and WVY 80-15 responding to NRC letter 11/9/79		
			NRC 10/19/81 Request for additional information		
		WVY 81-74 4/30/81	Additional response to NRC 3/12/80 letter		
		FVY 81-83 5/21/81	Response to NRC questionaire of 3/3/81		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
76			NRC Request (12/8/81) for full annunciation of overridden status of safety systems (including Purge and Vent Valves)		
		FVY 82-32 3/26/82	Modification of manner of operation of Purge and Vent Valves due to requirement to inert containment		
			NRC acceptance (5/3/82) of FVY 82-32 response question to on valve orientation		
		FVY 82-74 6/22/82	Response to NRC 5/3/82 questions		
		FVY 83-62 6/17/83	Withdrawal Notification		
77		WVY 79-09 1/30/79	Inservice Inspection and Testing Requirements		
		WVY 79-46 4/25/79	Re-request for delay in portion of ISI Program	7/9/79	A-53
		WVY 79-48 4/25/79	Licensing Fee for ISI		
		WVY 77-47 4/29/77	Delay in implementation of 10CFR50.55	4/11/79 (Denied)	
		WVY 79-51 4/30/79	ISI Program Description (Rev. 0)		
		WVY 79-70 6/25/79	ISI Program - Request for Relief		
		WVY 79-72 6/28/79	Correction for 6/25/79 Letter		
		WVY 79-122 10/23/79	ISI Program Description (Rev. 2)		
		WVY 80-75 5/14/80	ISI Program Description (Rev. 3)		
		WVY 80-92 7/1/80	ISI Program Description (Rev. 4)		
		WVY 80-142 10/10/80	ISI Program Description (Rev. 5)		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
77			NRC request (2/26/81) to meet guidelines of NUREG 0313, Rev. 1 regarding IGSCC (Generic Task A-41) (Generic Letter 81-04)		
		FVY 81-102 7/1/81	Commits to provide augmented ISI Program as required by NUREG-0313 Rev. 1		
		FVY 81-153 11/4/81	Implementation schedule for NUREG-0313, Rev. 1		
		FVY 1-177 12/28/81	ISI Program Description (Rev. 6)		
		FVY 82-39 4/8/82	Supplement 1: Augmented ISI and Leakage Detection Requirements (withdraws PC No. 70)		
		FVY 82-41 4/14/82	ISI Relief Request		
		FVY 82-48 5/3/82	ISI Relief Request		
		FVY 82-121 11/24/82	Response to NRC request (NVY 82-155, dated 9/28/82) for a Listing of IST Program Submittals		
		FVY 83-5 1/18/83	ISI Program (Rev. 7)		
			NRC Approval (NVY 83-125, dated 5/19/83) of Certain ISI Relief Requests		
		FVY 84-37 4/19/84	ISI Plans for 1984 RFO and Commitment to Submit Revised Program		
		FVY 84-139 11/27/84	ISI Program (Rev. 8)	SER 2/10/87 NVY 87-25 3/31/87 NVY 87-54 12/9/87 NVY 87-189 (Errata)	A-99 (partial)
			NRC RAI (NVY 85-220, dated 10/25/85)		
			NRC Acceptance (NVY 85-271, dated 12/19/85) of Request for Relief from requirements of 1st 10-Year ISI Program Plan, Rev. 7		
		FVY 85-124 12/30/85	Review of Section XI Requirements in ISI Program		
		FVY 86-2 1/6/86	Withdrawal of PC No. 77, Supplement. 1, and unapproved portions of PC No. 70	SER 2/10/87 NVY 87-25	

T. Silko review as part of UND 2003-101-04

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
		FVY 86-68 8/1/86	Response to NRC RAI (NVY 86-104 dated 5/27/86)	SER 2/10/87 NVY 87-25	
		FVY 86-77 8/22/86	Transmittal of GE documentation in support of FVY 86-68 Response to RAI (NVY 86-104) Question No. 8		
			NRC RAI (NVY 87-01, dated 1/5/87) on IST Program (Rev. 8)		
			NRC transmittal (NVY 87-25, dated 2/10/87) of SER for 2nd Interval ISI Program (Rev. 8)		
		FVY 87-72 7/1/87	ISI Program Revision 9		
			NRC (NVY 87-146, dated 9/11/87) schedule for meeting to discuss open items in NVY 87-01 (1/5/87)		
		FVY 88-09 2/12/88	Response to Request (NVY 87-189, dated 12/9/87) for ISI Information		
			NRC (NVY 88-074, dated 5/9/88) Summary of 10/14/87 IST Meetings		
		FVY 88-44 6/1/88	ISI Revision 9, Amendment 1		
		FVY 88-63 7/28/88	IST Revision 9		
78		WVY 79-15 2/13/79	Radiological Effluent Technical Specification (Appendix I Requirements) (Amendment A-83 effective 4/1/85)	NVY 84-224 10/9/84	83
		WVY 79-40 4/11/79	Off-Site Dose Calculation Manual (ODCM)		
			NRC Request (7/31/79) for Remittance of Class III Fee ($4,000.00)		
		FVY 83-6 1/24/83	Revised RETS		
		FVY 83-27 4/12/83	Submittal of Revised Draft RETS to Franklin Research		
		FVY 83-62 6/17/83	Requests PC be placed on hold		
		FVY 83-75 7/14/83	Submittal of Revised Draft of ODCM to Franklin Research		
		FVY 83-18 11/15/83	Schedule for submittals (RETS and ODCM)		
		FVY 83-127 12/27/83	Revises submittal dates		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
78		FVY 84-6 1/23/84	Revised RETS Program - Supersedes PC No. 78		
		FVY 84-17 3/5/84	Revised Off-Site Dose Calculation Manual (ODCM) - Supersedes 4/11/79 ODCM		
		FVY 84-122 10/12/84	Process Control Program Submittal for Review		
79		WVY 80-49 3/17/80	RPT/Analog Trip System	11/3/80	A-58
		WVY 80-72 5/9/80	Additional information		
		WVY 80-115 8/13/80	Supplemental Information Supporting RPT/Analog Trip System; response to NRC letter		
		WVY 80-134 9/23/80	Supplement 1 Information Supporting RPT/Analog Trip System		
80		5/18/79 WVY 79-63	Trip System Logic Surveillance Frequency Changes	Withdrawn	
		10/3/79 WVY 79-116	Defers submittal of response to 8/22/79 RAI		
		7/16/83 FVY 83-62	Requests PC be placed on hold		
		FVY 87-107 11/30/87	Withdrawal (see PC No. 142)		
			Requires Safeguards Contingency Plan	11/21/80	A-60
81		8/27/79 WVY 79-94	Reactor Vessel Pressure/Temperature Limitations (Appendix G)	1/14/81	A-62
		9/5/80 WVY 80-128	Supplement 1: Revised Bases Pages 117 and 118		
		1/19/81 FVY 81-12	Supplemental Information		
82		8/10/79 WVY 79-88	Fire Protection Technical Specification (Fire Brigade)	9/12/79	A-54

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
83		8/21/79 WVY 79-92	Revised MCPR Limits for Cycle 7	10/26/79	A-55
		9/18/79 WVY 79-106	Submittal of Reload 6 NEDO 24208 Report		
		10/5/79 WVY 79-114	Submittal of supplemental Reload 6 NEDO 24208 Report in response to request (9/27/79)		
84		10/5/79 WVY 79-113	Control Rod Hydraulic Return Line Isolation Valves	10/26/79	A-55
85		11/12/80 WVY 80-158	Alternative Testing Requirements (for Core Spray and LPCI systems)	7/21/89 NVY 89-153	A-114
		6/17/83 FVY 83-62	Requests PC be placed on hold		
		12/7/87 FVY 87-112	Supplement 1; surveillance testing of ECCS and SLC equipment		
		7/15/88 FVY 88-58	Response to RAI (NVY 88-077, dated 5/9/88)		
		6/8/89 BVY 89-49	Supplement 2, superceding Supplement 1		
86		4/23/80 WVY 80-65	Current Organizational Structure for VY and YAEC	4/6/81	A-65
		10/7/80 WVY 80-141	Supplement 1: Current Organizational Structure		
87		4/29/80 WVY 80-66	Appendix B Technical Specification	Withdrawn	
		6/17/83 FVY 83-62	Withdrawal Letter Appendix B to the Tech Specs were subsequently removed.		
88		8/1/80 WVY 80-110	Extension of MAPLHGR Limits	8/22/80	A-57
89		8/19/80 WVY 80-117	Reload 7 Licensing Submittal (change to MCPR and updates)	12/18/80	A-61

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
		10/7/80 WVY 80-140	Supplement 1: Additional Information Reload 7		
		11/21/80 WVY 80-162	Additional Information Reload 7: Response to NRC letter 10/30/80		
			NRC request (3/11/81) for change fee		
90		8/28/80 WVY 80-123	Hydrogen Monitoring System	11/3/80	A-58
		10/14/80 WVY 80-143	Additional Information		
91		9/12/80 WVY 80-131	SRV/SV Monitoring & STA Tech Specs (TMI-2 Lessons Learned Category "A" items)	3/2/81	A-63
		1/5/81 FVY 81-5	Supplement 1: LCOs for SRV/SV Monitoring, and Program Requirements for Integrity of Systems Outsider Containment and Iodine Monitoring; Response to NRC requests (12/1/80 and 12/10/80)		
92		2/12/81 FVY 81-28	Stability Testing	3/11/81	A-64
93		12/01/80 WVY 80-166	Common Reference Level for Reactor Water Level Instrumentation	11/16/81	A-68
		4/17/81 WVY 81-69	Extension of implementation date for Item I.K.3.27		
			NRC response (6/30/81) to WVY 81-69		
94		11/6/80 WVY 80-156	Allow Spiral Unloading and Reloading of the Reactor Core (Lowering of SRM Channel Count Rate)	11/10/80	A-59
95		10/5/81 FVY 81-144	Modifications to HPCI/RCIC Break Detection Logic and SDV Vent and Drain Valve Surveillance	11/27/81 11/29/82 NVY 82-204	A-69 A-73
		11/18/81 FVY 81-162	Response to Request (10/7/81) for Information regarding NUREG-0737, Item II.K.3.15		
96		6/30/81 FVY 81-96	Fire Protection Systems	11/10/81	A-67
		8/26/81 FVY 81-123	Supplement1		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
97		8/4/81 FVY 81-109	Organizational Changes	9/10/81	A-66
		8/25/81 FVY 81-121	Supplement 1		
98		9/2/81 FVY 81-128	Reload 8 Licensing Submittal	11/27/81	A-70
		10/28/81 FVY 81-151	Additional Information		
		10/30/81 FVY 81-152	Additional Information: Response to NRC RAI of 10/23/81		
		11/6/81 FVY 81-155	Supplemental Information (Errata sheets for FVY 81-152)		
		11/13/81 FVY 81-160	Additional Information		
		11/23/81 FVY 81-167	Validation of SIMULATE Code to support Tech Spec Change		
		11/23/81 FVY 81-168	Justification for MCPR Operating Limits BOC to EOC-2000 Mwd/t Cycle Exposure		
		3/31/82 FVY 82-36	Supplemental Information (YAEC-1299P)		
		6/24/82 FVY 82-90	Information in support of Cycle 9 Reload Analysis		
		8/19/82 FVY 82-93	Supplement (2)	9/16/82 NVY 82-146	A-72
		9/10/82 FVY 82-102	Additional Information to Supplement 2		
			Incorporates Guard Qualification and Training Program	6/9/82 NVY 82-99	A-71
99		12/29/81 FVY 81-178	High Range Noble Gas Effluent Monitor	1/29/87 NVY 87-12	A-98
100		6/2/82 FVY 82-64	Change to Inerting Technical Specification	Withdrawn	
		6/17/83 FVY 83-62	Withdrawal letter. VY tech specs for Inerting and containment atmosphere are consistent with or less restrictive than industry norms. No further action is warranted.		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
101		7/22/82 FVY 82-86	Suppressic[illegible] Tempe[illegible]nit increase to 100%	6/6/85 NVY 85-116	A-88
		7/20/83 FVY 83-76	Additional In[illegible]n subm[illegible] 'T' quencher elevation (from 7/15/83 telecon)		
102		7/22/82 FVY 82-85	Limiting C[illegible]or Opera[illegible]dby Liquid Control System	2/17/83 NVY 83-31	A-75
103		1/10/83 FVY 83-1	Analog Trip [illegible]m and Scram Disc[illegible] Volume Instrumentation	3/28/83 NVY 83-66	A-76
104		9/27/82 FVY 82-107	Organizational Changes - YNSD and Corporate Staff	2/17/83 NVY 83-31	A-75
105		8/5/83 FVY 83-88	RPS Power Protection Panel	6/2/89 NVY 89-122	A-112
		12/14/83 FVY 83-124	Response to telecon RAI on Voltage and Frequency Setpoints		
		3/4/85 FVY 85-26	Supplemental Submittal (supercedes 8/5/83 submittal)		
		5/18/89 BVY 89-45	RPS Power Protection Panel Specifications - Supporting Information and Clarifying Submittal responding to NRC letter (NVY 89-52, dated 3/24/89)		
106		12/7/82 FVY 82-129	Organizational Changes - Maintenance Superintendent	2/17/83 NVY 83-31	A-75
107		5/26/83 FVY 83-45	Reactor Pressure Vessel Temperature Curves	3/13/84 NVY 84-46A	A-81
108		5/20/83 FVY 83-41	Safety-Related Shock Suppressors (Snubbers) **(see also PC 117)**	7/9/85 NVY 85-136	A-89
		8/4/83 FVY 83-89	Supplement 1: Deletion of Certain Snubbers and Clarification		
109		2/7/84 FVY 84-7	HPCI Auto Suction Transfer	1/23/85 NVY 85-8	A-85
		5/18/84 FVY 84-47	Response to NRC RAI (NVY 84-71, dated 4/12/84)		
110		5/26/83 FVY 83-43	E-Plan Annual Drill Requirement	11/10/83 NVY 83-263	A-80

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
111		2/22/83 FVY 83-11	RPS Instrumentation (LPRM Disconnect)	4/11/83 NVY 83-83	A-78
112		2/28/83 FVY 83-14	Shift Technical Advisors	5/2/83 NVY 83-91	A-79
113		2/22/83 FVY 83-10	Spiral Unload/Reload	3/28/83 NVY 83-67	A-77
114		2/8/83 FVY 83-08	Primary Containment Isolation Valves	2/14/83 NVY 83-21	A-74
115		5/26/83 FVY 83-44	Reactor Coolant System Leakage Monitoring	Withdrawn	
			Confirmatory Order (NVY 83-150, dated 6/27/83) Requiring Implementation of Leakage Monitoring Limits Consistent with PC No. 115		
		3/29/84 FVY 84-29	Withdrawal letter. VY LCO's associated with Reactor Coolant System leakage are consistent with, our less restrictive than industry norms. No further action is required.		
116		1/23/84 FVY 84-5	Main Steam Line Low Pressure Isolation Setpoint Decrease (850 to 800 psig)	12/4/84 NVY 84-252	A-84
117		2/7/84 FVY 84-8	Revised Snubber Surveillance Criteria (See also PC 108)	7/9/85 NVY 85-136	A-89
			I&E Inspection Report 84-03 (NVY 84-63, dated 3/29/84)- Review of PC No. 117		
		7/9/84 FVY 84-87	Response to RAI (NVY 84-97, dated 5/3/84)		
		10/22/84 FVY 84-124	Request for Amendment - Safety-Related Shock Suppressors; supercedes FVY 84-8		
		11/6/84 FVY 84-133	Additional page		
118		2/7/84 FVY 84-9 Supercedes PC No. 118 (see PC 129) FVY 85-46 5/10/85	Appendix G, Reactor Vessel Pressure/Temperature Curves	Superceded	

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
119		3/26/84 FVY 84-28	Main Steam Line High Flow Setpoint Increase (120% to 140%)	2/21/85 NVY 85-29 3/19/85 NVY 85-49	A-86
		9/7/84 FVY 84-108	Response to RAI (NVY 84-137, dated 6/21/84)		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
120		6/5/84 FVY 84-58	Technical Specification Clarification - Standby Gas Treatment (SBGT) System Operability and Secondary Containment Integrity	Withdrawn	
		6/15/84 FVY 84-62	Supplement 1 - Revision		
		7/9/84 FVY 84-84	Clarification allowing refueling operations with only one EDG available		
		10/21/86 FVY 86-99	Withdrawal letter This PC was replaced by PC 236 and approved via LA # 197. The remainder of VY's LCO durations for SBGT are consistent with industry norms.		
121		6/26/84 FVY 84-76 5/30/86 FVY 86-51	Appendix J - Primary Containment Leak Rate Testing Program Response to RAI (NVY 86-29, dated 2/14/86)	The PCLRT program and LCOs for inop CIVs is consistent with industry norms. No further action is required.	
		10/10/86 FVY 86-97	Supplemental Response to RAI (NVY 86-29)		
		12/15/86 FVY 86-116	Requests expedited review decoupling RWCU V 12-68		
		4/11/89 BVY 89-35	Withdrawal of request for separate review of RWCU V 12-68		
121			Deletes requirement in 1/9/81 Order to Install an Automatic Air Dump System (responding to Request to Recind in FVY 84-13, dated 2/23/84)	8/1/84 NVY 84-175	A-82
122		11/2/84 FVY 84-129	Degraded Grid Voltage Protection System	1/29/87 NVY 87-12	A-98
		3/14/86 FVY 86/21	Clarification		

T. Silko review as part of UND 2003-101-04

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
			NRC Closeout (NVY 86-65, dated 3/31/86) of Multi-Plant Action Item B-23, Degraded Grid Voltage Protection		
		3/31/86 NVY 86-66	NRC Approval (NVY 86-66, dated 3/31/86) of Degraded Grid Procedures		
123		11/2/84 FVY 84-130	Mark 1 Containment - Technical Specification Change	NVY 85-137 7/1/85 NVY 85-164 8/7/85 (Correction)	Accepts PC w/o Amendment

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
124		12/14/84 FVY 84-146	NUREG-0737 Technical Specifications (Generic Letter 83-36)	NVY 86-167 8/11/86	A-96
		11/26/85 FVY 85-117	Supplement 1	NVY 86-255 12/19/86	Revised SE
125		1/15/85 FVY 85-3	Administrative Update (Ops. Super.)	4/1/85 NVY 85-55	A-87
126		1/15/85 FVY 85-05	Operation of Purge and Vent Valves, and Iodine Spike Limit for Reactor Coolant (response to NRC letter NVY 84-108, dated 5/22/84)	10/28/85 NVY 85-221 3/17/86 NVY 86-48 Correction	A-91
		8/2/85 FVY 85-70	Response to RAI		
127		3/27/85 FVY 85-31	Administrative Update and changes to certain trip level settings.	10/9/85 NVY 85-222 3/17/86 NVY 86-48 Correction	A-90
128		3/4/85 FVY 85-25	Administrative Changes relating to RETS	8/11/86 NVY 86-166	A-95
129		5/10/85 FVY 85-46	Reactor Vessel Pressure/Temperature Curves (supercedes PC No. 118)	6/24/86 NVY 86-121	A-93
		11/21/85 FVY 85-107	Response to RAI (from 11/7/85 telecon)		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
130		10/9/85 FVY 85-95	Deletion of Re n Syste lizer Piping Valves	3/27/86 NVY 86-60	A-92
		11/15/85 FVY 85-104	Resubmittal		
131		1/24/86 FVY 86-9	Change to RETS (Tables 3.9.2, 3.9.3 Definitions and Oil Incineration)	1/20/88 NVY 88-008	A-103
		5/13/86 FVY 86-42	Clarification	2/16/88 NVY 88-023 Correction	

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
131		6/9/86 FVY 86-53	Withdrawal of Request for Contaminated Oil Burn Portion of PC RETS is not currently a part of TS and therefore no further action is required.		
		1/16/87 FVY 87-11	Supplement 1		
		2/2/87 FVY 87-15	Errata to Supplement 1		
132		3/12/86 FVY 86/22	Single Loop Operation and Thermal-Hydraulic Stability (withdraws P.C. No. 51)	8/8/86 NVY 86-165	A-94
		3/27/86 FVY 86-24	Submittal of GE Report NEDO-30060		
		5/9/86 FVY 86-40	Clarification		
		6/9/86 FVY 86-54	Response to Request for Additional Information		
			Changes to Physical Security Plan, based on FVY 86-14 (11/26/86), FVY 86-112 (12/2/86), FVY 86-113 (12/2/86), FVY 86-114 (12/2/86), FVY 87-21 (2/12/87), FVY 87-97 (10/9/87), FVY 087-76 (10/16/87), FVY 88-02 (1/15/88), and FVY 88-14 (3/16/88)	8/25/88 NVY 88-188	A-107

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
133		4/25/86 FVY 86-34	Spent and New Fuel Storage Note: A-104;– allows rack installation and storage of up to 2000 assemblies (the current tech spec limit)A-130 grants final approval for increasing the number of stored fuel assemblies from 2000 to 2870. See also NRC SE (NVY 88-223, dated 10/14/88.)	5/20/88 NVY 88-093 7/10/91 NVY 91-144	A-104 A-130
		8/15/86 FVY 86-73	Response to NRC RAI (NVY 86-147, dated 7/24/86)		
		9/26/86 FVY 86-88	Response to NRC telecon RAI of 9/12/86 (Criticality)		
		10/21/86 FVY 86-98	Response to NRC RAI of 9/25/86 (k-infinity)		
		11/24/86 FVY 86-107	Response to NRC RAI (NVY 86-217, dated 10/22/86)		
		12/5/86 FVY 86-115	Transmittal of Supplemental Information (Proprietary Drawings)		
			NRC Transmittal (NVY 86-258, dated 12/23/86) of Hybrid Hearing Notice		
			NRC Transmittal (NVY 87-02, dated 1/5/87) regarding 12/23/86 Meeting on Heavy Loads		
			NRC Notice (NVY 87-03, dated 1/6/87) of 1/15/87 Meeting on Thermal Hydraulics		
133			NRC Summary (NVY 87-17, dated 2/8/87) of 1/15/87 meeting in Richland, WA to discuss Thermal Hydraulics		
			NRC Letter (NVY 87-32, dated 2/24/87) to Harmon and Weiss on: Rerack Meeting		
		2/25/87 FVY 87-23	Clarification of Information on Heavy Loads		
		3/19/87 FVY 87-32	Thermal-Hydraulics Information (Heat Load Calculations)		
		3/31/87 FVY 87-39	Materials Information		
		4/9/87 FVY 87-40	Justification for 150 F Temperature Limit		
		4/13/87 FVY 87-42	NES Rack Lifting Rig Design Information		
		5/22/87 FVY 87-57	PaR Rack Lifting Rig Design Information		
		6/11/87 FVY 87-65	Additional Information (Commitments)		
			NRC Notice (NVY 87-102, dated 6/30/87) of 7/14/87 Rerack Meeting		
			NRC Notification (NVY 87-111, dated 7/15/87) of continuing staff review		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
			NRC summary of (NVY 87-115, dated 7/30/87) of 7/14/87 meeting on amendment status		
		9/1/87 FVY 87-87	Response to NRC RAI (NVY 87-120, dated 8/7/87) (TAC 61351): withdraws commitment to implement license certain conditions in FVY 87-65		
		12/11/87 FVY 87-114	Clarification of information in FVY 87-87, on seismic qualification of pool makeup		
		12/16/87 FVY 87-118	Request for meeting		
			NRC Notice (NVY 88-05, dated 1/21/88) of 2/9/88 Meeting and Status Report		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
133			NRC Summary (NVY 88-027, dated 2/16/88) of 2/9/88 Meeting, with transcript		
		3/2/88 FVY 88-17	Documentation of Information presented at 2/9/88 Meeting		
			NRC Errata (NVY 88-042, dated 3/22/88) to transcript in 2/16/88 letter		
			NRC Environmental Assessment and Finding of No Significant Impact (NVY 88-145, dated 7/25/88)		
		6/7/88 FVY 88-47	Description of Enhanced Spent Fuel Pool Cooling System	10/14/88 NVY 88-223 (Safety Evaluation)	
			NRC Replacement page (NVY 88/238, dated 11/04/88) for Safety Evaluation		
		9/28/90 BVY 90-093	Supplemental Information on Standby Fuel Pool Cooling System		
134		8/26/86 FVY 86-80	Analog Equipment Replacement	3/29/89 NVY 89-62	A-110
		1/19/90 BVY 90-005	Change to Tech Spec 3.7.5.b Approved in Amendment No. 110	2/5/90 NVY 90-019	A-119
135		8/28/86 FVY 86-78	86/87 Operating Cycle Inspection/Repairs of RHR Pump Impeller Wear Rings (one time basis)	12/4/86 NVY 86-237	A-97
		11/3/86 FVY 86-102	Response to Request for Information (telecons of 10/20/86 and 10/22/86)		
136		1/16/87 FVY 87-10	SLC System (Testing)	12/30/87 NVY 87-199	A-102
137		1/12/87 FVY 87-08	Post-Accident Instrumentation	6/22/89 NVY 89-135	A-113

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
		1/29/88 FVY 88-08	Supplement 1		
138		4/28/87 FVY 87-48	Administrative Changes (Chem/H.P. and Procedures Approval Process)	12/29/87 NVY 87-196	A-101
		11/2/87 FVY 87-102	Clarification		
No PC No. used		5/29/87 FVY 87-59	Clarification of Reactivity Shutdown Margin Demonstration (doesn't require License Amendment)	8/21/87 NVY 87-133	No Amendment Number
139		6/24/87 FVY 87-67	Change for Cycle 13 Operating Limits	9/18/87 NVY 87-148	A-100
		8/11/87 FVY 87-78	Clarification of VY Cycle 13 Core Performance Analysis Report		
140		Not Submitted	FVY 87-87, dated 9/1/87, withdraws commitment re: Spent Fuel Pool Cooling System License Conditions. This PC was not submitted to the NRC. This is appropriate as no specs are required for the subject system. Since no specs exist, they are not restrictive and no further action is warranted.		
141		4/27/89 BVY 89-41	Construction Period Recapture		
		6/23/89 BVY 89-55	Revision of requested end of license date	12/17/90 NVY 90-217	A-127
142		11/30/87 FVY 87-107	Logic System Functional Test Intervals; withdraws PC No. 80	8/9/88 NVY 88-170	A-106
		1/20/88 FVY 88-04	Clarification	9/1/88 NVY 88-189 Errata	A-106
		3/10/88 NVY 88-036	NRC Notice (NVY 88-036, dated 3/10/88) of 3/15/88 Meeting to discuss Logic System Reliability		
		3/17/88 NVY 88-044	NRC Summary of (NVY 88-044, dated 3/17/99) 3/15/88 meeting		
		4/13/88 FVY 88-028	Additional Information regarding Relay Reliability		
		4/27/88 NVY 88-069	NRC letter (NVY 88-069, dated 4/27/88) to State of Vermont, "Logic System Functional Test at Vermont Yankee"		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
143		12/9/87 FVY 87-117	Automatic Depressurization System (Logic Modification)	8/4/88 NVY 88-155	A-105
144		5/23/88 FVY 88-40	New Fuel Type (16 gram limit) GE 8X8EB Fuel	9/9/88 NVY 88-198	A-108
		8/15/88 FVY 88-66	Response to Request for Supporting Document (NEDE-21697 supplement 1)		
145		11/18/88 FVY 88-98	Generic Letter 83-02: NUREG-0737 Technical Specifications, Items II.K.3.13 and II.K.3.22	4/24/89 NVY 89-84	A-111
146		11/30/88 FVY 88-99	Incorporates 1.04 Fuel Cladding Integrity Safety Limit	2/27/89 NVY 89-37	A-109
		12/21/88 FVY 88-103	Clarification		
		1/6/89 BVY 2/89	Further Clarification		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
147		5/12/89 BVY 89-44	Change regarding ATWS Rule (10CFR50.62) in response to NRC request (NVY 87-04, dated 1/8/87)	8/12/97 NVY 97-129 (Notice of Withdrawal)	WITHDRAWN
			NRC summary (NVY 92-208, dated 11/12/92) of 8/15/92 Meeting and clarification of NRC letter (NVY 92-96, dated 6/5/92)concerning open issue on signal conditioning		
147		4/14/93 BVY 93-40	Commitment to implement plant mods to meet ATWS rule		
		10/22/93 BVY 93-119	Updated Technical Specification pages responding to NRC request (NVY 92-96, dated 6/5/92)		
		7/25/97 BVY 97-95	Withdrawal letter deferring changes to ITS A review of this WITHDRAWN PC reveals that most of the TS changes contained within this submittal have worked their way into the specs. WRT to the issue of "outlier restrictive TS changes," the LCO periods contained with in VY's specs are less restrictive than that of STS. Accordingly, no further action is required.		
148		1/27/89 BVY 89-10	Administrative Changes	8/24/90 NVY 90-167	A-126

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
		11/28/89 BVY 89-109	Clarification		
149		2/2/89 BVY 89-14	Primary Containment Isolation Valve Testing in Head Spray Subsystem of RHR	9/7/89 NVY 89-187 10/10/89 NVY 89-213 SE Correction	A-115
150		5/12/89 BVY 89-43	Elimination of Cycle-Specific Parameter Limits Generic Letter 88-16	9/15/89 NVY 89-204	A-116
		7/14/89 BVY 89-67	Clarification removes FCSIL parameter from PC	10/10/89 NVY 89-212 Correction	
151		10/16/89 BVY 89-97	Compensatory Fire Watch frequency reduction	12/8/89 NVY 89-250	A-117
152		11/10/89 BVY 89-113	Revise Reactor Vessel Pressure-Temperature Curves (Generic Letter 88-11)	4/17/90 NVY 90-077	A-120
153		11/9/89 BVY 89-106	Emergency Change Request for 1989/1990 Operating Cycle Refurbishment/Repair of Uninterruptible Power Supply System	1/26/90 NVY 90-008	A-118
			NRC Temporary Waiver of Compliance (NVY 89-224, dated 11/9/89) from Technical Specification Section 3.5.A.4		
154		3/9/90 BVY 90-029	Type C Leakage Testing of New Inboard FW Check Valves	6/4/90 NVY 90-121	A-122
155		2/28/90 BVY 90-021	Utilization of Alternative Longer Life Control Blades	6/5/90 NVY 90-127	A-123
156		3/5/90 BVY 90-022	Removal of 3.25 Limit on Extending Surveillance Intervals	7/2/90 NVY 90-138	A-124
		6/7/90 BVY 90-066	Replacement of Uninterruptible Power Supply (incorporated into A-124)		
157		3/2/90 BVY 90-023	Administrative Update	4/25/90 NVY 90-91	A-121
158		6/11/90 BVY 90-069	Update Section 6.0, "Administrative Controls"	8/24/90 NVY 90-167	A-126
159		7/20/90 BVY 90-081	Surveillance Testing of Engineered Safeguards Equipment	3/4/91 NVY 91-38	A-128

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
160		4/8/90 BVY 90-044	Auxiliary Electrical Power System Technical Specification	8/23/90 NVY 90-161	A-125
		8/3/94 BVY 94-077	Correction of SER accompanying A-125	8/22/95 NVY 95-113 Correction to SE	
161		6/1/90 BVY 90-068	Corrects Typographical and Format Inconsistencies	10/7/91 NVY 91-183	A-131
		7/17/90 BVY 90-078	List of References (omitted from BVY 90-068)		
162		1/15/91 BVY 91-02	Toxic Gas Monitoring System NRC RAI (NVY 91-56, dated 4/16/91) .... VY Response to RAI (BVY 91-53, dated 5/16/91) VY response to 2nd RAI (BVY 91-65, dated 7/12/91) – 2nd RAI via 7/3/91 telecon	10/24/91 NVY 91-205 11/7/91 NVY 91-206 Corrections to A-132 and SE	A-132
163		1/15/91 BVY 91-03	Surveillance of Indication of LPCI Crosstie Monitor (Valve RHR-20)	3/25/91 NVY 91-69	A-129

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
164		12/27/91 BVY 91-125	Administrative Changes regarding Plant Operations Review Committee and testing of Primary Containment Isolation Valves	7/21/92 NVY 92-136	A-134 (partial)
			Denial of that part of PC No. 164 dealing with Plant Operations Review Committee and approval of reinstatement of portion dealing with testing primary containment isolation valves		
165		12/23/91 BVY 91-120	Analog System Replacement NRC RAI (NVY 92-35, dated 3/3/92) .... VY Response to RAI (BVY 92-41, dated 3/31/92)	5/8/92 NVY 92-097	A-133
166		12/15/92 BVY 82-139	One-Time Extended Emergency Diesel Generator (EDG) LCO Period to Support Maintenance Activities.	3/25/93 NVY 93-59 Notice of	

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
		3/9/96 BVY 93-025	Withdrawal letter. The issue of extending the EDG LCO from 7 days to 14 days has been identified on the PC list as a wish list items for over 5 years. Periodically during the Monthly Licensing Meeting, the desire for this PC is discussed. Thus far, the consensus has been that that this change is not warranted.	withdrawal	
167		12/15/92 BVY 92-140	Calibration Requirements for Control Rod Block Instrumentation	8/25/93 NVY 93-132	A-136
168		8/4/93 BVY 93-30	Auxiliary Power System Tech Specs and Associated Revision to IST Program	3/22/94 NVY 94-45	A-138
169		3/26/93 BVY 93-29	Updates Section 6.0, "Administrative Controls" NRC Reissue (NVY 93-088, dated 5/28/93)of A-135 due to failure to put Amendment No. on TS pages NRC transmittal (NVY 93-089, dated 6/11/93) of TS pages omitted from NVY 93-088	5/26/93 NVY 93-87	A-135
170		8/27/93 BVY 93-81	Revisions relating to 10CFR20 Provides FSAR Figure 2.2-5 to assist NRC in review (BVY 93-124, dated 11/9/93) Response to comments in telecon of 8/10/95 (BVY 96-53, dated 4/26/96) Response to comments in telecon of 9/25/96 (BVY 96-111, dated 9/25/96)	4/3/95 NVY 95-48 6/19/97 NVY 97-92	A-144 (Non-Part 20 portion only) A-151 (Part 20 portion)
171		7/14/93 BVY 93-068	Additions in Response to Generic Letter 88-01 on Intergranular Stress Corrosion Cracking	6/1/94 NVY 94-86	A-139
172		6/25/93 BVY 93-063	Core Alteration Definition	9/3/93 NVY 93-144	A-137
173		3/31/94 BVY 94-36	BWR Thermal-Hydraulic Stability and Plant Information Requirements for BWROG Option 1-D Long Term Stability Solution NRC RAI (NVY 94-84 dated 6/9/94)......... VY Response (BVY 94-90, dated 9/9/94) NRC Acceptance (NVY 95-43, dated 3/30/95) of Report submitted with BVY 93-72 (dated 7/7/93) Application of BWROG Thermal Hydraulic Stability Long-Term Solution Optioin I-D Submittal of Updated TS Pages (BVY 95-70, dated 6/22/95)	8/9/95 NVY 95-106	A-146
174		10/28/94 BVY 94-103	Removal of Neutron Flux Instrumentation from Post-Accident Monitoring Technical Specifications	6/20/95 NVY 95-84	A-145
175		12/6/93 BVY 93-134	Revisions relating to Jet Pump Surveillance Requirement	10/26/94 NVY 94-190	A-141
176		5/20/94 BVY 94-51	Removal of Core Spray High Sparger Pressure Instrumentation from Emergency Core Cooling System Actuation Instrumentation	8/22/94 NVY 94-134	A-140

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
177		12/8/94 BVY 94-105	Standby Gas Treatment Power Supply Requirements During Refueling Operations Clarification (BVY 95-20, dated 2/16/95)	3/23/95 NVY 95-47	A-143
178		11/7/94 BVY 94-109	Diesel Fuel Oil Procurement and Testing using ASTM D975(1993)	**WITHDRAWN**	**WITHDRAWN**
		2/2/95 BVY 95-15	Withdrawal letter This issue was subsequently submitted as part of PC 256 and approved as part of LA # 214. No further action required.		
179		12/14/94 BVY 94-123	Instrument Identification Change for ECCS Actuation Instrumentation (ATWS Diversity)	3/3/95 NVY 95-15	A-142
180	J. Meyer	2/5/96 BVY 96-06 7/14/98 BVY 98-105	Administrative Change to Correct Typographical Error and Text Inconsistencies Withdrawal letter	Withdrawn 7/23/98 NVY 98-102 Notice of Withdrawal	N/A
181	J. Meyer	8/22/96 BVY 96-99	High Range Stack Noble Gas Monitor Action Statement	4/8/98 NVY 98-50	A-158
182	N/A	Not Submitted	Minimum Core Cooling System Availability During Cold Shutdown and Refueling Conditions - Incorporated into ITS effort	Not Submitted	
183	J. Meyer	4/4/96 BVY 96-37	Control Rod Over-Travel Indication Surveillance	9/30/96 NVY 96-155	A-149
		10/17/96 BVY 96-126B	Updates approved Technical Specification pages to reflect effect of Amendments 148 and 149		
184	J. Meyer	4/4/96 BVY 96-39	Secondary Containment Integrity Requirements	7/10/96 NVY 96-123	A-147
185	J. Meyer	9/11/96 BVY 96-104	Safety and Relief Valve Setpoint Tolerance Increase and Power Operation with an inoperable SRV .... Proposed schedule extension (BVY 97-14, dated 11/6/97) for responses to RAI (NVY 97-151, dated 10/7/97) .... RAI Response submitted on 12/8/97, BVY 97-164	4/15/98 NVY 98-55	A-160
186	J. Meyer	8/9/96 BVY 96-98	Safety Limit Minimum Critical Power Ratio	10/4/96 NVY 96-154	A-150
		9/17/96 BVY 96-109	Revision		
187	J. Meyer	6/28/96 BVY 96-84	Core Shutdown Margin	9/25/96 NVY 96-150	A-148

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
		10/17/96 BVY 96-126B	Updates approved Technical Specification page to reflect effect of *Amendments* 148 and 149		
188	J. Meyer	10/11/96 BVY 96-120	Revised Recirculation Motor Generator Set Fire Protection Foam System Capacity	3/31/98 NVY 98-46	A-156
189	J. Meyer	12/10/96 BVY 96-155	Relocation of Fire Protection Requirements from Technical Specifications to Fire Protection Plan and UFSAR	2/24/99 NVY 99-21	A-168
		1/22/99 BVY 99-04	Revised pages		
190	J. Meyer	7/11/97 BVY 97-90	10 CFR 50, Appendix J, Option B	2/26/98 NVY 98-24	A-152
		11/21/97 BVY 97-154	Modification (Revised Pages)	3/10/99 NVY 99-28 Correction	
		12/22/97 BVY 97-170	Additional corrected pages		
		2/16/98 BVY 98-18	Option B Modification		
		3/3/98 BVY 98-32	Final results of Core Monitoring relating to MCPR calculations		
191	J. Meyer	6/9/97 BVY 97-77	Revises Section 6 to reference NRC Approved Methodology for Thermal-Hydraulic Stability (LAPUR5)	4/7/98 NVY 98-51	A-157
192	J. Meyer	8/20/97 BVY 97-107	Adds reference to FIBWR2 Method	2/23/99 NVY 99-20	A-167
		9/18/97 BVY 97-118	Marked-up Tech Spec pages for PC Nos. 192 and 193		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
193	J. Meyer	8/22/97 BVY 97-106	Revision of CO_2 System Technical Specifications	3/6/98 NVY 98-26	A-154
		9/18/97 BVY 97-118	Marked-up Tech Specs pages for PC Nos.. 192 and 193		

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
194	J. Meyer	10/10/97 BVY 97-130	Offsite Power System Technical Specifications	3/24/98 NVY 98-39	A-155
195	J. Meyer	11/20/97 BVY 97-155	Revision of Main Station (Spare) 125v Battery Charger	3/5/98 NVY 98-25	A-153
196	J. Meyer	12/11/97 BVY 97-165	Safety Limit MCPR for Cycle 20	4/10/98 NVY 98-56	A-159
		3/3/98 BVY 98-32	Final results of Core Monitoring relating to MCPR calculations		
197	H. Heilman	11/2/98 BVY 98-16	CS/LPCI Pump Start Time Delay (Table 3.2.1)	4/26/99 NVY 99-44	A-170
198	J. Meyer	9/21/99 BVY 99-119	EDG Fuel Oil Storage Tank Minimum Volume	11/22/99 NVY 99-114	A-180
199	H. Heilman	6/30/98 BVY 98-15	CS/LPCI Aux Power Monitor (Table 4.2.1)	9/1/98 NVY 98-127	A-162
200	J. Stanton	4/23/98 BVY 98-52	Service Water/Alternate Cooling Tower System	3/11/99 NVY 99-30	A-169
		1/25/99 BVY 99-11	Duplication of applicable Tech Spec pages into Technical Requirements Manual		
201	T. Silko	3/20/98 BVY 98-43	Containment Purge and Vent	5/14/98 NVY 98-71	A-161
202	J. Meyer	5/1/98 BVY 98-63	Administrative Change to Section 6.0	Superceded By PC-208 PC 208 approved via LA# 171. No further action required.	**WITHDRAWN**
		5/8/98 BVY 98-70	Revision to Tech Spec page		
		2/1/99 BVY 99-20	Supercedes PC No. 202		
203	J. Meyer	5/26/99 BVY 99-75	Proposed Change No. 203 - Suppression Pool Water Temperature Surveillance	8/30/99 NVY 99-82	A-174

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
204	T. Silko	5/8/98 BVY 98-69	Maximum Torus Water Temperature	12/28/98 NVY 98-170 1/21/99 NVY 99-06 SE Correction	A-163
		7/10/98 BVY 98-102	Calculation to support change		
205	L. Gucwa	11/3/98 BVY 98-118	Administrative Change to TS *(This PC replaces PC 180)*	1/5/99 NVY 99-02	A-164
		12/15/98 BVY 98-167	Correction		
206	L. Gucwa	12/10/98 BVY 98-119	Calibration of Hydrogen Monitors (Offgas)	2/12/99 NVY 99-14	A-166
207	T. Silko	9/4/98 BVY 98-130	Proposed Change No. 207 – Increased Spent Fuel Assembly Storage Capacity	12/21/99 NVY 99-124	A-182
		2/8/99 BVY 99-19	Supplement		
208	L. Gucwa	2/1/99 BVY 99-20	Proposed Change 208 - TS Section 6.0 Rewrite (Replaces PC 202.)	7/19/99 NVY 99-69	A-171
209	W. Limberger	12/4/98 BVY 98-162	Proposed Change 209 Intermittent Opening of Primary (Replaced by PC-210) PC 210 approved via LA #165. No further action required.	**WITHDRAWN**	**WITHDRAWN**
210	W. Limberger	12/11/98 BVY 98-165	Proposed Change No. 210 – Intermittent Opening of Manual Primary Containment Isolation Valves (Withdrawal of PC-209)	1/19/99 NVY 99-04	A-165
211	D. Pendry	4/20/99 BVY 99-58	Proposed Change No. 211 – Spiral Reload (Replaced by PC-223) PC 223 approved via LA #181. No further action required.	**WITHDRAWN**	**WITHDRAWN**
212	L. Gucwa	5/5/99 BVY 99-69	Proposed Change No. 212 – ATWS Rule (10CFR50.62) / Standby Liquid Control System	9/17/99 NVY 99-85	A-175
213	W. Limberger	4/16/99 BVY 99-13	Proposed Change No. 213 – *Generic Letter* 88-01 and Use of Code Case N560	8/13/99 NVY 99-76	A-172
214	L. Gucwa	6/24/99 BVY 99-85	Proposed Change No. 214 – TCV and TSV Closure Scram Bypass	8/13/99 NVY 99-75	A-173

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
215	D. Pendry	5/6/99 BVY 99-68	Proposed Change No. 215 - Removal of Main Steam Isolation Valve Leakage (Replaced by PC-220) PC 220 approved via LA#178. No further action required.	**WITHDRAWN**	**WITHDRAWN**
216			NOT USED		
217	B. Drews	6/15/99 BVY 99-76	Proposed Change No. 217 – Surveillance Test Interval / Allowable Out-Of-Service Time (BVY 99-161 Supplement)	4/3/00 NVY 00-35	A-186
218	D. Pendry	7/20/99 BVY 99-82	Proposed Change No. 218 - Increased Core Flow	4/25/00 NVY 00-42	A-187
219	L. Gucwa	7/20/99 BVY 99-86	Proposed Change No. 219 - High Pressure Cooling (HPCI/RCIC) and ADS Operability	10/1/99 NVY 99-90	A-177
220	D. Pendry	6/29/99 BVY 99-83	Proposed Change No. 220 – Main Steam Line Isolation Valve Leakage (Replacement for 215)	10/1/99 NVY 99-91	A-178
221	J. Meyer	7/12/99 BVY 99-91	Proposed Change No. 221 – SLMCPR Revision	9/21/99 NVY 99-87	A-176
222	W. Limberger	5/22/00 BVY 00-25	Proposed Change No. 222 – Inservice Inspection of Class MC Components	7/19/00 NVY 00-66	A-192
223	D. Pendry	8/18/99 BVY 99-104	Proposed Change No. 223 – Spiral Core Loading Around a Source Range Monitor (Replacement for 211)	12/14/99 NVY 99-120	A-181
224	L. Gucwa	11/5/99 BVY 99-139	Proposed Change No. 224 – Reactor Power Distribution Limits Applicability	6/21/00 NVY 00-60	A-188
225	L. Gucwa	8/18/99 BVY 99-106	Proposed Change No. 225 - Missed Technical Specifications Surveillance	10/13/99 NVY 99-98	A-179
226	L. Gucwa	12/21/99 BVY 99-160	Proposed Change No. 226 – Control Rod Block Instrumentation (Withdrawn per BVY 00-115) Resubmitted as part of PC 247 – approved via LA# 211. No further action warranted.	**WITHDRAWN**	**WITHDRAWN**
227	J. Meyer	10/18/99 BVY 99-132	Proposed Change No. 227 – Revised SBGT Charcoal Testing Standard	7/11/00 NVY 00-62	A-189
228	L. Gucwa	10/21/99 BVY 99-134	Proposed Change No. 228 – Administrative Change	1/11/00 NVY 00-14	A-183
229	L. Gucwa	12/14/99 BVY 99-159	Proposed Change No. 229 – Relocation of Radiological Effluent Tech Specs (RETS)	8/24/00 NVY 00-87	A-193
230	W. Limberger	1/20/00 BVY 00-11	Proposed Change No. 230 - Testing of Augmented Off-Gas (AOG) Instrumentation	3/6/00 NVY 00-24	A-184

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
231	L. Gucwa	2/11/00 BVY 00-20	Proposed Change No. 231 - Main Steam Isolation Valve Surveillance Requirements	3/9/00 NVY 00-25	A-185
232			NOT USED		
233	L. Gucwa	5/23/00 BVY 00-47	Proposed Change No. 233 – LPRM Calibration Frequency	7/18/00 NVY 00-65	A-191
234	L. Gucwa	5/23/00 BVY 00-48	Proposed Change No. 234 – Reactor Coolant Chemistry – Conductivity and Chlorides	7/18/00 NVY 00-64	A-190
235	L. Gucwa	8/10/00 BVY 00-70	Proposed Change No. 235 - ECCS Requirements During Refueling	11/17/00 NVY 00-101	A-195
236	L. Gucwa	9/26/00 BVY 00-88	Proposed Change No. 236 - Standby Gas Treatment System Operability During Refueling	3/23/01 NVY 01-18	A197
237	B. Hobbs	9/14/00 BVY 00-84	Proposed Change No. 237 – Table 4.7.2 Notes	10/31/00 NVY 00-108	A-194
238	J. Meyer	10/25/00 BVY 00-97	Proposed Change No. 238 – Administrative Changes	1/23/01 NVY 01-10	A-196
239	T. Silko	11/30/00 BVY 00-90	Proposed Change No. 239 - Refueling Interlocks	4/20/01 NVY 01-40	A-200
240	B. Hobbs	11/1/00 BVY 00-101	Proposed Change No. 240 - High Pressure Core Cooling Systems Isolation Function	4/20/01 NVY 01-42	A-202
241	J. Meyer	10/25/00 BVY 00-98	Proposed Change No. 241 – 125 Vdc Battery Chargers	3/27/01 NVY 01-25	A-198
242	J. Meyer	11/27/00 BVY 00-107	Proposed Change No. 242 – 24 Vdc ECCS Battery Removal	4/20/01 NVY 01-41	A-201
243	L. Gucwa	12/7/00 BVY 00-112	Proposed Change No. 243 – LPCI Operability During Shutdown	3/30/01 NVY 01-22	A-199
244	T. Silko	12/19/00 BVY 00-113	Proposed Change No. 244 – Revised P/T Limit Curves	5/4/01 NVY 01-46	A-203
245	L. Gucwa	4/17/01 BVY 01-33	Proposed Change No. 245 - Post-Accident Monitoring Instrumentation	10/2/01 NVY 01-98	A-204
246	B. Hobbs	4/23/01 BVY 01-31	Proposed Change No. 246 - Administrative Changes to the Facility Operating License	10/22/01 NVY 01-106	A-206
247	L. Gucwa	6/21/01 BVY 01-51	Proposed Change No. 247 – Control Rod Block Instrumentation	8/27/02 NVY 02-77	A-211

Proposed Change No.	Initiated by	Date Submitted Letter No.	Title	Date Approved Letter No.	NRC Approval (A: Lic. Amend) (C: TS Change)
248	L. Gucwa	8/20/01 BVY 01-65	Proposed Change No. 248 – Elimination of Alternate Train Testing	8/14/02 NVY 02-71	A-209
249	L. Gucwa	8/14/01 BVY 01-64	Proposed Change No. 249 – HPCI and RCIC LCO Extension to 14 days	10/18/01 NVY 01-100	A-205
250	L. Gucwa	3/19/02 BVY 02-18	Proposed Change No. 250 - Elimination of the Main Steam Isolation Valve Closure and Scram Functions of the Main Steam Line Radiation Monitors	9/18/02 NVY 02-89	A-212
251	J. Meyer	11/20/01 BVY 01-85	Proposed Change No. 251 – Table 4.7.2, SBGT Heater and Miscellaneous Admin Changes	8/21/02 NVY 02-74	A-210
252	L. Gucwa	11/20/01 BVY 01-86	Proposed Change No. 252 - Allowed Outage Times for PAM Instrumentation	5/10/02 NVY 02-39	A-207
253	T. Silko	11/4/02 BVY 02-62	Proposed Change No. 253 – ILRT Interval Extension		
254	D. Green	2/26/02 BVY 02-12	Proposed Change No. 254 - Definition of Operable	2/4/03 NVY 03-14	A-213
255	L. Gucwa	1/9/03 BVY 03-04	Proposed Change No. 255- Definition of LSFT		
256	R. Daflucas	12/10/02 BVY 02-95	Proposed Change No. 256 – Admin Change to update Titles in Section 6.0 and Table of Contents and EDG Fuel Oil Specification	2/27/03	A-214
257	J. Devincentis		Proposed Change No. 257 – ARTS/MELLLA		
258	L. Gucwa		Proposed Change No. 258 - Reactor Pressure Vessel Integrated Surveillance Program		
259	L. Gucwa		Proposed Change No. 259 – Instrumentation Tech Specs		

Entergy

Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Operations, Inc.
185 Old Ferry Road
Brattleboro, VT 05302-0500

March 26, 2003
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

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

Docket No. 50-271
BVY 03-29

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. Pressure-Temperature Limitations

update current pressure and temperature (P-T) limit curves for the reactor ...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 ...ves are valid through the end of the current operating license or 32 effective ...PY) and generally satisfy the requirements of Appendix G to 10CFR Part 50, "Fracture ...ness 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 *ıstrumentation in lieu of process instrumentation for the downcomer region fluid :mperature and permanent flange region outside surface temperature. The test :strumentation uncertainty must be less than +/- 2°F. The flange region temperatures ·ust be maintained greater than or equal to 72°F when monitored with test ·strumentation during tensioning, detensioning, and when tensioned.*

- :e tabulation of pressure and temperature data on Figure 3.6.3 is being revised to more :urately reflect the plot of the curves (the curves are unchanged). At 116°F the bottom :d pressure is changed to 4:3 psig, instead of the current 416 psig. At 120°F, there ·uld be only two data points on Figure 3.6.3, and these are at 253 psig for the upper :on and at 439 psig for the bottom head region. Therefore, the tabulation ·esponding to a temperature of 120°F will only specify pressures of 439 psig and 253 · for the bottom head region and upper region, respectively.

1.3 :EDULE

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

2.0 B KGROUND

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

2.1 RI MATERIAL SURVEILLANCE PROGRAM

Licensees of clear power plants are required by Appendix H to 10CFR Part 50 to implement RPV materi urveillance programs (including the withdrawal and analysis of surveillance capsules) for nitoring changes in the fracture toughness properties of ferritic materials in the reactor vesse ltline region which result from neutron irradiation. These programs consist of surveillance ules installed inside the RPV that include specimens from RPV plate, weld and heat-affected e materials. These specimens are removed at periodic intervals, tested and analyzed to m or the radiation embrittlement of the RPV. Appendix H provides two alternative methods for c liance:

The first altern e 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 [illegible] ial surveillance programs do not need to be included in the TS, because there would be [illegible] n of controls that have been established by regulations (i.e., Appendix H to 10CFR50). [illegible], 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 [illegible] RPV surveillance program to the UFSAR.

VY [illegible] questing 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 x 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 $\geq$ 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

1. NRC letter from W. H. Bateman to C. Terry (BWRVIP Chairman), "Safety Evaluation Regarding EPRI Proprietary Reports 'BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan (BWRVIP-78)' and 'BWRVIP-86: BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan,'" February 1, 2002

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

5. Boiling Water Reactor Vessel and Internals Project report, "BWRVIP-86-A: BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," October 2002

6. NRC letter from W.H. Bateman to C. Terry (BWRVIP Chairman), "NRC Staff Review of BWRVIP-86-A, "BWR Vessel and Internals Project, Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," December 16, 2002

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

10. NRC Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, May 1988

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-T CURVES FOR VERMONT YANKEE NUCLEAR POWER STATION

Prepared by: Enrico J Betti 3/18/2003
Enrico J. Betti, PE
Senior Mechanical/Structural Engineer

Reviewed by: Michael Selling 3/18/03
Michael Selling
Mechanical/Structural Engineer

Reviewed by: 3/18/03
James Fitzpatrick
Senior Lead Mechanical/Structural Engineer

Approved by: S D Goodwin 3-19-03
Scott Goodwin
Mechanical/Structural Engineering Supervisor

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 1×10^{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, Appro ved February 26, 1999.

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

4. U. S ode of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Re ments," December 1995.

5. anch 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, <u>Vermont Yankee Reactor Vessel</u>, 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 Informatio[illegible] 2001.

20. [illegible] ear Regulatory Commission, NVY 01-046 to Vermont Yankee Nuclear Power Corporation, "Issuance of Amendment RE: P/T Limit Curves (TAC No. MB0764) (Tac No. M[illegible]0764).

21. Reg[illegible] 190, "Calculational and Dosimetry Methods for Determining Pres[illegible] Ves[illegible] [illegible]tron Fluence," U.S. NRC, March 2001.

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

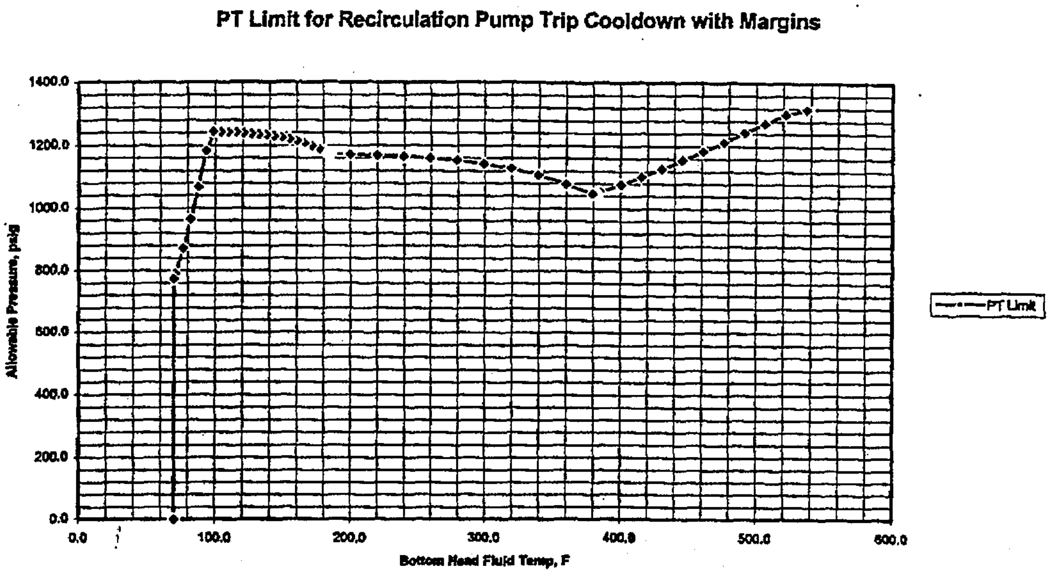


Figure 1: Bottom Head Recirculation Pump Trip Pressure/Temperature Limit Curve

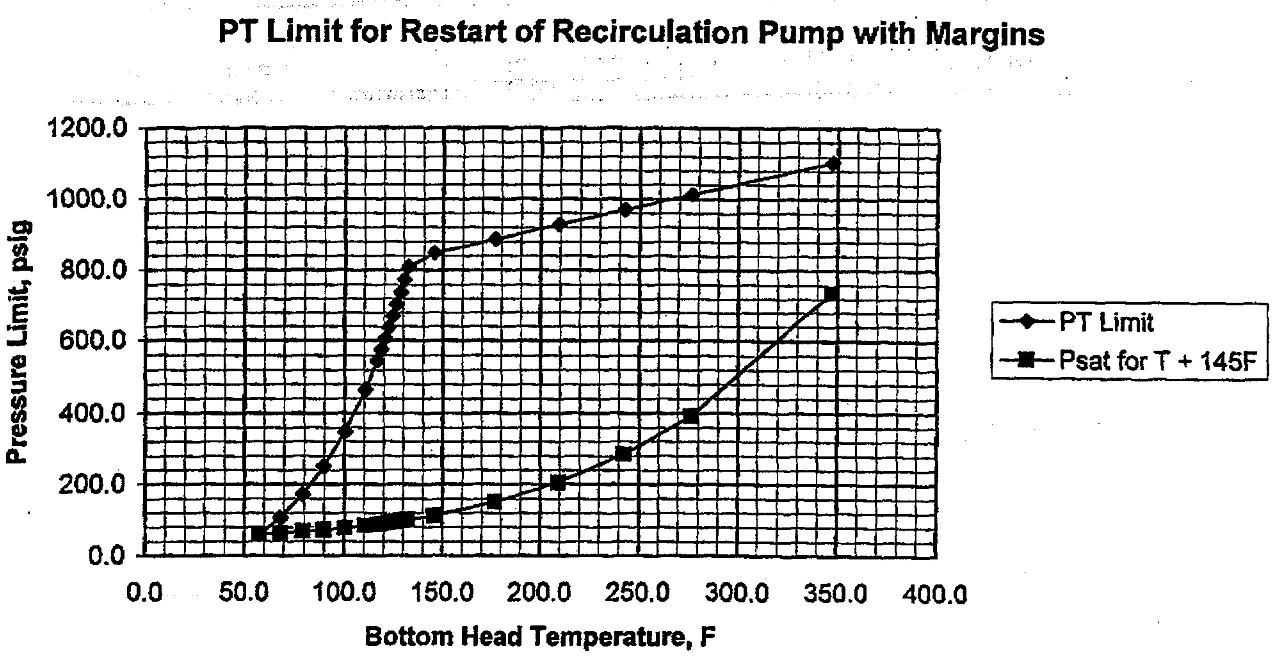
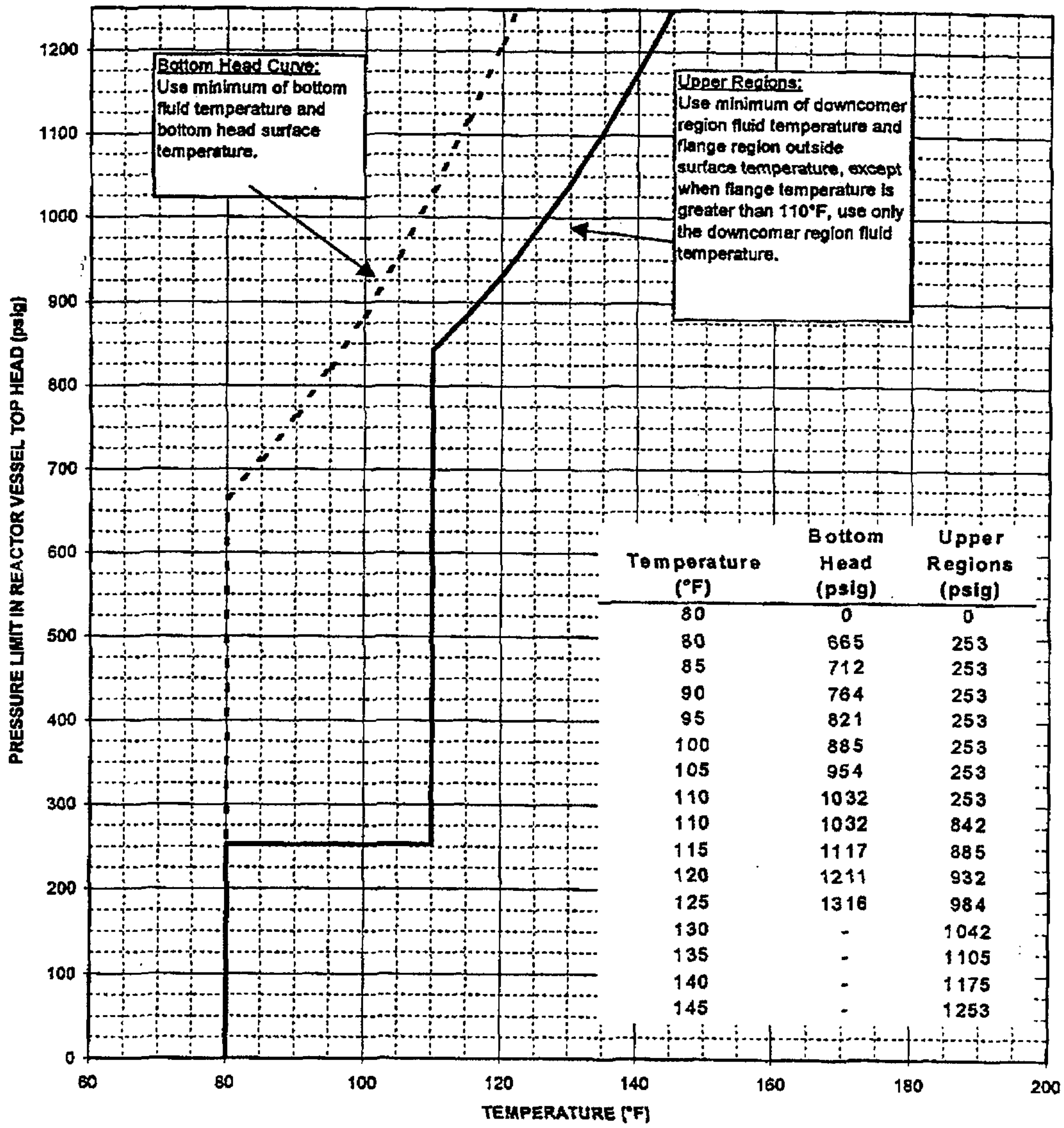


Figure 2: Pressure/Temperature Limit Curve for Recirculation Pump Start

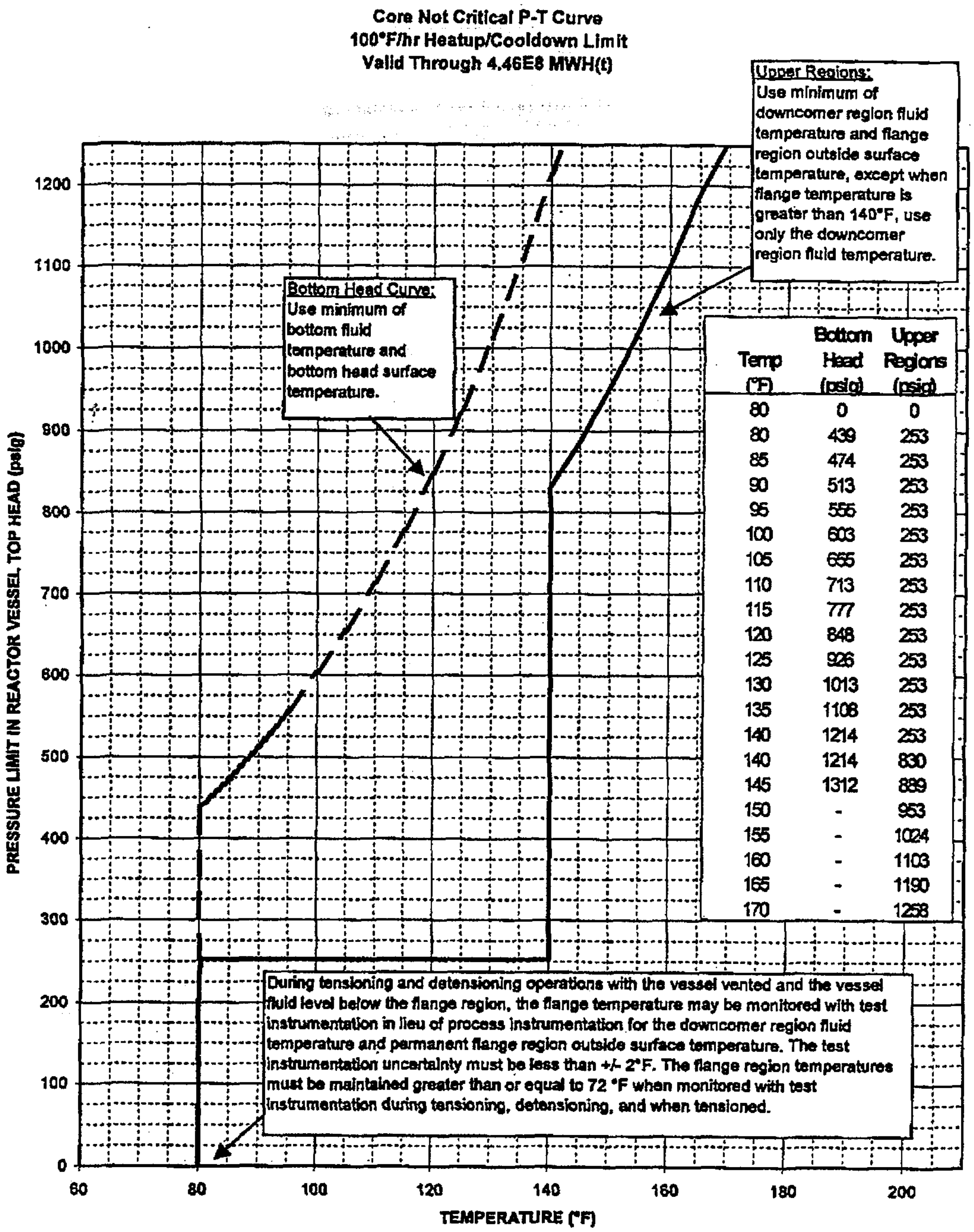
Leak Test and Hydro P-T Curve
40°F/hr Heatup/Cooldown Limit
Valid Through 4.46E8 MWH(t)



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: Pressure Test P-T Curve (Curve A)

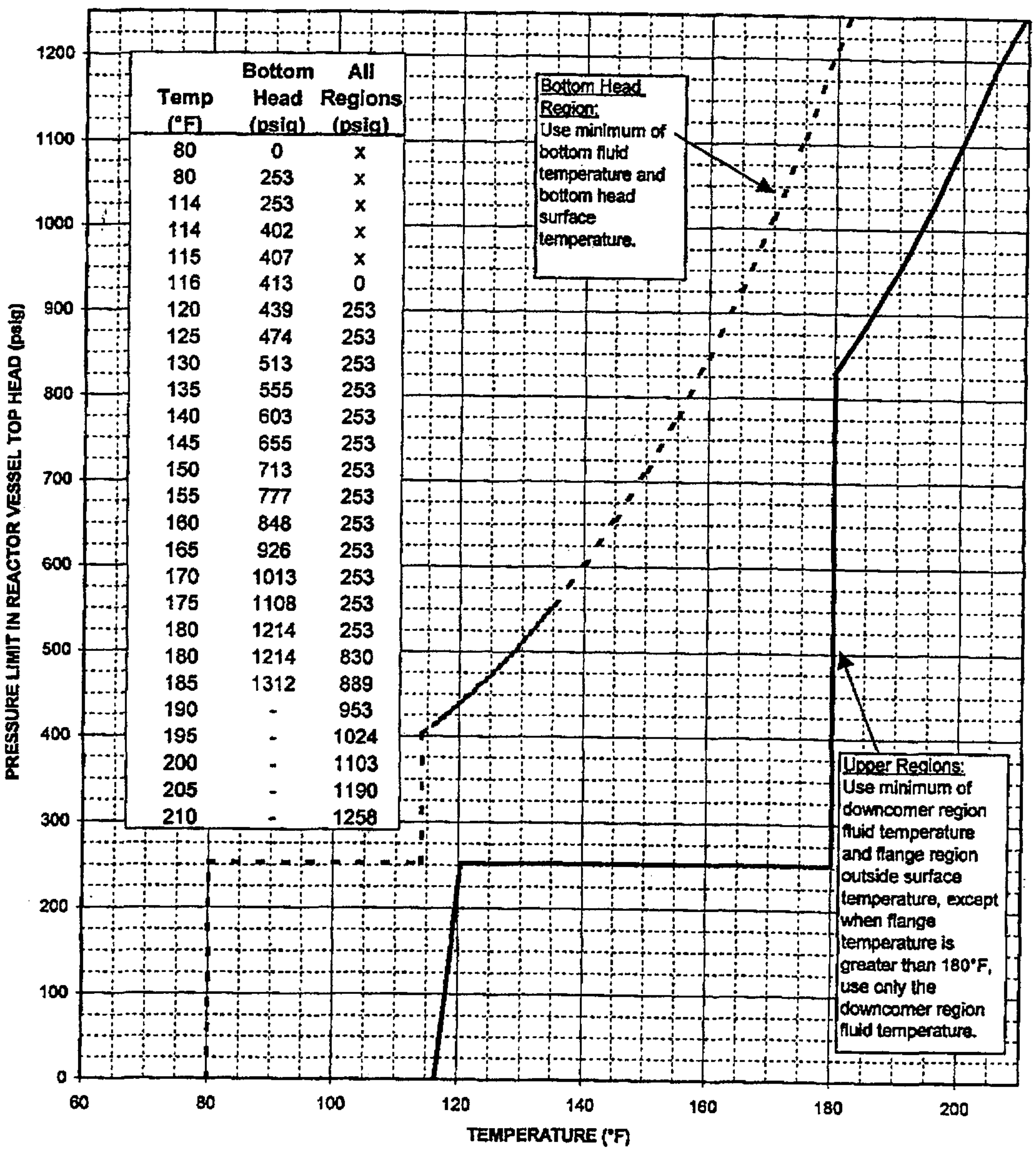
Figure 4: Core Not Critical P-T Curve (Curve B)



Temp (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	439	253
85	474	253
90	513	253
95	556	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)

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



Temp (°F)	Bottom Head (psig)	All Regions (psig)
80	0	x
80	253	x
114	253	x
114	402	x
115	407	x
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

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

Calculation of Effective Peak Beltline Fluence Value	Units	Find Reg Guide 1.99 equivalent fluence 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 $\sqrt{t}$ = 2.249 $\sqrt{\text{inch}}$
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 =	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	43.9	44.78	28.69	700	60.0	650
55.0	48.[illegible]	45.99	29.51	720	65.0	670
		47.34	30.40	742	70.0	692
		48.83	31.39	766	75.0	716
[illegible].0	[illegible]	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.[illegible]	[illegible]	61.67	39.96	975	105.0	925
		64.67	41.96	1024	110.0	974
		67.98	44.16	1078	115.0	1,028
[illegible].0	103.9	71.64	46.60	1138	120.0	1,088
115.0	108.9	75.68	49.30	1203	125.0	1,153
120.0	113.9	80.15	52.27	1276	130.0	1,226
125.0	118.9	85.08	55.57	1356	135.0	1,306

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

Pressure-Temperature Curve Calculation

(Pressure Test w/ Cooldown = Curve A)

Inputs:

Plant =	Yankee	
Component =	Beltline	
Vessel thickness, t =	5.0600	inches, so $\sqrt{t}$ = 2.249 $\sqrt{inch}$
Vessel Radius, R =	103.1875	inches
ART_{NDT} =	89.0	°F
Cooldown Rate, CR =	40	°F/hr
K_{IT} =	2.20	ksi*inch$^{1/2}$ (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 =	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	42.70	27.01	636	60.0	586
55.0	55.0	43.70	27.67	651	65.0	601
60.0	60.0	44.81	28.41	669	70.0	619
65.0	65.0	46.03	29.22	688	75.0	638
70.0	70.0	47.38	30.12	709	80.0	659
75.0	75.0	48.87	31.12	733	85.0	683
80.0	80.0	50.52	32.22	758	90.0	708
85.0	85.0	52.34	33.43	787	95.0	737
90.0	90.0	54.35	34.77	819	100.0	769
95.0	95.0	56.58	36.25	853	105.0	803
100.0	100.0	59.04	37.89	892	110.0	842
105.0	105.0	61.75	39.71	935	115.0	885
110.0	110.0	64.76	41.71	982	120.0	932
115.0	115.0	68.08	43.92	1034	125.0	984
120.0	120.0	71.74	46.37	1092	130.0	1,042
125.0	125.0	75.80	49.07	1155	135.0	1,105
130.0	130.0	80.28	52.05	1225	140.0	1,175
135.0	135.0	85.23	55.35	1303	145.0	1,253

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

Pressure-Temperature Curve Calculation

(Core Not Critical/ Heatup = Curve B)

Inputs:

Plant =	Yankee	
Component =	Beltline	
Vessel thickness, t =	5.0600	inches, so $\sqrt{t}$ = 2.249 $\sqrt{inch}$
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 = $(K_{IT}/M_T) \cdot 0.92$ 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 $\sqrt{t}$ = 2.249 $\sqrt{inch}$
Vessel Radius, R =	103.1875	inches
ART_{NDT} =	89.0	°F
Cooldown Rate, CR =	100	°F/hr
K_{IT} =	5.49	ksi*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 EEPYs	
$K_{IT} + 2 \times K_{IPL}$	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 ======>	All EFPYs	
K_{IT} for 552F - 50F Step =	106.56	ksi*inch$^{1/2}$	Temp. Change	502 °F Step
Safety Factor =	2.00	(for level A/B)		
K_{IP} for 1025 psig =	33.80	ksi*inch$^{1/2}$		
Temperature Adjustment =	10.0	°F		
Pressure Adjustment =	45.0	psig (hydrostatic pressure + uncertainty)		

Fluid Temperature T (°F)	1/8t Temperature (°F)	K_{IC} (ksi*inch$^{1/2}$)	K_{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 =	56.0	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)	[redacted]
Upper Flange 1 CD	40 F/HR CD plus Bolt Preload	3/4T	note 1	[redacted]
Upper Flange 1 HU	40 F/HR HU plus Bolt Preload	3/4T	note 2	[redacted]
Upper Flange 2 CD	40 F/HR CD plus Bolt Preload	3/4T	note 1	[redacted]
Upper Flange 2 HU	40 F/HR HU plus Bolt Preload	3/4T	note 2	[redacted]
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 Preloac	3/4T	note 1	*67.91*
Upper Flange 1 HU	100 F/HR HU plus Bolt Preloa	3/4T	note 2	*67.88*
Upper Flange 2 CD	100 F/HR CD plus Bolt Preloac	3/4T	note 1	*69.51*
Upper Flange 2 HU	100 F/HR HU plus Bolt Preloa	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 =	53.80	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 Critical - 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 EFPY	
$K_{IT} + 2 \times K_{IPL}$	5.18	ksi*inch$^{1/2}$ (Note: Factor of 2 is Safety Factor)		
Safety Factor =	2.00	(for level A/B)		K_I ksi*inch$^{1/2}$
K_{1P} for 1000 psig =	31.21	ksi*inch$^{1/2}$	K_{IPL}=0.0*Preload =	0
Temperature A djustment =	10.0	°F	K_{IT}=Thermal =	5.18
;justment =	35.0	psig (hydrostatic pressure + uncertainty)		

...tur... T (°F)	..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)
[illegible]	[illegible]	45.78	20.30	650	-5	615
		47.10	20.96	672	0	637
	.0	8.56	21.69	695	5	660
	0.0	0.18	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

Docket No. 50-271
BVY 03-29

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.''

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.

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 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.

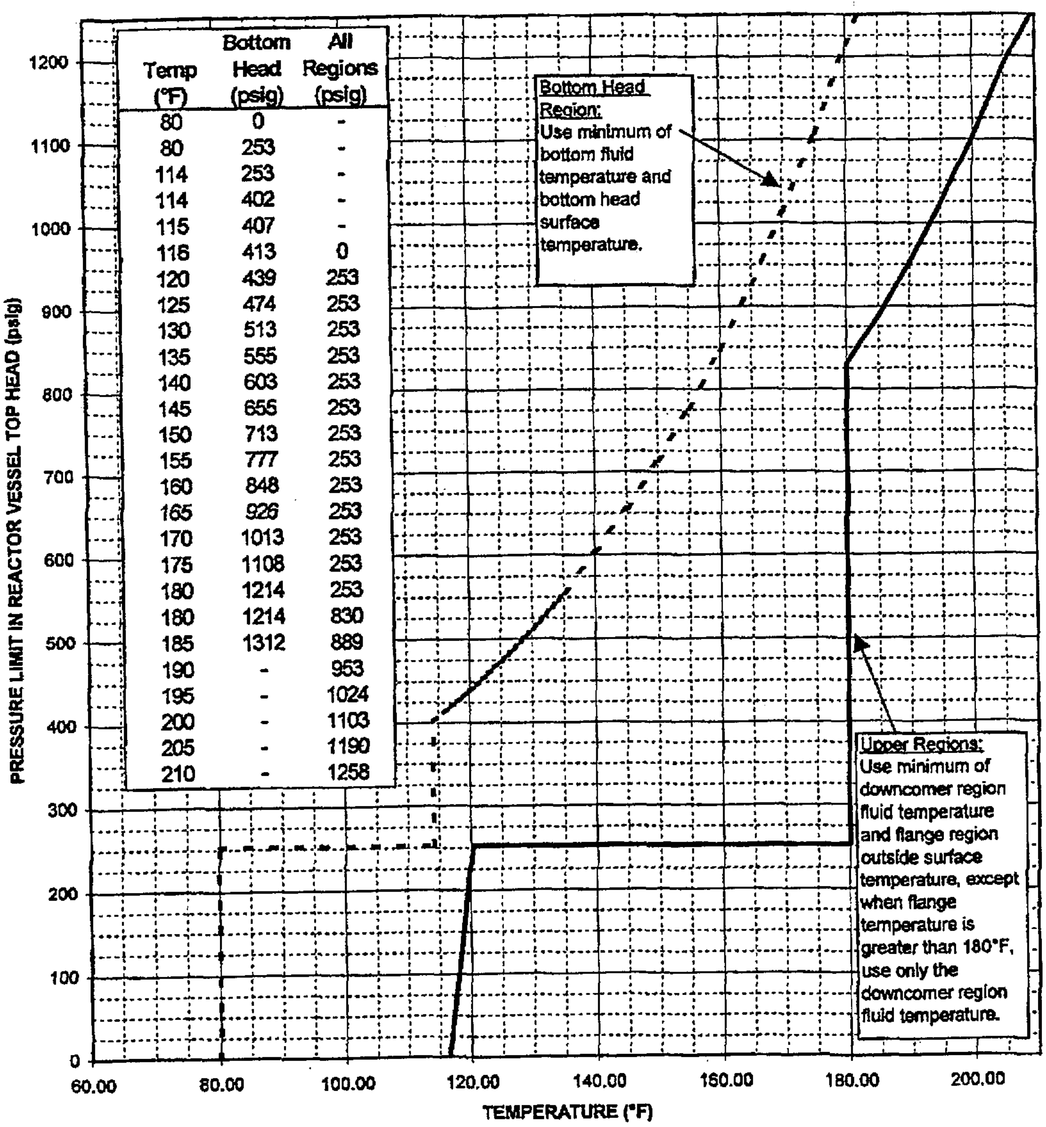
There were five regions of the reactor pressure vessel (RPV) that were evaluated in the development of the P/T Limit curves: (1) the reactor vessel beltline region, (2) the bottom head region, (3) the feedwater nozzle, (4) the recirculation inlet nozzle, and (5) the upper vessel flange region. These 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.

FIGURE 3.6.3

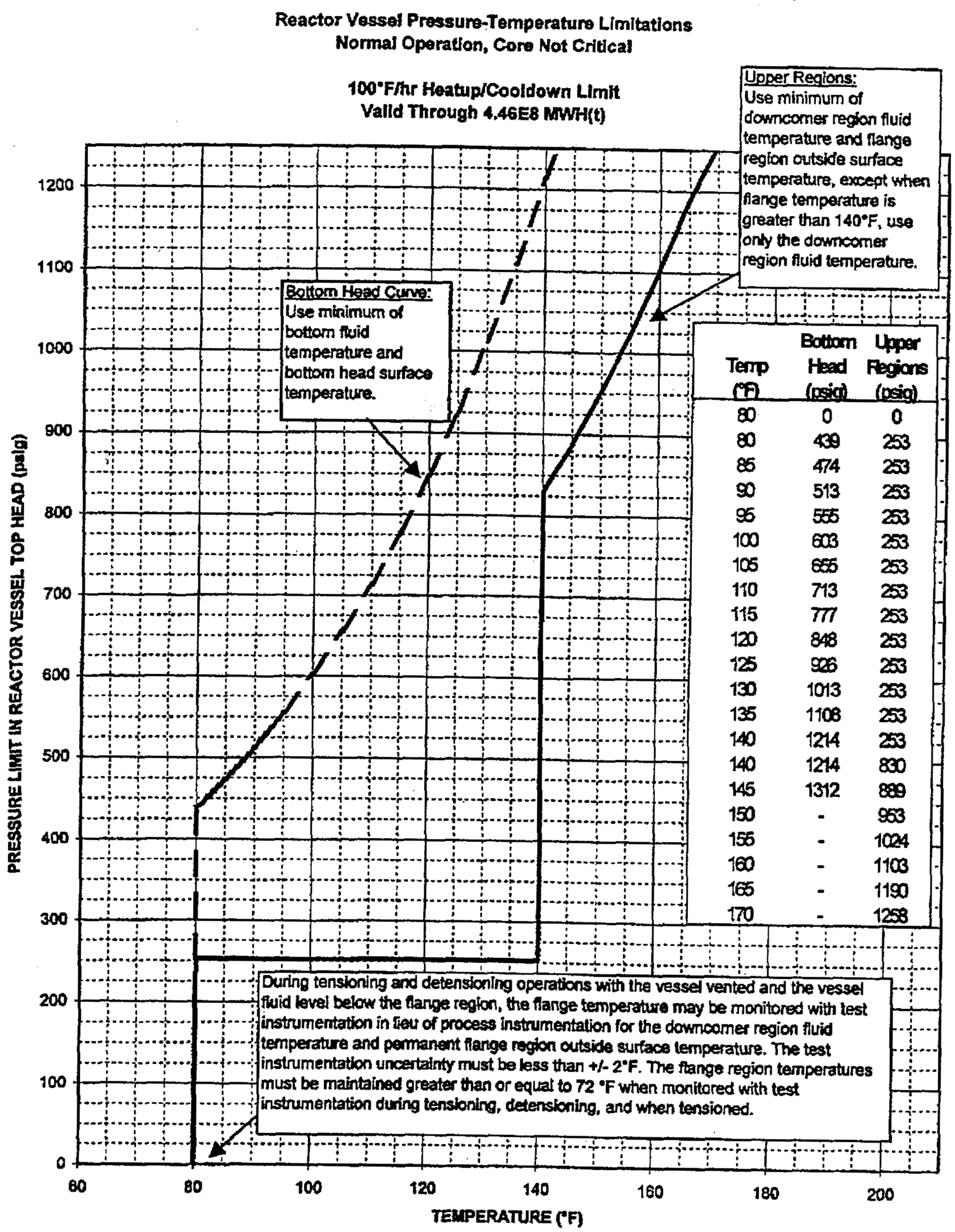
Reactor Vessel Pressure-Temperature Limitations
Normal Operation, Core Critical

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

FIGURE 3.6.2

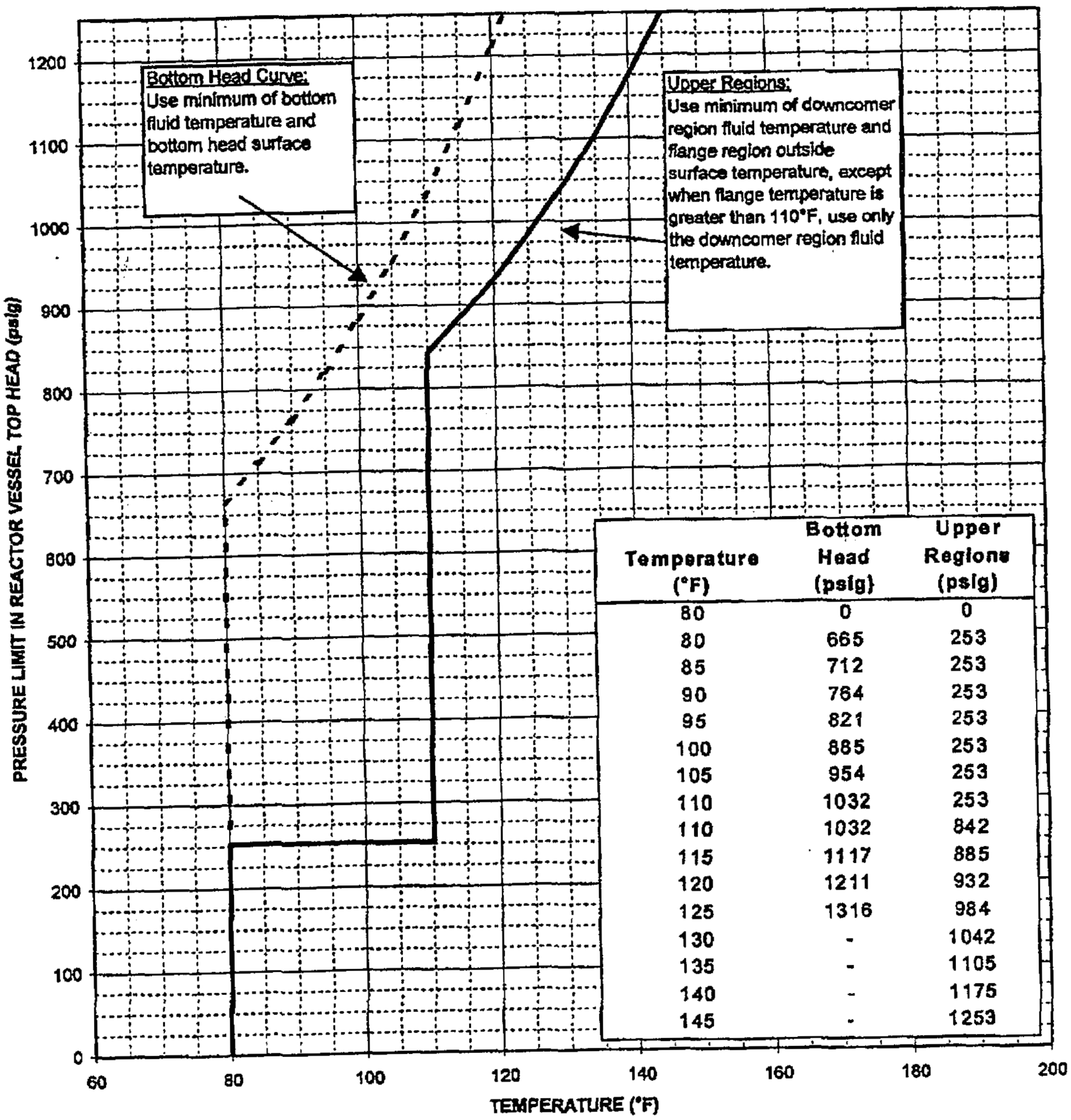


Temp (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	0
80	439	253
85	474	253
90	513	253
95	555	253
100	603	253
105	655	253
110	713	253
115	777	253
120	848	253
125	926	253
130	1013	253
135	1108	253
140	1214	253
140	1214	830
145	1312	889
150	-	953
155	-	1024
160	-	1103
165	-	1190
170	-	1258

Figure 3.6.1

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

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.

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

Docket No. 50-271
BVY 03-29

Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 258

RPV Fracture Toughness and Material Surveillance Requirements

Retyped Technical Specification Pages

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.

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 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 ($2.3x10^{17}$ n/cm^2) for a gross power generation of $4.46x10^8$ 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 1

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.

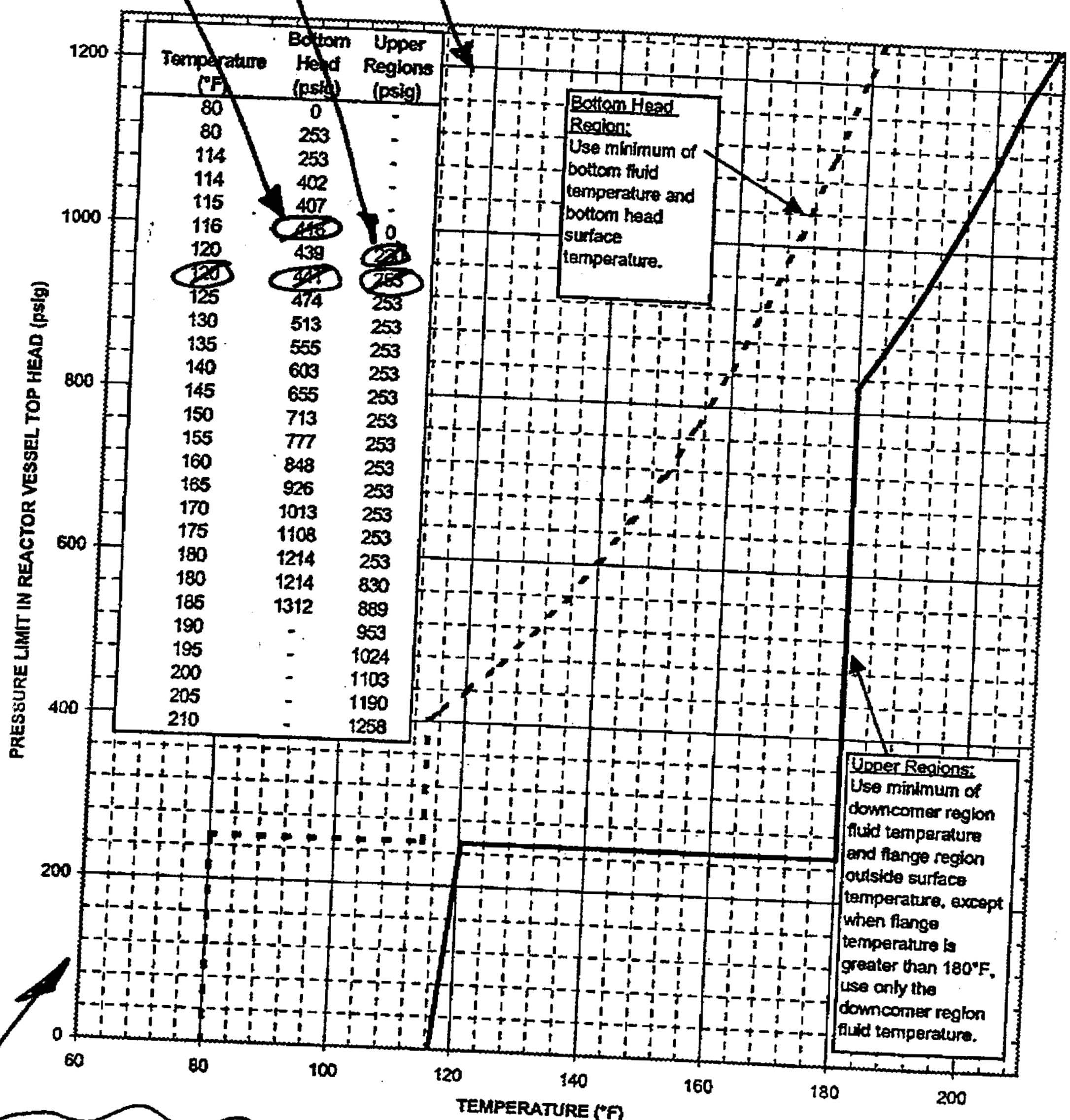
CHANGE GRID LINE DIVISIONS AND USE MORE DATA TO PLOT CURVES

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)



Temperature (°F)	Bottom Head (psig)	Upper Regions (psig)
80	0	-
80	253	-
114	253	-
114	402	-
115	407	-
116	~~446~~ 413	0
120	439	~~226~~ 253
~~120~~	~~441~~	~~263~~
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

CHANGE GRID LINE DIVISIONS AND USE MORE DATA TO PLOT CURVES

VYNPS

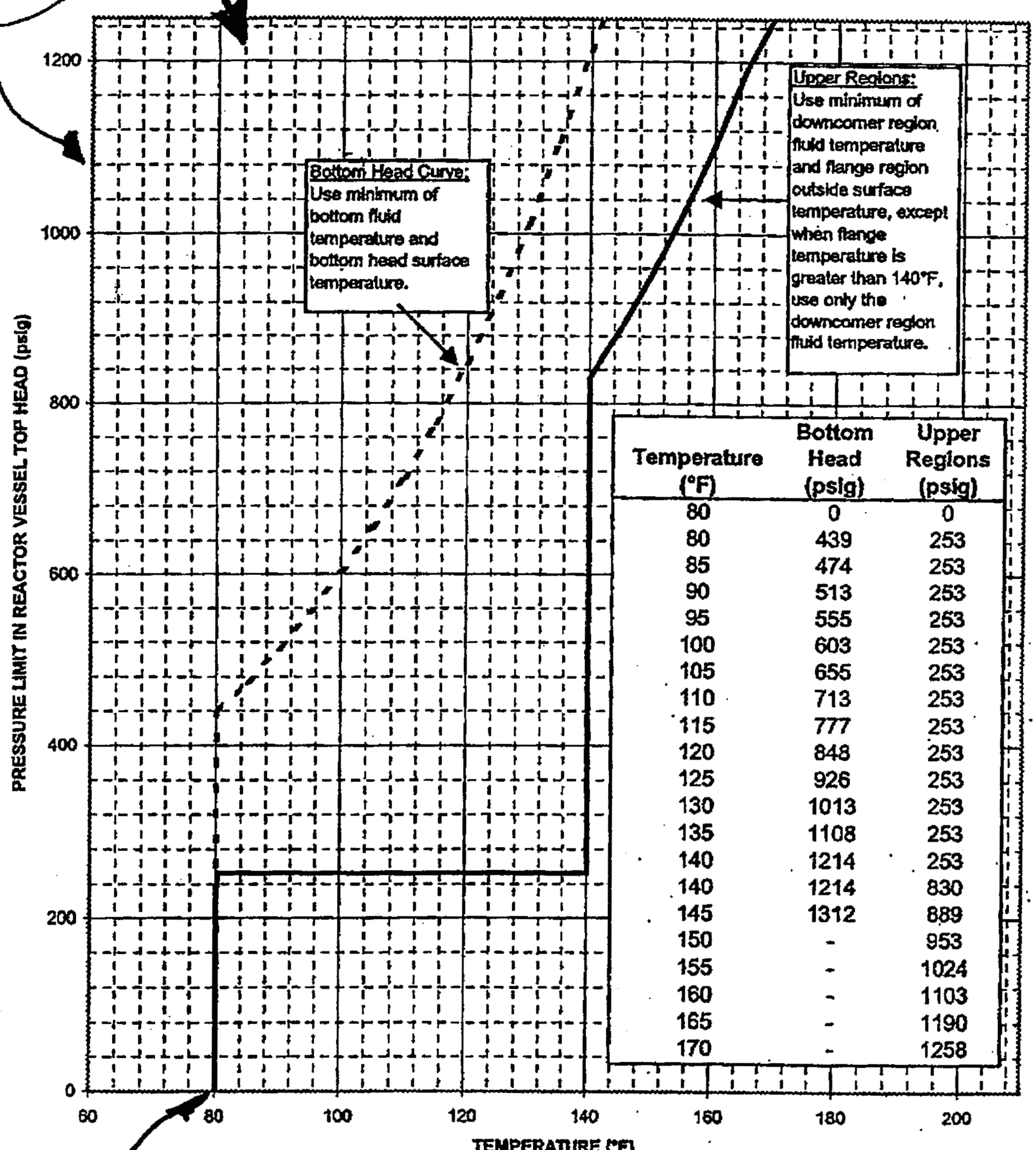
FIGURE 3.6.2

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 7

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.

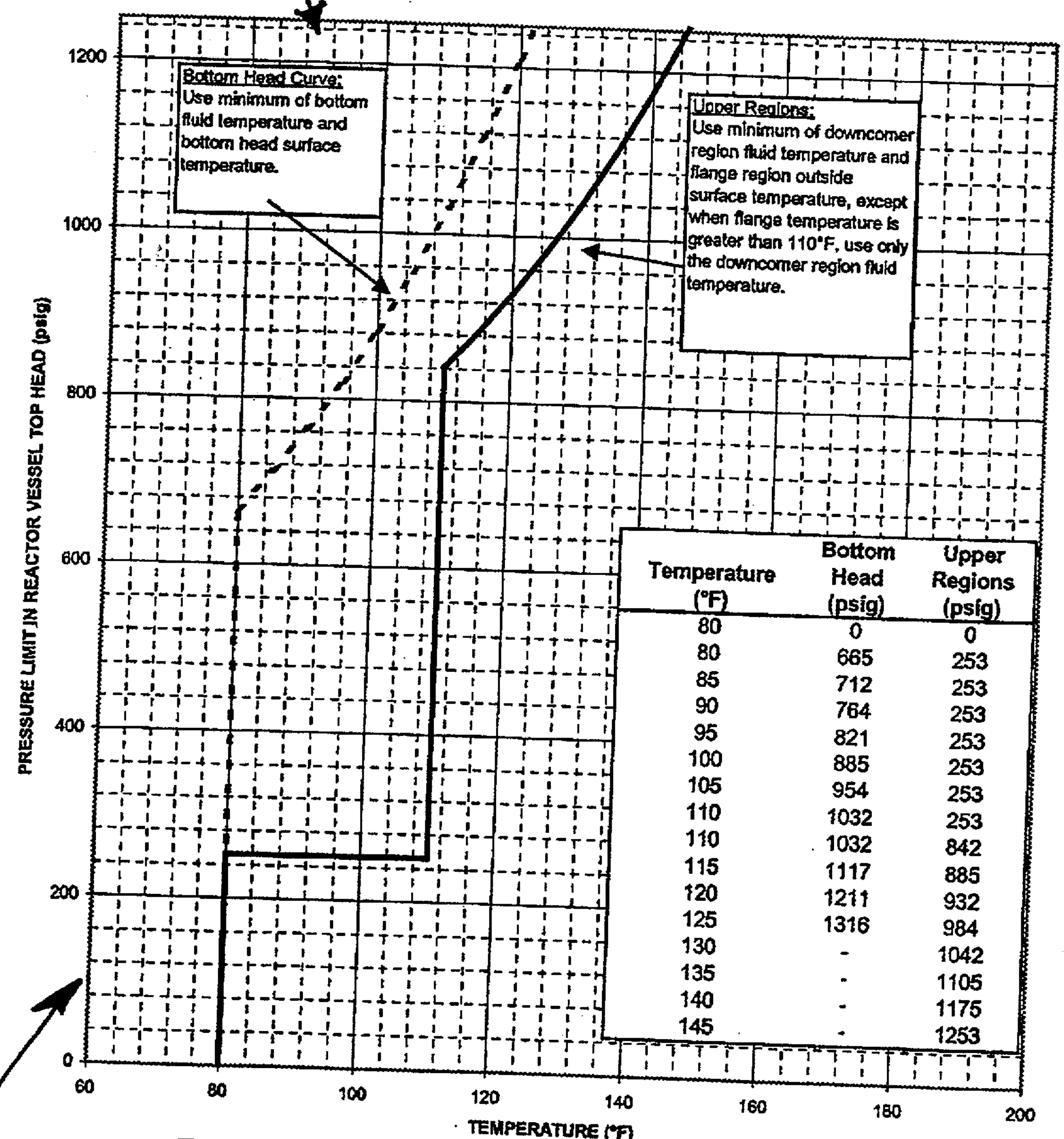
CHANGE GRID LINE DIVISIONS AND USE MORE DATA TO PLOT CURVES

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

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.

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.

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.

Docket No. 50-271
BVY 03-29

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

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~~

VERMONT YANKEE NUCLEAR POWER STATION

PROGRAM PROCEDURE

PP 7027

REVISION 3

REACTOR VESSEL INTERNALS MANAGEMENT PROGRAM

USE CLASSIFICATION: **INFORMATION**

RESPONSIBLE PROCEDURE OWNER: **Manager, System Engineering**

REQUIRED REVIEWS		**Yes/No**
E-Plan	10CFR50.54(q)	No
Security	10CFR50.54(p)	No
Probable Risk Analysis (PRA)		No
Reactivity Management		No

LPC No.	Effective Date	Affected Pages

Implementation Statement: N/A

Effective Date: 12/02/04

TABLE OF CONTENTS

1.0 PURPOSE, SCOPE, AND DISCUSSION

1.1. Purpose

The purpose of the Vermont Yankee Reactor Vessel Internals Management Program Procedure is to identify all Reactor vessel internals required to be inspected and outline their inspection requirements. This program also provides direction for evaluation of flaws and repair of Reactor internal components. In addition, it provides guidance for control of Reactor water chemistry and mitigation of Intergranular Stress Corrosion Cracking (IGSCC). This program describes how Vermont Yankee complies with 10CFR 50 Appendix B, ASME Section XI, and Boiling Water Reactor Vessel and Internals Program (BWRVIP) guidance with regard to reactor vessel internals inspection and program management.

This program procedure includes the following:

- Identification of the Reactor vessel internals components to be inspected
- Methods acceptable for inspection
- Required frequency of inspection
- Planned schedule for inspection
- Basis for inspection requirements
- Direction for flaw evaluation
- Direction for component repair
- Guidance for control of Reactor water chemistry
- Guidance on mitigation systems

In accordance with AP 6002, Preparing 50.59 Evaluations, the results of an Applicability Determination (AD) has determined that an AD is not required for future changes provided the procedure scope is not changed. The basis for this conclusion is that this document provides directions for implementing a maintenance or administrative process, subject to 10CFR50 Appendix B, that does not alter the design, performance requirements, operation, or control of systems, structures, or components (SSCs).

1.2. Scope

The Vermont Yankee Reactor Vessel Internals Management Program includes all of the Reactor vessel internals, with the exception of components that are considered consumable, such as the fuel bundles, control rods, and incore instruments. This program also includes the vessel shell cladding, but does not include any of the Reactor vessel pressure boundary. The Reactor vessel pressure boundary shell, heads, nozzles, flange and RPV flange bolting are governed by the Vermont Yankee Inservice Inspection (ISI) Program, PP 7015.

There is one exception to the above statement. The BWRVIP augments the ISI Program for one weld that is outside the Reactor vessel. This is weld N10-SE, the Standby Liquid Control safe-end-to-vessel nozzle connection. The requirements for this weld are discussed in Appendix A.

1.3. Discussion

This program addresses the requirements of ASME Section XI, Table IWB-2500-1, Categories B-N-1 and B-N-2. It also meets the requirements of various BWRVIP documents, as Vermont Yankee has committed to do so. It also addresses additional other commitments to the NRC and internal commitments, such as to address GE SILs. Finally, certain internals components and subcomponents have been determined to be significant as a risk to generation, and inspection recommendations have been assigned for these, as well.

Appendix A lists for each of the Reactor vessel internal components: method of inspection; frequency of inspection; and the planned schedule for inspection.

The inspection frequencies in Appendix A are based on an 18-month cycle. If cycle length is changed, Appendix A must be revised accordingly. In addition, when Vermont Yankee incorporates hydrogen water chemistry and if the NRC accepts BWRVIP-62, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, inspection frequencies for various internals components may be reduced.

<u>Background</u>

Vermont Yankee is a General Electric designed boiling water Reactor (BWR) power plant built in accordance with the ANSI B31.1 Construction Code. Vermont Yankee is sometimes described as a BWR 3/4 plant, however the more accurate designation is a BWR 4 with BWR 3 jet pumps and steam dryer. The Reactor vessel and shroud support were fabricated onsite by Chicago Bridge and Iron. The shroud and lower core spray piping was fabricated by Rotterdam Drydock in the Netherlands. The internals were installed on-site by Installation and Services Engineering for General Electric.

Until 1994, inspections of the Reactor vessel internals have been driven by the few required ASME Section XI inspections, NRC mandates, and the recommendations of GE Services Information Letters. In recent years Reactor vessel internals have received much attention because of intergranular stress corrosion cracking (IGSCC) that has been discovered at a significant number of BWRs. The BWR Vessel and Internals Project (BWRVIP) was formed in 1994 at the direction of the BWR Owners' Group to address this issue.

The BWRVIP identifies safety related internals components and their likely failure modes, specifies inspection methods and frequencies, and provides the methodology for evaluating flaws. It also specifies acceptable methods for demonstrating nondestructive examination (NDE) techniques and for determining technique uncertainty. It specifies requirements for repair or replacement of Reactor internals. Finally, it also addresses various methods of chemical control to mitigate potential future cracking. Every utility identifies members to represent the various BWRVIP disciplines. Each utility has also identified an executive for membership in the Executive Committee that controls funding and overall direction of the BWRVIP. ENN-DC-135, BWRVIP Inspection Program, provides guidance and requirements for managing and implementing the BWRVIP program.

2.0 DEFINITIONS

2.1. None

3.0 PRIMARY RESPONSIBILITIES

3.1. Program Owner (WPO):

3.1.1. Is responsible for the maintenance and coordination of the Vermont Yankee Reactor Vessel Internals Management Program. Is responsible for meeting the expectations of the program Owner, as described in Appendix A of AP 0098.

3.1.2. Prepares and maintains the inspection aspects of this program.

3.1.3. Reviews NRC Generic Letters, Information Notices, or regulations; BWRVIP documents; and General Electric SILs or RICSILs as they are issued for applicability to the Reactor Vessel Internals Management Program and documents this review per paragraph 4.1.4.

3.1.4. Ensures that technical justification are prepared, reviewed, and approved if an exception is taken to BWRVIP guidelines.

3.1.5. Prepares the inspection plan for each refueling outage in accordance with this program.

3.1.6. Assists the VY Site Reactor Internals Coordinator with selection of examination vendor and personnel.

3.1.7. Determines any additional (expanded sample) inspections made necessary by discovery of unacceptable indications in accordance with ASME Section XI or BWRVIP Inspection and Evaluations Guidelines.

3.1.8. Provides input to the VY Site Reactor Internals Coordinator and Design Engineering of the details of inspection findings, inspection technique limitations, and inspection coverage.

3.1.9. Determines any successive (follow-up) inspections made necessary by discovery of unacceptable indications in accordance with BWRVIP Inspection and Evaluations Guidelines.

3.1.10. Determines, in conjunction with licensing, the necessity for communications with the NRC. If an exception is taken to BWRVIP guidelines, this determination will be made in accordance with BWRGVIP-94.

3.1.11. Maintains a history of all Reactor vessel internals inspections.

3.1.12. Provides a refueling outage inspection report to the BWRVIP.

3.1.13. Publishes the Reactor Vessel Internals Health Report within 90 days of completion of each refueling outage.

3.2. VY Site Reactor Internals Management Program Coordinator (RIMPC):

3.2.1. Is responsible for site implementation of the Vermont Yankee Reactor Vessel Internals Management Program.

3.2.2. Provides technical advice and input for all aspects of the program.

3.2.3. Arranges for contractor on-site services for the performance of Reactor vessel internals inspection. Staffing levels should be adequate to provide coverage at all times during the inspection; for example, during In Vessel Visual Inspection (IVVI), at least one camera operator and one Level II should be available on each bridge.

3.2.4. Ensures that specific site and vendor inspection procedures are prepared, reviewed, and approved in accordance with AP 0095, AP 0096, AP 0097, and AP 0098, and administered in accordance with AP 6024.

3.2.5. Discusses requirements for voiding particular fuel bundle or control cell locations with Reactor Engineering.

3.2.6. Arranges for NDE Level III or other technical oversight including shared services.

3.2.6.1. Responsible for the review and approval of vendor NDE procedures.

3.2.6.2. Assures that, for NDE techniques other than visual, a performance demonstration has been conducted, which meets all key elements of the vendor NDE procedure.

3.2.6.3. Ensures that all NDE personnel qualifications are current and that they meet ASME Section XI and NE 8048, Procedure Paragraph 1, as appropriate.

3.2.6.4. Oversees or conducts NDE personnel indoctrination to meet NE 8042 and NE 8048, Procedure Paragraph 1.3.

3.2.6.5. Provides assurance that inspection activities meet the requirements of the ASME Section XI Code, BWRVIP guidelines, and this program.

3.2.6.6. Provides assurance that NDE data is of high quality.

3.2.6.7. Initiates Indication Discrepancy Reports in accordance with DP 4027, as required.

3.2.6.8. Responsible for review of all NDE documentation, including the final report, to ensure proper documentation in accordance with BWRVIP-03, ASME Section XI, or NE 8048, as applicable.

3.2.6.9. Prepares a technical justification per 4.2.3 and notifies the Program Owner if the VY Site Internals Coordinator elects to take an exception to BWRVIP guidance. Review is required by the Program Owner.

3.2.7. Responsible for completion of the refueling outage inspection plan.

3.2.8. Arranges for engineering evaluation of flaws.

3.2.9. Ensures that any additional (expanded sample) inspections specified by the Program Owner are completed.

3.2.10. Ensures ANII has reviewed NDE procedures, NDE personnel qualifications, and NDE reports when ASME Section XI is applicable.

3.2.11. Verifies that contractor special process procedures to be used in repair or replacement have been reviewed and approved by Design Engineering prior to use.

> **NOTE**
>
> Repairs shall be performed in accordance with AP 0070, ASME Section XI if applicable – or, if not specified therein – in accordance with the construction Code. In addition, repairs shall be performed in accordance with applicable BWRVIP documents.

3.2.12. Arranges for contractor support for Reactor vessel internal repair or replacement activity.

3.2.13. Is responsible for proper installation of Reactor vessel internals repairs or replacements.

3.2.14. Verifies that repair or replacement procedures have been reviewed and approved in accordance with AP 6001 and AP 0070, as appropriate, prior to use.

3.2.15. Monitors maintenance, repair, and replacement activities to ensure that required in-service and baseline inspection specified by Program Owner are performed prior to placing systems or components into service.

3.2.16. Coordinates with site scheduling, radiation protection, and ALARA personnel as it pertains to the Reactor vessel internals inspections, repairs, or replacements.

3.2.17. Arranges for Authorized Nuclear Inservice Inspector (ANII) review of appropriate Reactor internals inspection data, flaw analysis reports and repair or replacement activities.

3.2.18. Keeps the Code Programs Supervisor informed of inspection, repair, or replacement task progress of the Reactor vessel internals.

3.2.19. Ensures that cognizant departments are informed of unacceptable conditions to facilitate completion of appropriate paperwork (Condition Reports, Inservice Discrepancy Reports, WRs, etc.).

NOTE

Unacceptable inspection results are reported to the RIMPC by the examination agency or cognizant department for resolution. The Authorized Nuclear Inservice Inspector (ANII) is informed of the resolution.

3.3. Design Engineering Manager:

3.3.1. Is responsible for evaluation of any flaws found in Reactor vessel internals components;

3.3.2. Is responsible for the design of any Reactor vessel internals component repair or replacement.

3.3.3. Assures that Nobel Metal Chemical Application (NMCA) is scheduled as necessary and is accomplished to meet system goals.

3.3.4. Prepares a technical justification per 4.2.3 and notifies the RIMPC if Design Engineering elects to take an exception to BWRVIP guidance.

3.4. The examination vendor:

3.4.1. Provides staff and NDE services as specified in the purchase order and/or contract.

3.4.2. Notifies the RIMPC and Program Owner if the examination agency intends to take an exception to BWRVIP guidance, and assists in preparing a technical evaluation per 4.2.3. Also notifies the RIMPC if a Code requirement cannot be met.

3.4.3. Notifies the NDE Level III in a timely manner of any rejectable indications.

3.4.4. Provides IVVI or NDE Reports, which meet the requirements of NE 8048 or BWRVIP-03, as applicable.

3.4.5. Provides NDE certifications and training records for NDE personnel.

3.5. The Chemistry Department Superintendent

3.5.1. Reviews BWRVIP guidance relative to water chemistry and IGSCC mitigation and incorporates that guidance into plant procedures. Ensures that other industry guidelines relating to IGSCC mitigation are reviewed in a timely manner and incorporated into plant procedures where applicable.

3.5.2. Prepares a technical justification per 4.2.3 and notifies the RIMPC if Chemistry elects to take an exception to BWRVIP guidance.

3.5.3. Assures that operation of the Mitigation Monitoring System (MMS) is conducted in a safe and efficient manner. Tracks MMS availability and works to maximize its availability to meet or exceed system goals. Ensures that the MMS is routinely monitored and that coupons from the panel are evaluated per GE recommendations. Assures that MMS coupon test results are fed back for system operation.

3.5.4. Ensures that adequate trending of Reactor vessel chemistry is done in order to identify adverse trends.

3.5.5. Ensures that procedures are in place to identify and mitigate transient conditions such as condenser leaks and resin intrusions.

3.5.6. Ensures that the Chemistry Staff understands their role in vessel internals management and that staff members are adequately trained to accomplish required chemistry mitigation activities.

3.6. Operations Manager:

3.6.1. Assures the Reactor is shutdown when degradation of Reactor internals could potentially challenge safe plant operation.

3.6.2. Assures that Chemistry is informed when the MMS system trips or is taken out of service.

3.7. Mitigation Systems Engineer:

3.7.1. The functions of the Mitigation System Engineer, for complying with Industry Guidance related to IGSCC mitigation activities, will be performed by the Chemistry Department.

3.8. ALARA Engineer:

3.8.1. Works with Chemistry Department and Mitigation Systems Engineer to provide solutions for minimizing dose impact of the mitigation systems with regard to maximizing system availability.

3.9. Foreign Material Exclusion (FME) Coordinator:

3.9.1. Manages the FME Program. The FME Program assures that personnel perform their responsibilities in accordance with AP 6024 and AP 6026 relative to Reactor internal cleanliness, and that foreign objects are removed or dispositioned prior to re-assembly of the Reactor vessel.

3.10. Licensing Program Manager:

3.10.1. Provides Interface with the NRC for notification when ASME requirements cannot be met or if notification is required for not following BWRVIP guidance.

3.11. Code Programs Supervisor (Responsible Procedure Owner): (UND 2002-074_02)

3.11.1. Provides overall management of the Reactor Vessel Internals Management Program.

3.11.2. Functions as the overall single point of contact for Reactor vessel internals interdepartmental issues.

3.11.3. Chairs the Reactor Internals Management Committee. This committee is comprised of personnel from Code Programs, Systems Engineering, Mechanical/Structural Design Engineering, Plant Chemistry, Reactor Engineering, and management. This group is structured to have a comprehensive background related to BWR Reactor internals issues. This committee, through Code Programs, provides recommendations to VY management related to key Reactor internals related issues.

4.0 PROCEDURE

4.1. Governing Codes, Regulatory Commitments, and Basis for Inspection Requirements

4.1.1. ASME Section XI and PP 7015 - Title 10 Code of Federal Regulations, Part 50, Section 50.55a, Codes and Standards (10CFR50.55a) references the American Society for Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components. PP 7015, Vermont Yankee Inservice Inspection Program for the Fourth Interval provides the requirements for compliance with most parts of ASME Section XI. However, contained within Section XI is Table IWB-2500-1 – and Categories B-N-1 and B-N-2. These two categories address Reactor vessel internals inspection. This program (rather than PP 7015) addresses ASME Section XI, Table IWB-2500-1, Categories B-N-1 and B-N-2. Category B-N-1, Item No. B13.10 (which is a general interior inspection of the vessel performed each period) is not addressed specifically within this program, but will be more than satisfied by adherence to this program.

A relief request shall be initiated in accordance with PP 7015, Vermont Yankee Inservice Inspection Program, if it is desired to change or eliminate a particular ASME Code requirement.

4.1.2. BWRVIP Documents - The BWRVIP has issued a series of documents (a portion of which are contained in references) which contain requirements and recommendations for dealing with potential flaws in Reactor internals. The BWRVIP Executive Committee in a letter to the USNRC (Letter, Carl Terry to Brian Sheron, dated May 30, 1997) committed the U.S. utilities to the requirements of these BWRVIP documents. Vermont Yankee reiterated these commitments in its own letter to the NRC, BVY 97-123, dated September 30, 1997. Restated, those commitments are:

- Continue to provide the financial and technical resources needed to complete the BWRVIP Program Plan
- Actively participate in completing the BWRVIP Program Plan
- Implement the BWRVIP products at Vermont Yankee Nuclear Power Station as appropriate considering plant schedule, configuration and needs
- Provide timely notification to the NRC staff if Vermont Yankee does not implement the applicable BWRVIP product
- Continue to work closely with the NRC staff for the successful and timely conclusion of the BWRVIP Program Plan

4.1.3. Other Regulatory Commitments - In addition to the above documents, Vermont Yankee may make or may have made internal commitments or may have made commitments to the NRC, relative to various other industry documents, which deal with Reactor vessel internals. These may be GE Services Information Letters (SILs), Rapid Communication Services Information Letters (RICSILs), NRC Generic Letters, Information Notices, NUREGs, or others. These commitments are assimilated in this program in cases where they will continue to be followed. This program identifies where ongoing internal commitments are being revised and, which will in effect, act as the closeout of these old commitments. Old commitments, which have been closed out by completion of the commitment (e.g. a one-time component inspection), are not addressed in this program.

4.1.4. Inspection for Risk-to-Generation Purposes - In general, the inspections that are performed in accordance with the above documents or commitments are performed for safety related reasons. Notwithstanding, there are many Reactor vessel internals components which do not require inspections in accordance with the above documents or commitments. However, Vermont Yankee may elect to perform inspections on a regular basis of these components because they have been identified as a risk to generation. This type of inspection is also included in this program. When this is the case, this program (in Appendix A) identifies the non-mandatory nature of these inspections using should statements. These inspections may be driven by industry documents, such as GE Services Information Letters (SILs), Rapid Communication Services Information Letters (RICSILs), NRC Generic Letters, Information Notices, NUREGs, or others. Each new industry document relative to the Reactor internals should be assessed. This assessment should be controlled through the PCRS LO-CA process. Each BWRVIP document, BWRVIP revision, or BWRVIP-to-NRC piece of correspondence should be assigned an individual tracking item.

4.2. Implementation of BWRVIP Documents – Vermont Yankee implements the requirements of BWRVIP documents as follows (BWRVIP-94, Section 1.3):

4.2.1. When a BWRVIP document is newly published or revised, Vermont Yankee shall assess the impact on this program and consider the guidance contained therein and determine if immediate compliance is warranted. In addition, Vermont Yankee shall evaluate in the same manner any BWRVIP correspondence approved by the BWRVIP Executive Committee to the NRC that supplements BWRVIP documents. (UND 2002-074_04, BWRVIP-94, Section 1.3). This assessment shall be controlled through the PCRS LO-CA process. Each BWRVIP document, BWRVIP revision, or BWRVIP-to-NRC piece of correspondence shall be assigned an individual tracking item. Typically BWRVIP documents will be implemented within 2 outages of Executive Committee (EOC) approval. The 2-outage implementation would be the start of any required frequency over a period of time. For example, if a BWRVIP document requires that a group of components be inspected in a 6 year period (100% in 6 years), this schedule must be started within 2 outages or the 2nd refueling outage from EOC approval of the BWRVIP document. Changes to the BWRVIP Water Chemistry guideline will be implemented within 6 months.

Regardless of this determination, Vermont Yankee shall revise this program accordingly prior to the ensuing refueling outage. However, Vermont Yankee may elect to take exceptions to this requirement under the following circumstances:

4.2.1.1. If it is within eight months of the next refueling outage and the guideline pertains to performance of additional in-vessel visual inspections

4.2.1.2. If it is within 24 months of the next refueling outage and the guideline pertains to performance of additional ultrasonic inspections, or

4.2.1.3. If the guideline would affect a potential repair, replacement, or plant modification, the lead-time for design changes and hardware may be considered.

4.2.2. If Vermont Yankee elects to not comply with a particular BWRVIP requirement, it shall notify the NRC within 45 days of the publication of a BWRVIP document that incorporates all NRC/BWRVIP agreements OR the issuance of a closeout NRC Safety Evaluation Report (SER) on that document. A closeout NRC SER is one in which the NRC does not take any exceptions to the subject BWRVIP document as published. Notification is not required for work completed prior to either of these times.

4.2.3. In addition, if Vermont Yankee elects to not comply with a particular BWRVIP requirement - at any time - it shall prepare a technical justification, which justifies the deviation using the guidance provided in BWRVIP-94, Appendix A. Use VYPPF 7027.01 form at the end of Appendix C to document this deviation.

4.3. The Level III prepares the Refueling Outage Inspection Plan using the inspection requirements and guidelines in Appendix A. This is done well in advance of the outage, so as to allow adequate preparation time for the plant and the examination vendor. The Refueling Outage Inspection Plan lists all welds and subcomponents that require inspection for the upcoming refueling outage and identifies the required type of inspection, e.g., EVT-1, VT-3, UT, etc.

4.4. NE 8067 contains implementation requirements for inspection of Reactor internals and provides details of Reactor internals components and their inspection.

4.5. If inspections are done by ultrasonic testing (UT) or eddy current testing (ET), they are performed in accordance with a vendor procedure qualified in accordance with BWRVIP-03, Standards 2.2 and 2.3. The vendor UT or ET procedure shall also meet the requirements of BWRVIP-03, Standard 2.6 or 2.7, and other sections, as applicable. Vendor UT or ET procedures shall be approved by Vermont Yankee.

4.6. If inspections are performed visually, they shall be performed in accordance with NE 8048.

4.7. NDE personnel indoctrination shall be conducted to meet NE 8042 and NE 8048.

4.8. Disassembly of the Reactor vessel internals will not be required to examine any component, beyond that which is normally performed for a refueling outage.

4.9. Flaws shall be reported in accordance with DP 4027. Flaws shall be evaluated in accordance with BWRVIP Inspection and Flaw Evaluation Guidelines for components that perform a safety function. Subsequent BWRVIP/NRC correspondence should also be considered when evaluating flaws (BWRVIP-80_02).

4.10. If unacceptable indications are discovered, additional (expanded sample) inspections shall be performed in accordance with ASME Section XI or BWRVIP Inspection and Evaluations Guidelines, as appropriate.

4.11. Repairs or replacements of vessel internals shall be performed in accordance with AP 0070 and ASME Section XI if applicable – or if not specified therein – in accordance with the construction Code. In addition, repairs or replacements shall be performed in accordance with the appropriate BWRVIP Repair Guideline or BWRVIP Replacement Guideline. BWRVIP-04-A or BWRVIP-95, as applicable, will be used as a guide for format and content of a repair submittal to the NRC. (BWRVIP-004-A_01, BWRVIP-095_02) Subsequent BWRVIP/NRC correspondence should also be considered in the design, installation, and inspection of repairs (BWRVIP-2003-250_02).

4.12. All NDE documentation, including the final report, shall be reviewed to ensure proper documentation in accordance with BWRVIP-03, ASME Section XI, or NE 8048, as applicable.

4.13. Program Procedure Revisions – This program shall be revised – or an LPC issued - as needed, which includes the following situations. Revisions or changes may be held until just prior to the next refueling outage.

- Upon adoption of a new ASME Section XI Code edition or addendum.
- Upon implementation of a new BWRVIP guideline or guideline revision, this program shall be revised as soon as practical.
- When flaws are found, this program shall be revised to address possible changes in frequency of inspection, follow-up inspections, and repair or replacement determinations.
- When new commitments are made – either internally, or to the NRC.

5.0 REFERENCES AND COMMITMENTS

5.1. Technical Specifications and Site Documents

5.1.1. T.S. Section 3.6E
5.1.2. T.S. Section 4.6.E.1
5.1.3. T.S. Section 6.6
5.1.4. VOQAM, Vermont Yankee Operational Quality Assurance Manual
5.1.5. UFSAR Section 4.2 and Appendix K

5.2. Administrative Limits

5.2.1. None

5.3. Code, Standards, and Regulations

5.3.1. Code of Federal Regulations, 10CFR50.55.a
5.3.2. ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition through 2000 Addenda
5.3.3. NUREG-1544, Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components
5.3.4. CP-189-1995, ASNT Standard for Qualification and Certification of Nondestructive Testing Personnel

5.4. Commitments

5.4.1. Letter Vermont Yankee to USNRC, dated October 6, 1993, Reactor Vessel Clad Inspection during the 1993 Refueling Outage
5.4.2. Letter Vermont Yankee to USNRC, BVY 94-07, dated February 11, 1994, Request for Relief from NUREG-0619 Inspection Requirements
5.4.3. Memorandum T. G. Stetson to R. E. McCullough, dated October 25, 1996, Response to Commitment SIL0465S1RE2
5.4.4. Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC), dated May 30, 1997, BWR Utility Commitments to the BWRVIP

5.4.5. Letter Brian Sheron (USNRC) to Carl Terry (BWRVIP Executive Chairman), dated July 29, 1997, BWR Utility Commitments to the BWRVIP
5.4.6. Letter Vermont Yankee to USNRC, dated September 30, 1997, Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals
5.4.7. Letter Carl Terry to Brian Sheron, dated October 30, 1997, BWR Utility Commitments to the BWRVIP
5.4.8. Letter USNRC to VYNPC, dated April 29, 1999, NVY 99-46, Jet Pump Riser Circumferential Weld Inspections at Vermont Yankee Nuclear Power Station (TAC No. MA5109) (includes two-cycle SER)
5.4.9. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages
5.4.10. Action Item SIL-0462R1_01, dated March 27, 2001, Evaluate SIL No. 462 Rev. 2 'Access Hole Cover Cracking' OE
5.4.11. Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC), dated April 16, 2002, Utility Implementation of BWRVIP Products
5.4.12. Memorandum C.B. Larsen to D.C. Girroir, dated October 21, 2002, Evaluation of Clad Crack Indications Under the Reactor Head and in the Vessel

5.5. Supplemental References

5.5.1. BWRVIP-03, dated December 2001, BWR Vessel and Internals Project Reactor Pressure Vessel and Internals Examination Guidelines, Revision 4, EPRI TR-105696-R5
5.5.2. Letter NRC to BWRVIP, dated July 15, 1999, Final Safety Evaluation of BWRVIP Reactor Pressure Vessel and Internals Examination Guidelines (BWRVIP-03) Revision 1
5.5.3. BWRVIP-04-A, dated April 2002, Guide for Format and Content of Core Shroud Repair Design Submittals, EPRI TR-1006600
5.5.4. BWRVIP-06-A, dated March 2002, Safety Assessment of BWR Reactor Internals, EPRI TR-105707
5.5.5. BWRVIP-16, dated March 1997, BWRVIP, Internal Core Spray Piping and Sparger Replacement Design Criteria, EPRI TR-106708
BWRVIP-18, dated July 1996, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines, EPRI TR-106740
5.5.7. BWRVIP-19, dated September 1996, Internal Core Spray Piping and Sparer Repair Design Criteria, EPRI TR-106893
5.5.8. BWRVIP-25, dated December 1996, BWR Core Plate Inspection and Flaw Evaluation Guidelines, EPRI TR-107284
BWRVIP-26, dated December 1996, BWR Top Guide Inspection and Flaw Evaluation Guidelines, EPRI TR-107285
5.5.10. BWRVIP-27-A, dated August 2003, BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines, EPRI TR-107286
5.5.11. BWRVIP-28-A, dated April 2002, Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking
5.5.12. BWRVIP-38, dated September 1997, BWR Shroud Support Inspection and Flaw Evaluation Guidelines, EPRI TR-108823
5.5.13. BWRVIP-41, dated October 1997, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines, EPRI TR-108728
5.5.14. BWRVIP-42, dated December 1997, LPCI Coupling Inspection and Flaw Evaluation Guidelines, EPRI TR-108726

5.5.15. BWRVIP-47, dated December 1997, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines, EPRI TR-108727
5.5.16. BWRVIP-48, dated February 1998, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines, EPRI TR-108724
5.5.17. BWRVIP-49-A, dated March 2002, Instrument Penetration Inspection and Flaw Evaluation Guidelines, EPRI TR-108695
5.5.18. BWRVIP-50, dated May 1998, BWRVIP, Top Guide/Core Plate Repair Design Criteria, EPRI TR-108722
5.5.19. BWRVIP-51, dated May 1998, BWRVIP, Jet Pump Repair Design Criteria, EPRI TR-108718
5.5.20. BWRVIP-52, dated June 1998, BWRVIP, Shroud Support and Vessel Bracket Repair Design Criteria, EPRI TR-108720
5.5.21. BWRVIP-53, dated July 1998, BWRVIP, Standby Liquid Control Line Repair Design Criteria, EPRI TR-108716
5.5.22. BWRVIP-55, dated September 1998, BWRVIP, Lower Plenum Repair Design Criteria, EPRI TR-108719
5.5.23. BWRVIP-57, dated December 1998, BWRVIP, Instrument Penetrations Repair Design Criteria, EPRI TR-108721
5.5.24. BWRVIP-58, dated December 1998, CRD Internal Access Weld Repair, EPRI TR-108703
5.5.25. BWRVIP-62, dated December 1998, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, EPRI TR-108705
5.5.26. BWRVIP-76, dated November 1999, BWRVIP Core Shroud Inspection and Flaw Evaluation Guidelines, EPRI TR-114232
5.5.27. BWRVIP-79, dated February 2000, EPRI BWR Water Chemistry Guidelines - 2000 Revision, EPRI TR-103515-R2
5.5.28. BWRVIP-94, dated August 2001, BWRVIP Program Implementation Guide
5.5.29. BWRVIP-95, dated October 2001, BWRVIP Guide for Format and Content of BWRVIP Repair Design Submittals
5.5.30. BWRVIP-104, dated September 2002, BWRVIP Evaluation and Recommendations to Address Shroud Support Cracking in BWRs, EPRI TR-1003555
5.5.31. EDCR 75-30, Revision 2, dated September 15, 1976, Feedwater Sparer Replacement
5.5.32. EDCR 80-52, dated October 30, 1980 with Change No. 1 dated November 11, 1980, Change No. 2 dated December 12, 1980, and Change No. 3 dated March 4, 1982, Design and Installation of Clamping Device for Core Spray Sparer Junction Box C
5.5.33. EDCR 95-406, Revision 2, dated July 30, 1996, Specification for Design, Fabrication, and Installation Services for Reactor Pressure Vessel Core Shroud Repair at Vermont Yankee Nuclear Power Station, VYS-046, Revision 2
5.5.34. ENN-NDE-2.10, Certification of NDE Personnel
5.5.35. ENN-NDE-2.11, Certification of Ultrasonic Examination Personnel
5.5.36. ENN-NDE-2.12, Certification of Visual Testing (VT) Personnel
5.5.37. GE-NE-523-B13-01805-66, Revision 0, dated September 1996, Core Spray Flaw Evaluation for Vermont Yankee
5.5.38. GE-NE-B13-01935-02, Revision 1, dated July 1998, Jet Pump Assembly Welds Flaw Evaluation Handbook for Vermont Yankee
5.5.39. Letter BWRVIP to USNRC, dated January 11, 1999, BWRVIP Response to NRC Safety Evaluation of BWRVIP-18

5.5.40. Letter USNRC to BWRVIP, dated December 2, 1999, Final Safety Evaluation of Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)
5.5.41. Memorandum C. B. Larsen to D. C. Girroir, dated May 13, 1999, Definition of Core Support Structures (ASME Section XI, Category B-N-2)
5.5.42. Memorandum Carl Larsen to Dennis Girroir, dated September 26, 2002, Bases for PP 7027 Requirements and Recommendations
5.5.43. Memorandum John Hoffman to D. C. Girroir, dated November 26, 1999, Jet Pump Assembly Inspection Discrepancy Report Evaluation
5.5.44. MPR-1730, Revision 0, dated April 1996, Vermont Yankee Nuclear Power Station Core Shroud Repair - Design Report
5.5.45. Technical Evaluation No. 2001-030, dated May 14, 2001, Evaluation of Jet Pump Riser Flaws
5.5.46. Technical Justification 2003-03, dated August 18, 2003, Justification to Perform Less Than 5% of CRD Guide Tube Weld Exams Within the First Six-Year Interval
5.5.47. Technical Justification 2003-04, dated August 18, 2003, Continued Operation Without a Feedwater Zinc Injection System
5.5.48. Technical Justification 2003-05, dated December 17, 2003, Feedwater Copper Concentrations above Recommended Limits
5.5.49. Technical Justification 2004-01, dated March 26, 2004, Justification for Alternative Inspection of Core Plate Rim Hold-Down Bolts
5.5.50. Technical Justification 2004-02, dated March 26, 2004, Justification for Deferral of Inspection of Inaccessible Welds
5.5.51. Technical Evaluation 2004-0018, dated April 2004, Justification to Inspect Portions of Shroud Horizontal Welds H1, H2, H3 on the OD in Lieu of the Top Guide Spacer Block Welds, the Shroud Flange Ring Segment Welds, and the Top Guide Ring Segment Welds
5.5.52. Technical Justification TE-2003-0021, dated April 9, 2003, Justification to Revert to EVT-1 Inspection of Jet Pump Circumferential Welds with UT Indications
5.5.53. Technical Justification TE-2003-0023, dated July 7, 2003, Technical Assessment for Delaying Hydrogen Injection Into the Reactor Core
5.5.54. VY Calculation, VYC-2218, dated November 25, 2002, Structural Evaluation of RPV Top Guide Aligner
5.5.55. VY Snapshot Self Assessment Report BWRVIP Program, dated July 21, 2004
5.5.56. VYDC 2003-12, dated April 2004, Steam Dryer Strengthening
5.5.57. ENN DC-135, BWRVIP Inspection Program
5.5.58. AP 0009, Condition Reports
5.5.59. AP 0028, Learning Organization Action Tracking
5.5.60. AP 0070, ASME Section XI Repair and Replacement Procedure
5.5.61. AP 0095, Plant Procedures
5.5.62. AP 0096, Procedure Development, Review, Issuance and Cancellation
5.5.63. AP 0097, Limited Procedure Changes
5.5.64. AP 0098, Procedure Writer's Guide
5.5.65. OP 1111, Control Rod Removal and Installation
5.5.66. OP 1417, Disassembly/Re-Assembly of Fuel Cell
5.5.67. OP 2617, Chemistry Action Response Guide
5.5.68. OP 2638, Operation of the Mitigation Monitoring System (MMS)
5.5.69. DP 4027, Disposition of Inservice Inspection Findings
5.5.70. OP 4612, Sampling and Treatment of the Reactor Water System
5.5.71. AP 6001, Installation, Test and Special Test Procedures

5.5.72. AP 6024, Plant Housekeeping and Foreign Material Exclusion/Cleanliness Control
5.5.73. AP 6026, Refuel Floor Foreign Material Exclusion Control Procedure
5.5.74. AP 6045, Engineering Record Correspondence (ERC) and Technical Evaluations (TE)
5.5.75. AP 6807, Collection, Temporary Storage and Retrieval of Quality Assurance Records
5.5.76. PP 7015, Vermont Yankee Inservice Inspection Program
5.5.77. NE 8042, Training for Contract NDE Personnel
5.5.78. NE 8048, In-Vessel Visual Inspection
5.5.79. NE 8067, Reactor Vessel Internals Inspection Details

6.0 FINAL CONDITIONS

6.1. All required inspections and evaluations have been completed.

6.2. The NRC has been notified when required by BWRVIP-94 where Vermont Yankee has taken exception to BWRVIP guidance.

6.3. The BWRVIP has been notified where Vermont Yankee has taken exception to BWRVIP guidance. The BWRVIP has been notified when meaningful results are not obtained or when examinations cannot be performed because NDE techniques or equipment do not exist. The BWRVIP Program Manager has been notified of modifications to plant operation or configurations that may affect BWRVIP guidance (e.g., power uprate).

6.4. The vendor final report has been received and reviewed.

6.5. Applicable ASME Section XI inspections have been entered on the NIS-1 report.

6.6. All NDE inspection results, including IVVI, with supporting documentation and resolution of nonconformances (if applicable) shall be submitted for filing in accordance with AP 6807.

6.7. In-vessel inspection results pertinent to BWRVIP guidelines have been reported to the BWRVIP within 90 days of completion of the refueling outage.

6.8. Reactor Vessel Internals Health Report has been published within 90 days of completion of each refueling outage.

6.9. This program has been updated to include information and any additional requirements that have resulted from an inspection, including supplemental inspections.

6.10. This program has been updated to include any new BWRVIP commitments.

7.0 ATTACHMENTS

7.1.	Appendix A	Reactor Vessel Internals Components Inspection Scope and Schedule
7.2.	Appendix B	Reactor Vessel Internals Components Basis for Inspection and Other Management Requirements
7.3.	Appendix C	Technical Justifications

8.0 QA REQUIREMENTS CROSS REFERENCE

	Source Document	Section	Procedure Section
8.1	QAPM	Section B.11	Special Process Control, Subsections A, B.3, and C
8.2	ANSI N18.7	Section 5.2.18	Control of Special Processes

APPENDIX A

REACTOR VESSEL INTERNALS COMPONENTS INSPECTION SCOPE AND SCHEDULE

1.0 Control Rod Drive (Including Guide Tubes and Stub Tubes)

By RFO 23 (2002), at least five of the 89 CRD guide tube assemblies were due to have been inspected by the EVT-1 and VT-3 methods. One CRD guide tube 10-19 WAS inspected late in RFO 24 (2004). (Ref. 5.4.47) By RFO 27 (2008), a total of nine CRD guide tube assemblies shall have been inspected. (BWRVIP-47, Table 3.2-1) It is recommended that those inspections be grouped into outages where this minimum amount may be performed in conjunction with blade change-outs. These inspections are scheduled for RFO 22 (2001) and RFO 26 (2007) so that if the minimum number is not completed in that refueling outage they are completed in the refueling outage in which they are due. Inspection of the same location during different outages does not count towards satisfying the minimum sample requirement.

If access is gained to the lower plenum (areas below the core plate) for any reason, accessible surfaces of the CRD housings, CRD housing caps, and CRD stub tubes shall be visually inspected by the VT-3 method. (ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.40)

2.0 Core Plate

Core plate rim hold-down bolts shall be inspected by the UT method when tooling becomes available. (BWRVIP-25, Table 3-2) Until that time VT-3 shall be conducted of the topside of 50% of the rim hold-down bolts every other refueling outage.

If access is gained to the lower plenum (areas below the core plate) for any reason, accessible core plate beam fillet welds, rim hold-down bolts, and alignment hardware should be inspected by the VT-3 method. (Appendix B, 2.4)

3.0 Core Shroud (Including Tie Rod Repair and Spacer Ring)

Welds H1, H2, and H3 were inspected by EVT-1 in RFO 24 (2004) and shall be reinspected by EVT-1 in RFO 28 (2010) (TE 2004-0018). The reinspection by either EVT-1 or UT of the vertical welds and core plate ring segment welds required by RFO 25 (2005) were performed by EVT-1 during RFO 24 (2004) The vertical welds and core plate ring segment welds shall be reinspected by EVT-1 in RFO 28 (2010). (BWRVIP-76, Figure 3-3, TE 2004-0018)

All four of the tie-rods were reinspected in RFO 21 (1999). Two tie-rods were reinspected by the VT-3 method in RFO 24 (2004), the other two shall be reinspected in RFO 27 (2008), and so forth. If the tie-rods ever require retorquing, they shall be inspected for a baseline inspection following that activity, and then again following one cycle of operation. (BWRVIP-76, Section 3.5)

Accessible surfaces of the core shroud shall be visually inspected once per Ten-year ISI Interval by the VT-3 method. (ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.40) The Third Ten-year Interval inspection was performed in RFO 23 (2002).

4.0 <u>Core Shroud Support</u> (Including Access Hole Cover)

Welds H8 and H9 of the shroud support shall be reinspected by UT in RFO 25 (2005). A minimum coverage of 10% of weld H8 and 10% of weld H9 shall be achieved. (BWRVIP-38, Figures 3-4 and 3-5)

The two access hole cover welds should be inspected by the EVT-1 method every other refueling outage until a BWRVIP document is published that addresses this component or until this internal commitment is changed. (SIL046R1_01)

There are also radial welds in the shroud support baffle plate and vertical welds in the shroud support cylinder, which are not specifically required to be inspected except as part of the overall VT-3 inspection described below.

Accessible surfaces of the core shroud support shall be visually inspected once per Ten-year ISI Interval by the VT-3 method. (ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.40) This was performed for the Third Interval in RFO 23 (2002).

Because of a risk to generation the two shroud support flange vertical welds located between H7 and H8 will be visually examined by EVT-1 during RFO 25 (2005) and reinspection shall be both welds per 6 year cycle thereafter.

If access is gained to the lower plenum (areas below the core plate) for any reason, accessible surfaces of the shroud support legs and their welds, and the underside of the shroud support baffle plate and its welds shall be inspected by the VT-3 method. (ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.40)

The annulus floor should be inspected by the VT-3 method for debris and loose parts each refueling outage. (Appendix B,4.4)

5.0 <u>Core Spray Internal Piping and Spargers</u>

5.1 Thermal Sleeve Welds

There are three hidden welds inside each of the two core spray nozzles, which shall be inspected when an ultrasonic technique becomes available. (BWRVIP-18, Section 3.2.4) 100% of the welds shall be inspected. The reinspection frequency for this inspection is every eight cycles. (BWRVIP-18, Figure 3-3)

5.2 Internal Piping

The BWRVIP core spray piping reinspection frequency for ultrasonic inspection is two cycles and for EVT-1 it is one cycle. For either inspection method, all target welds shall be inspected. (BWRVIP-18, Figure 3-3) The target welds include the 24 creviced welds and the four tee-box-to-piping (P3) welds, and five or six of the 16 remaining welds (the elbow welds), for a total of 33 or 34 target welds. The 16 elbow welds shall be inspected on a rotating basis over three inspections. The fourth inspection slot will be filled by the thermal sleeve weld UT (BWRVIP-18, Figure 3-3) if a technique and tooling become available. Welds 1P9 and 3P9 shall be inspected when an ultrasonic technique becomes available. (UND2002-243_03)

5.3 Spargers

The five large circumferential welds (identified as S1, S2, and S4 in BWRVIP-18) in each sparger (20 welds total) shall be inspected every other cycle with the EVT-1 method. Fifty percent of the nozzle welds (identified as S3 in BWRVIP-18) shall be inspected every other cycle with the VT-1 method. The sparger welds received a baseline inspection in RFO 20 (1998), so sparger reinspections (the 20 large circumferential welds and two of four spargers' worth of nozzle welds on an alternating basis) would be performed in RFO 22 (2001), RFO 24 (2004), etc. (BWRVIP-18, Figure 3-4)

The sparger tee-box repair at 193 degrees shall be reinspected every other refueling outage with the VT-1 method. (BWRVIP-18, Section 3.3.3)

5.4 Piping and Sparger Brackets

The core spray piping brackets shall be inspected every four cycles with the EVT-1 method beginning in RFO 23 (2002). The core spray sparger brackets shall be inspected every other cycle with the VT-1 method beginning in RFO 23 (2002). (BWRVIP-18, Section 3.3.3, BWRVIP-48, Table 3-2, References 5.4.40 and 5.4.41, ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.30, ER-2001-2480_01)

6.0 <u>Feedwater Spargers</u>

The feedwater sparger tee welds and end bracket-to-vessel attachment welds shall be inspected by the VT-1 method every other refueling outage, i.e. RFO 22 (2001), RFO 24 (2004), RFO 26 (2007), etc. The other locations in the feedwater spargers shall be inspected by the VT-3 method during the same refueling outages. (Reference 5.3.2)

In addition, inspection by the EVT-1 method of the sparger end bracket-to-vessel attachment welds shall be performed once per Ten year ISI Interval. (BWRVIP-48, Table 3-2, and ASME Section XI, Table IWB-2500-1, Item B13.30) The Third Ten-year Interval inspection was performed in RFO 23 (2002).

7.0 Guide Rods

The entire guide rod assembly should be inspected by the VT-3 method in conjunction with the guide rod attachment welds (see Appendix K, Miscellaneous Vessel Internal Attachments) once per Ten-year ISI Interval. (Appendix B, 7.4) The Third Ten-year Interval inspection was performed in RFO 23 (2002). The next VT-3 inspection shall be performed in RFO 29 (2011).

8.0 Incore Flux Monitors (Including Housings, Guide Tubes, Dry Tubes)

Two dry tubes out of the ten total number should be inspected by the VT-1 and VT-3 methods every third refueling outage. (Appendix B, 8.3) Two dry tubes were inspected in RFO 21 (1999) and two more were inspected in RFO 24 (2004). Starting in RFO 25 (2005), 50% of the dry tubes that are 20 years old should be inspected every refueling outage. (SIL-409R2_02)

If access is gained to the lower plenum (areas below the core plate) for any reason, accessible incore housings, incore guide tubes, and incore guide tube stabilizers should be visually inspected by the VT-3 method. (Appendix B, 8.4)

9.0 Instrument Penetrations

No inspections of the instrument penetrations are required (beyond those contained in PP 7024 for nuclear boiler pressure testing).

10.0 Jet Pumps

10.1 Jet Pump Beams

UT inspection of 100% of the beams is required in RFO 23 (2002). Following that, reinspection of 100% of the beams using UT is required in the six-year interval between and including RFO 24 (2004) and RFO 27 (2008), and in each six-year interval thereafter. (BWRVIP-41, Table 3.3-1)

10.2 Jet Pump Riser Thermal Sleeve Welds

These welds inside the ten inlet nozzles shall be inspected when an ultrasonic technique becomes available. Between 50% and 100% of the welds shall be inspected at that first opportunity. If all of the welds are inspected, they do not require reinspection for twelve years after that. If only 50% are inspected, the other 50% shall be inspected in the next six-year interval. Reinspection of 25% of the welds would be required in the six-year interval following that. (BWRVIP-41, Table 3.3-1 and Section 3.2.4)

10.3 Jet Pump Riser Welds

The two thermal sleeve-to-elbow welds with flaws shall be reinspected by the EVT-1 method in RFO 24 (2004), RFO 26 (2007), and RFO 28 (2010). (Reference 5.4.53) If there is no flaw growth, these two welds could continue to be inspected every other cycle after that. It is also possible that if there were no flaw growth, the inspection frequency (which follows here) for the two flawed welds could be reassessed and extended.

Reinspection by the EVT-1 method of 50% of the riser welds was performed during RFO 24 (2004). Reinspection is 25% of the Jet Pump Riser Welds in the six year interval beginning RFO 28 (2010) and so on. (BWRVIP-41, Table 3.3-1)

10.4 Jet Pump Riser-to-Restrainer and Riser-to-Brace Welds

Inspection by the EVT-1 method of 50% of the riser-to-restrainer and riser-to-brace welds that were not inspected in RFO 20 (1998) were completed in RFO 24 (2004). Reinspection of 25% of the welds would be required in the six-year interval following RFO 27 (2008). (BWRVIP-41, Table 3.3-1)

10.5 Jet Pump Riser Braces

Inspection by the EVT-1 method of 50% of the riser brace welds that were not inspected in RFO 20 (1998) were inspected in RFO 24 (2004). Reinspection of 25% of the welds would be required in the six-year interval following RFO 27 (2008). (BWRVIP-41, Table 3.3-1, BWRVIP-48, Table 3-2, and ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.20)

10.6 Jet Pump Inlet Clamp Bolts

Inspection by the VT-3 method of the inlet clamp bolted connections in Loop B (50% of the total) was performed in RFO 24 (2004). Reinspection of 25% of the bolted connections would be required in the six-year interval following RFO 27(2008). (BWRVIP-41, Table 3.3-1)

10.7 Jet Pump Restrainer Assemblies

Since RFO 20 (1998) VY has visually examined 50% of the jet pump wedge assemblies every other outage. During RFO 22 (2001) and RFO 24 (2004) 100% of the jet pump wedge assemblies were examined by VT-1. The re-inspection cycle for jet pump wedge assemblies is 25% over each inspection cycle (6 years). Therefore, during RFO 25 (2005), RFO 26 (2007), RFO 27 (2008), and RFO 28 (2010) 5 jet pump wedge assemblies will have to be inspected and then 25% more over the next 6-year interval and so on. Current BWRVIP 41-A guidance does not require jet pump set screw inspections to identify vibration. This is because jet pump wedge assembly wear would be a tell tale sign that vibration was occurring. If wedge bearing surface wear is detected, inspection of the adjusting screws, hex nuts, etc. as applicable shall be examined during the same outage when the wedge wear was detected to determine the cause of wear.

10.8 Jet Pump Mixer Inlets

No inspections are currently required. (Reference 5.3.3)

10.9 Jet Pump Mixer/Diffuser Circumferential Welds above Diffuser Shell

25% of the mixer/diffuser welds above the diffuser shell shall be reinspected by either the UT or EVT-1 methods sometime in the six-year interval following October 2009. The welds included in this group are MX-1, MX-2, MX-4, and DF-1. (BWRVIP-41, Table 3.3-1)

10.10 Jet Pump Diffuser/Adapter Circumferential Welds below Diffuser Shell

The four diffuser welds with flaws shall be reinspected by the EVT-1 method in RFO 25 (2005), RFO 27 (2008), and RFO 29 (2011). (Reference 5.4.53) If no flaws are detected, these welds can revert to the normal inspection frequency (which follows here). Reinspection by either the UT or EVT-1 methods of the diffuser/adapter circumferential welds below the diffuser shell is required in the six-year interval between and including RFO 24 (2004) and RFO 27 (2008) and in each subsequent six-year interval. The welds included in this group are DF-2, DF-3, AD-3b, AD-1, and AD-2. (BWRVIP-41, Table 3.3-1)

10.11 Jet Pump Sensing Lines

Inspection by the VT-3 method of the sensing lines and their brackets should be performed in one loop every other outage. Inspection of the sensing lines in Loop B (jet pumps 1 through 10) was performed during RFO 22 (2001). In RFO 24 (2004), the sensing lines in Loop A (jet pumps 11 through 20) should be inspected, and so forth. (Appendix B, 10.1.11, 10.4)

11.0 <u>Lower Plenum</u>

If access is gained to the lower plenum (areas below the core plate) for any reason, accessible areas of the following components shall be inspected by the VT-3 method (ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.40):

- CRD housings
- CRD housing caps
- CRD stub tubes
- Core shroud support legs
- Core shroud support baffle plate underside

If access is gained to the lower plenum (areas below the core plate) for any reason, accessible areas of the following components should be inspected by the VT-3 method (Appendix B, 2.4, 8.4, 15.4):

- Core plate beam fillet welds
- Core plate rim hold-down bolts
- Core plate alignment hardware
- Incore flux monitor housings
- Incore flux monitor guide tubes
- Incore flux monitor guide tube stabilizers
- SLC and core plate ΔP lines

In addition, if access is gained to the lower plenum, the vessel bottom head and the bottom head drain should be inspected by the VT-3 method for debris or crud buildup. (Appendix B, 11.2)

12.0 Miscellaneous Vessel Internal Attachments (Including Steam Dryer, Specimen Holder, Guide Rod)

The steam dryer support attachment welds (EVT-1), the surveillance specimen holder bracket attachment welds (VT-1), the steam dryer hold-down bracket attachment welds (VT-3), and the guide rod bracket attachment welds (VT-3) shall be inspected once per Ten-year ISI Interval. The Third Ten-year Interval inspection was performed in RFO 23 (2002). (BWRVIP-48, Table 3-2, and ASME Section XI, Table IWB-2500-1, Category B-N-2, Item Nos. B13.20 and B13.30)

The dryer support bracket at 215 degrees was inspected in the Third Period of the Third Interval during RFO 22 (2001). This exam was performed in a similar fashion to the reinspections performed in RFO 17 (1993) and RFO 20 (1998). These exams are now complete. (Reference 5.3.1)

13.0 Orificed Fuel Support Castings

No inspection requirements. (Reference 5.4.42)

14.0 Specimen Holders

An inspection of the surveillance specimen holders should be performed each Ten-year ISI Interval. These inspections can be performed in conjunction with the bracket attachment welds (see Miscellaneous Vessel Internal Attachments above). (Appendix B, 14.4)

15.0 Standby Liquid Control/Core Plate Delta Pressure

The nozzle-to safe end weld and the safe-end extension shall be volumetrically inspected once every 10-year ISI Interval in accordance with the requirements of ASME Section XI, Appendix VIII, Supplement 10, when UT detection and sizing techniques are available and the weld surface condition is determined to be acceptable for U.T. Until such time they shall be examined by PT every other refueling outage. (TJ-2004-05, BWRVIP-27, Sections 3.3.1 and 3.4.1)

The SLC and core plate ΔP lines should be inspected if they are made accessible through other vessel activities. (Appendix B, 15.4)

16.0 Steam Dryer

Steam dryer modifications performed at VY during RFO 24 (2004) were done in support of scheduled operation at Extended Power Uprate conditions GE performed complete In Vessel Visual Inspection (IVVI) on the interior and exterior of the dryer plus eight (8) types of modifications and repairs. Attachment 4 of proposed technical specification change No. 263 (Reference 16.5.19), requires a detailed inspection of the steam dryer during the next and subsequent two refueling outages following power uprated operation. Additionally, inspections will be performed in accordance with the recommendations of SIL 644, Revision 1. (VYDC2003 12).

A VT-3 inspection of the steam dryer lifting lugs and associated hardware should be performed every fourth refueling outage. (Appendix B, 16.4), etc. This was performed in RFO 24 (2004) and should be reinspected in RFO 28 (2010).

17.0 Steam Separator/Shroud Head (Including Hold-down Bolts)

A VT-3 inspection of the steam separator/shroud head lifting lugs and associated hardware, standpipes, peripheral standpipe attachments, peripheral standpipe assembly welds, the tie bars, the tie bar attachment welds, the shroud head flange, and accessible areas of the shroud head should be performed every fourth refueling outage. (Appendix B, 17.4) This was performed in RFO 24 (2004), and should be reinspected in RFO 28 (2010), etc.

18.0 Top Guide

Two top guide hold-down assemblies 180 degrees apart shall be inspected every other refueling outage. The assemblies at 108 and 288 degrees were inspected in RFO 23 (2002), the assemblies at 18 and 198 shall be reinspected in RFO 25 (2005), and so forth. (BWRVIP-26, Table 3-2)

As part of the power uprate approval process VY committed to perform inspection of the top guide grid beams in accordance with the methods of SIL 544. The selection sample and frequency will be the same sample of cell locations chosen for CRD guide tube examination per BWRVIP-47, except the sample should be biased towards the higher fluence areas of the top guide. Over a twelve-year period 10% of the top guide grid beam cells are to be inspected, with RFO 25 (2005), RFO 26 (2007), and RFO 27 (2008). Five (5) of the 89 top guide grid beam cells are required to be inspected in the first 6-year interval.

An inspection of one quadrant of the top guide rim bolts and the perforated cover sheet bolts should be performed every fourth refueling outage on a rotating basis beginning in RFO 22 (2001). (Appendix B, 18.4)

There are no other top guide inspection requirements, pending a decision by the BWRVIP regarding the analysis of the removed Oyster Creek top guide grid samples.

19.0 Vessel Cladding

A sample of Reactor vessel cladding indications was reinspected in RFO 17 (1993), RFO 19 (1996), and RFO 23 (2002). These inspections are now complete. (Reference 5.3.1)

During refueling outage RFO 24 (2004) crud deposits were identified on the reactor vessel shell cladding at the elevation of the steam dryer support lugs. It resembled the "white stucco" that has been noted at other BWRs. The extent of the crud deposit was not determined, but it did not affect or mask any of the areas that were examined (e.g. core spray piping under head hold down brackets). This crud has not been observed prior to RFO 24. VY implemented NMCA in Spring 2001, with HWC implemented in November 2003.

APPENDIX A (Continued)

TABLE 1

December 3, 2003 — Reactor Internal Component / Outage Year / Outage	Inspection Basis	1995 RFO18	1996 RFO19	1998 RFO20	1999 RFO21	2001 RFO22	2002 RFO23	2004 RFO24	2005 RFO25	2007 RFO26	2008 RFO27	2010 RFO 28	2011 RFO29
Control Rod Drive Guide Tube Body Welds	BWRVIP-47, Table 3.2-1					EVT1 (4)		EVT1 (1)		EVT1 (4)	EVT1 (IN)		
Control Rod Drive Guide Tube Lug and Pin	BWRVIP-47, Table 3.2-1	*VT3*	*VT3*	VT3	VT3	VT3		VT3	VT3	VT3	VT-3	VT3	VT3
Core Plate Rim Hold-Down Bolts	BWRVIP-25, Table 3-2		VT3		VT3 (50%)	VT3 (50%)	VT3 (50%)	VT3 (50%)	UT				
Core Shroud Horizontal Welds (H1, H2, H3)	BWRVIP-76, Figure 2-3	UT						EVT1				EVT1	
Core Shroud Horizontal Welds (H4-H7)	BWRVIP-76, Section 3.2	UT											
Core Shroud Vertical Welds	BWRVIP-76, Figure 3-3		UT/ET					EVT1				EVT1	
Core Shroud TG Ring Segment Welds	BWRVIP-76, Section 3.4		UT/ET										
Core Shroud CP Ring Segment Welds	BWRVIP-76, Section 3.4		UT/ET					EVT1				EVT1	
Core Shroud Flange Ring Segment Welds	BWRVIP-76, Section 3.4												
Core Shroud Tie-Rod Repair	BWRVIP-76, Section 3.5		VT3 (all)	VT3 (all)	VT3 (all)			VT3 (2)			VT3 (2)		
Core Shroud Support Welds (H8, H9)	BWRVIP-38, Figures 3-4, 3-5		UT/ET						UT				
Core Shroud Support Access Hole Cover	GE SIL 462, Revision 1	*VT1*	*VT1*	*MVT1*	*EVT1*		*EVT1*		*EVT1*		*EVT1*		*EVT1*
Core Shroud Cylinder Vert Welds SSC-V1 & SSC-V2	BWR-76 Section 3.2.3								*EVT1*				*EVT1*
Core Shroud Support Annulus Floor	Risk To Generation	*VT3*	*VT3*	*VT3*	*VT3*	*VT3*	*VT3*	*VT3*	*VT3*	*VT3*	*VT3*	*VT3*	*VT3*
Core Spray Thermal Sleeve Welds (Hidden)	BWRVIP-18, Section 3.2.4								UT				
Core Spray Piping Welds (except P9)	BWRVIP-18, Figure 3-3	*VT1*	UT	EVT1	EVT1	EVT1	EVT1	EVT1	EVT1	EVT1	EVT1	EVT1	EVT1
Core Spray P9 Welds	BWRVIP-18, Section 3.2.4								UT		UT		UT
Core Spray Sparger Large Circ Welds	BWRVIP-18, Figure 3-4	CSVT1	CSVT1	MVT1		EVT1		EVT1		EVT1		EVT1	
Core Spray Sparger Nozzle Welds	BWRVIP-18, Figure 3-4	CSVT1	CSVT1	VT3		VT1 (50%)		VT1 (50%)		VT1 (50%)		VT1 (50%)	
Core Spray Piping Brackets	BWRVIP-18, Section 3.3.3	*VT1*	*VT1*				EVT1				EVT1		
Core Spray Sparger Brackets	BWRVIP-18, Section 3.3.3	CSVT1	CSVT1	VT3			VT1		VT1		VT1		VT1
Core Spray Sparger Tee-Box Repair (Old)	BWRVIP-18, Section 3.2.4	*VT3*	*VT3*	*VT3*	*VT3*			VT1		VT1		VT1	
Feedwater Sparger Tee Welds	NUREG 0619	VT1		MVT1		VT1		VT1		VT1		VT1	
Feedwater Sparger End Bracket Attachment	BWRVIP-48, Table 3-2	VT1		MVT1		VT1	EVT1	VT1		VT1		VT1	
Feedwater Sparger Piping and Brackets	NUREG 0619	VT3		VT3		VT3		VT3		VT3		VT3	
Guide Rods	Risk To Generation						VT3						
Incore Dry Tubes	SIL 409, Revision 2	*MVT1 (3)*			*VT1,3 (2)*			*VT1,3 (2)*	*VT1,3 (5)*	*VT1,3 (4)*	*VT1,3 (5)*	*VT1,3 (4)*	*VT1,3 (5)*
Integrally Welded Core Support Structures	ASME XI, Cat. B-N-2						VT3						
Jet Pump Beams	BWRVIP-41, Table 3.3-1	*VT3 (50%)*	*VT3 (50%)*	UT	UT (50%)		UT, VT-1		UT				
Jet Pump Thermal Sleeve Welds (Hidden)	BWRVIP-41, Table 3.3-1								UT				UT (50% IN)
Jet Pump Riser Welds (RS-1, RS-2, RS-3)	BWRVIP-41, Table 3.3-1			UT		UT (flaws)		EVT1(flaws)		UTorEVT1 (50%)		EVT1 (flaws)	
Jet Pump Riser Welds (RS-4, RS-5, RS-8, RS-9)	BWRVIP-41, Table 3.3-1	*VT1 (50% - 8,9)*	*VT1 (50% - 8,9)*	MVT1 (50%)				EVT1 (50%)				EVT1 (25%)	
Jet Pump Riser Brace Welds	BWRVIP-41, Table 3.3-1	*VT1 (50%)*	*VT1 (50%)*	MVT1 (50%)				EVT1 (50%)				EVT1 (25%)	
Jet Pump Inlet Bolted Connection	BWRVIP-41, Table 3.3-1			VT3 (50%)				VT3 (50%)				VT3 (25%)	
Jet Pump Restrainer Wedges	BWRVIP-41-A, Table 3.3-1	*VT3 (50%)*	*VT3 (50%)*	VT (50%)		*VT1 (50%)*		VT1 (50%)		*VT1 (25% thru RFO 28)*			VT1(25%) thru RFO 32
Jet Pump Restrainer Setscrews	BWRVIP-41-A	*VT3 (50%)*	*VT3 (50%)*	*VT (50%)*		*VT3 (50%)*							
Jet Pump Mixer Weld MX-1	BWRVIP-41, Table 3.3-1				EVT1 (100%)								
Jet Pump Mixer/Diffuser Welds (above shell)	BWRVIP-41, Table 3.3-1				UT (100%)								
Jet Pump Diffuser/Adapter Welds (below shell)	Risk To Generation				UT (100%)		UT (4 flaws)		EVT1 (4 flaws)		UTorEVT1(50%		EVT1 (4 flaws)
Jet Pump Sensing Lines	Risk To Generation	*VT3 (50%)*		*VT (50%)*		*VT3 (50%)*		*VT3 (50%)*		*VT3 (50%)*		*VT3 (50%)*	
Lower Plenum (CRD, Core Shroud Support)	BWRVIP-47 NRC Correspondence							WHEN ACCESSIBLE					
Lower Plenum (Core Plate, Incore, SLC)	Risk To Generation							*WHEN ACCESSIBLE*					
Miscellaneous Vessel Internal Attachments	BWRVIP-48, Table 3-2						EVT1,VT1,3						EVT1,VT1,3
Orificed Fuel Support Castings	BWRVIP-47, Table 3.2-1	*VT3*	*VT3*	*VT3*									
SLC Nozzle-to-Safe End Weld	BWRVIP-27, Section 3.3.1			EVT2* !	EVT2*	EVT2*	PT	PT	UT				
Steam Dryer	SIL 644, Revision 1			*VT3*	*VT3 (flaws)*			*VT1 & VT3*	*VT1 & VT3*	*VT1 & VT3*	*VT1 & VT3*	*VT3*	
Steam Dryer Support Bracket (at 215°)	BWRVIP-48, Table 3-2	VT3		VT3,UT(flaw)		VT3,UT(flaw)	EVT1						
Steam Separator/Shroud Head	Risk To Generation			*VT3*				*VT3*				*VT3*	
Steam Separator Hold-down bolts	Risk To Generation		*VT3*										
Top Guide Aligner Assemblies	BWRVIP-26, Table 3-2 and Calc.		*VT1 (2)*		VT1 (2)								
Top Guide Hold-down Assemblies	BWRVIP-26, Table 3-2		*VT3 (4)*		VT1 (2)		VT1 (2)		VT1 (2)		VT1 (2)		VT1 (2)
Top Guide Bolts (Rim and Cover Plate)	Risk To Generation					*VT3*				*VT3*			
Top Guide Grid Beams	BWRVIP-26, Section 3.2.2 & SIL 554	*VT*	*VT*	MVT1	VT1				EVT1	EVT1	EVT1(IN)		
Vessel Cladding	NRC Commitment		UT (aut)				UT (man)						

Table Key

Standard Print	= Inspections mandated by ASME, BWRVIP, or NRC commitments
Italics	= Inspections recommended for Risk-to-Generation purposes
UT	= Ultrasonic Testing performed or planned
UT (aut or man)	= Either automated or manual Ultrasonic Testing
ET	= Eddy Current Testing performed or planned
PT	= Penetrant Testing performed or planned
VT	= Visual Testing performed or planned
EVT1	= EVT-1; Enhanced Visual Test to look for cracking; 1/2 mil wire resolution with cleaning assessment
EVT2*	= Enhanced Leakage Inspection (direct view of component during pressure test)
VT1	= VT-1; Visual Test to look for cracks, wear, corrosion, etc.; resolution required: 1/32 black line
VT3	= VT-3; Visual Test to determine general mechanical/structural condition; no resolution requirements
CSVT1 or MVT1	= CSVT-1 or MVT-1; Core Spray Visual Test or Modified VT-1, no longer a defined test method; 1 mil wire resolution
?	= Inspections not yet determined
(IN)	= If necessary (to complete minimum number of inspections not performed in previous outage)
(all, number, %, or flaw)	= Perform inspection on all components, limited number (or percentage) of components, or just flawed components

APPENDIX B

REACTOR VESSEL INTERNALS COMPONENTS BASIS FOR INSPECTION AND OTHER MANAGEMENT REQUIREMENTS

1.0 Control Rod Drive (Including Guide Tubes and Stub Tubes)

1.1. BWRVIP Document Applicability

BWRVIP-47, published in December 1997, governs inspection of the control rod drive assemblies internal to the vessel, including the guide tubes and stub tubes. With the exception of two circumferential welds in one guide tube (Reference 1.5.21), Vermont Yankee will complete the minimum required inspections within the periods established in BWRVIP-47 as of the date of its publication. The document establishes six-year inspection intervals for specific inspections described below. Vermont Yankee defines the first six-year interval to include RFO 20 (1998), RFO 21 (1999), RFO 22 (2001), and RFO 23 (2002). The second six-year interval will include RFO 24 (2004), RFO 25 (2005), RFO 26 (2007), and RFO27 (2008). The third six-year interval will begin with RFO 28 (2010) and RFO 29 (2011).

The inspection requirements are established in BWRVIP-47, Table 3.2-1. This table requires inspection of four items on a CRD guide tube assembly: the guide tube sleeve-to-alignment lug weld (CRGT-1), the guide tube body-to-sleeve weld (CRGT-2), the guide tube base-to-body weld (CRGT-3), and the guide tube and fuel support alignment pin-to-core plate weld and the pin itself (FS/GT-ARPIN-1). CRGT-1 and FS/GT-ARPIN-1 require a VT-3 inspection and CRGT-2 and CRGT-3 require an EVT-1 inspection. Over a twelve-year period 10% of the CRD guide tube assemblies are to have had these four inspections performed, with 5% performed within the first six years. Those twelve and six-year intervals begin at the date of publication of BWRVIP-47, December 1997.

The two VT-3 inspections are actually satisfied during the orificed fuel support reinstallation/realignment procedure. The criteria for satisfying these VT-3 requirements are stipulated in BWRVIP-47, Table 3.2-1. The 10% sample will be completed during the normal course of blade change-outs over a twelve-year period. (Blade change-out requires orificed fuel support reinstallation and realignment). There are 89 CRD guide tubes at Vermont Yankee. Typically, there are between three and ten blade change-outs each outage, so it is reasonable to expect that there will be at least nine blade change-outs during the next twelve years. These inspections began in RFO 22 (2001). During RFO 22 (2001) and RFO 23 (2002) only four blades were changed out. Therefore, the 5% sample was not quite satisfied (four of 89 is 4.5%) in the first six-year inspection interval, as defined above. A technical justification in accordance with BWRVIP-94 was produced (see Reference 1.5.21).

Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method (such as for CRGT-2 and CRGT-3) with the EVT-1 method. These EVT-1 inspections may be performed from the ID of the guide tubes in conjunction with the blade change-out procedure. A minimum of five CRD guide tubes must have these inspections performed within the first six-year interval, and a minimum of nine must be performed within the next twelve years.

The stub tubes do not require inspection per BWRVIP-47.

The BWRVIP stated in response to NRC SE Issue 3.2.2 (Reference 1.5.13) that when utilities have access to the lower plenum due to maintenance activities not related to the inspection recommendations of the BWRVIP, they will have the opportunity to perform a visual inspection of a portion of the lower plenum and that results of this inspection will be reported to the BWRVIP. This will be treated as a commitment for those items listed in 1.2 below in the event that Relief Request RI-01 is accepted.

1.2. ASME Section XI Applicability

The CRD housings and stub tubes are part of the core support structure and are integrally welded. Therefore, the CRD housings and stub tubes will be examined in accordance with ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.40, "Core Support Structure". Table IWB-2500-1 requires accessible surfaces to be visually inspected by the VT-3 method once per ten-year interval. VY has submitted a Relief Request (RI-01) for the fourth ten-year Section XI interval that would allow using the BWRVIP guidance rather than the Section XI Categories B-N-1 and B-N-2 requirements. VY will perform inspections accordingly, based on the outcome of the Relief Request. These surfaces are not accessible during the normal course of a refueling outage and would only be accessible if there were another reason to gain access below the core plate. The last time this area was accessible and, therefore, inspected was in 1983. Because this occurrence is so rare, any time that there is an opportunity for this ASME Section XI inspection, it must be used. The inspection would include the control rod drive housing, control rod drive housing-to-stub weld, and the stub tube-to-vessel weld.

1.3. Other Commitments – None.

1.4. Inspections for Risk to Generation Purposes – None.

1.5. References

1.5.1. GE RICSIL No. 042, dated June 7, 1989, "BWR Under-Vessel Leakage"

1.5.2. Letter J. W. Lukas (GE) to M. P. Benoit, September 29, 1993, "Guide Tube Integrity"

1.5.3. Memorandum M. P. Benoit to J. T. Herron, October 1, 1993, "Recommendation On Reuse of Guide Tube 22-15"

1.5.4. Memorandum F. J. Helin to J. R. Hoffman, July 8, 1994, "VY Guide Tube"

1.5.5. Memorandum F. J. Helin to AP0028 File 'UND94010', November 7, 1994, "Reuse of Guide Tube 22-15"

1.5.6. GE Nuclear Energy Report GE-NE-523-A190-1294 DRF 137-0010-7, December 1994, "Vermont Yankee Control Rod Guide Tube Impact Analysis"

1.5.7. BWRVIP-03, dated October 1995, "Reactor Pressure Vessel and Internals Examination Guidelines"

1.5.8. Memorandum E. J. Taintor to D. C. Girroir, dated October 20, 1995, "Inservice Inspection of Vessel Internal Items Located Below the Core Support Plate"

1.5.9. BWRVIP-47, dated December 1997, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"

1.5.10. BWRVIP-55, dated September 1998, "Lower Plenum Repair Design Criteria"

1.5.11. BWRVIP-58, dated December 1998, "CRD Internal Access Weld Repair"

1.5.12. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"
1.5.13. Letter BWRVIP to USNRC, dated June 2, 1999, "BWRVIP Response to NRC SE on BWRVIP-47"
1.5.14. Letter USNRC to BWRVIP, dated October 13, 1999, "Final Safety Evaluation of 'BWRVIP, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47),' EPRI Report TR-108727, (TAC No. MA1102)"
1.5.15. Letter NRC to BWRVIP, dated December 7, 2000, "Acceptance for Referencing of BWRVIP, BWR Lower Plenum Inspection and Evaluation Guidelines (BWRVIP-47) for Compliance with the License Renewal Rule (10 CFR Part 54)"
1.5.16. Memorandum T. G. Stetson to Outage 22 File, dated January 25, 2001, "2001 Refuel Outage Blade Changeout Recommendation"
1.5.17. Action Item / Regulatory Commitment BWRVIP-047_01, dated November 28, 2001
1.5.18. Action Item / Regulatory Commitment BWRVIP-047-A_01, dated August 5, 2002
1.5.19. Action Item / Regulatory Commitment UND-2002-282_01, dated December 12, 2002
1.5.20. Action Item / Regulatory Commitment SEN-238_01, dated June 3, 2002
1.5.21. Technical Justification 2003-03, dated August 18, 2003, "Justification to Perform Less Than 5% of CRD Guide Tube Welds Within the First Six-Year Interval"

2.0 Core Plate

2.1. BWRVIP Document Applicability

BWRVIP-25 governs inspection of the core plate. Vermont Yankee was not able to complete the minimum required inspections in BWRVIP-25 as of the date of publication of that document (December 1997), and Vermont Yankee has not performed an inspection that would comply with this document as of yet. Therefore, Vermont Yankee is not in compliance with BWRVIP-94 for this component (compliance within two cycles following the publication of BWRVIP-25). Also, per BWRVIP-94, Vermont Yankee is required to notify the BWRVIP that VY will not be performing inspections in accordance with BWRVIP-25, until such time as this is possible.

Vermont Yankee is not now planning to install core plate wedges. Table 3-2 of BWRVIP-25 requires an EVT-1 inspection below the core plate of the rim hold-down bolts for BWR/4 plants without installed wedges. As an alternative, a UT of these bolts may be performed from the top. Vermont Yankee did not or will not perform either of these examinations in RFO 21 (1999), RFO 22 (2001), RFO 23 (2002), or RFO 24 (2004) due to the difficulty of removing CRD guide tubes for the EVT-1 (this amount of vessel disassembly is not normally performed during a refueling outage), and because no tooling now exists to perform the UT.

In the absence of core plate wedges during RFO 21 (1999), ~~and~~ RFO 22 (2001), and RFO 23 (2002), the tops of 50% of the core plate rim hold-down bolts were inspected during each outage with the VT-3 method. The NRC was notified that this was an alternative examination to EVT-1 from beneath the core plate, as stipulated in Table 3-2 of BWRVIP-25 for BWR/4 plants without installed wedges (see References 2.5.17 and 2.5.19). In RFO 24 (2004), VY will again inspect 50% of the core plate rim hold-down bolts with the VT-3 method. Technical Justification TJ-2004-01 was prepared in accordance with PP 7027, Paragraph 4.2.3 and BWRVIP-94 to perform this alternative examination (50% every other refueling outage) until such time that tooling to perform UT of the rim hold-down bolts becomes available.

Internal commitments in References 2.5.7 and 2.5.9 below to address SIL No. 588 will no longer be applicable with periodic inspection of the core plate rim hold-down bolts (or if core plate wedges were installed). These commitments are considered revised accordingly, with the issuance of PP 7027.

If new core plate wedges are ever installed, they may require some periodic inspection. BWRVIP-50, Paragraph 10.2, states, "Inspections required for the entire repaired top guide/core plate structures for the remaining life of the unit, shall be specified commensurate with design considerations and code requirements applicable to the specific design. This shall include inspections of the repair hardware and inspection of the reactor internal components utilized for repair anchorage." These inspection requirements would be delivered as a piece of the wedge design scope. Barring any guidance, the new wedges would all be reinspected after one cycle of operation. Thereafter, two wedges would be alternately inspected every third outage. This would ensure that all four core plate wedges are inspected every ten years.

Core plate plugs will reach their end of life (14 EFPY) in the cycle following RFO 25 according to Reference 2.5.21. This will require that the plugs be replaced or re-evaluated.

The BWRVIP stated in response to NRC SE Issue 3.2.2 (Reference 1.5.13) that when utilities have access to the lower plenum due to maintenance activities not related to the inspection recommendations of the BWRVIP, they will have the opportunity to perform a visual inspection of a portion of the lower plenum and that results of this inspection will be reported to the BWRVIP. This will be treated as a commitment for those items listed in 2.4 below.

2.2. ASME Section XI Applicability

The core plate is part of the core support structure; however, the core plate is not integrally welded as stated in the title of ASME Section XI, Table IWB-2500-1, Category B-N-2. Therefore the core plate is not subject to ASME Section XI (see Reference 2.5.15 below).

2.3. Other Commitments – None.

2.4. Inspections for Risk to Generation Purposes

The only surfaces accessible for visual inspection would be on the underside of the core plate and these surfaces are not accessible during the normal course of a refueling outage. They would only be accessible if there were another reason to gain access below the core plate. The last time this area was accessible and was inspected was in 1983. Because this occurrence is so rare, any time that there is an opportunity for inspection, it should be used. This nonmandatory inspection would include accessible core plate beam fillet welds, rim hold-down bolts and alignment hardware. The rim hold-down bolts and alignment hardware would not be considered a part of the safety-related core support structure when the core plate wedges are in place; however, they would be inspected from underneath the core plate for loose part considerations when accessible for other reasons.

2.5. References

2.5.1. Letter from Paul J. Kinder, GENE, to BWR Owners Group Core Plate Plug Evaluation Committee, dated August 11, 1992, "Transmittal of Final Evaluation Report"

2.5.2. GE RICSIL No. 071, Revision 0, dated November 22, 1994, "Top Guide and Core Plate Cracking"

2.5.3. Letter from BWRVIP to USNRC, dated January 3, 1995, "Request for Information Regarding the Impact of BWR Core Plate and Top Guide Ring Cracking"

2.5.4. GE SIL No. 588, dated February 17, 1995, "Top Guide and Core Plate Cracking"

2.5.5. NRC Information Notice 95-17, dated March 10, 1995, "Reactor Vessel Top Guide and Core Plate Cracking"

2.5.6. GE SIL No. 588, Revision 1, dated May 18, 1995, "Top Guide and Core Plate Cracking"

2.5.7. Memorandum T. G. Stetson to R. E. McCullough, dated February 5, 1996, "Response to Commitment SIL0588 on Top Guide and core Plate Cracking"

2.5.8. Memorandum T. G. Stetson to R. E. McCullough, dated February 5, 1996, "Response to Commitment INF 95017 on Top Guide and core Plate Cracking"

2.5.9. Memorandum T. G. Stetson to R. E. McCullough, dated July 11, 1996, "Response to Commitment SIL0588_01"

2.5.10. BWRVIP-25, dated December 1996, "BWR Core Plate Inspection and Flaw Evaluation Guidelines"

2.5.11. Letter Vermont Yankee to NRC, dated September 30, 1997, "Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals"

2.5.12. Letter NRC to Vermont Yankee, dated March 25, 1998, "Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals – Vermont Yankee Nuclear Power Station"

2.5.13. BWRVIP-50, dated May 1998, "Top Guide/Core Plate Repair Design Criteria"

2.5.14. Memorandum E. J. Taintor to D. C. Girroir, dated April 23, 1999, "Accessibility Following Installation of Proposed Top Guide and Core Support Assemblies"

2.5.15. Memorandum C. B. Larsen to D. C. Girroir, dated May 13, 1999, "Definition of Core Support Structures (ASME Section XI, Category B-N-2)"

2.5.16. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"

2.5.17. Letter Vermont Yankee to USNRC, dated October 29, 1999, BVY 99-137, "Deferral of Top Guide and Core Plate Wedge Installation"

2.5.18. Letter USNRC to BWRVIP, dated December 19, 1999, "Final Safety Evaluation of 'BWR Core Plate Inspection and Flaw Evaluation Guideline (BWRVIP-25),' EPRI Report TR-107284, December 1996 (TAC No. M97802)"

2.5.19. Letter Vermont Yankee to USNRC, dated September 26, 2000, BVY 00-89, "Cancellation of Top Guide and Core Plate Wedge Installation"

2.5.20. Letter NRC to BWRVIP, dated December 7, 2000, "Acceptance for Referencing of BWRVIP, BWR Core Plate Inspection and Evaluation Guidelines (BWRVIP-25) Report for Compliance with the License Renewal Rule (10 CFR Part 54) and Appendix B, BWR Core Plate Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule"

2.5.21. Memorandum T. G. Stetson to Core Plate Plug Lifetime File, dated May 15, 2003, "Surveillance 9021 on Core Plate Plug Removal"

2.5.22. Technical Justification 2004-01, dated 03/26/04, "Justification for Alternative Inspection of Core Plate Rim Hold-down Bolts"

3.0 Core Shroud (Including Tie Rod Repair and Spacer Ring)

3.1. BWRVIP Document Applicability

BWRVIP-01, published in September 1994, governed the baseline inspection of the horizontal welds in the core shroud. Vermont Yankee completed its baseline examination of the horizontal welds in RFO 18 (1995). As a result of this baseline examination, Vermont Yankee installed a tie-rod repair of the core shroud horizontal welds in RFO 19 (1996). BWRVIP-07, published in February 1996, governed reinspection of the core shroud welds and associated repairs. Vermont Yankee performed a baseline examination of the vertical welds in RFO 19 (1996). BWRVIP-63 governed inspection of the core shroud vertical welds. BWRVIP-76 was issued in November 1999, with the intent of subsuming BWRVIP-01, BWRVIP-07, and BWRVIP-63. Vermont Yankee has complied with these documents as of their publication.

3.1.1. Core Shroud Horizontal Welds – Per UFSAR, Appendix K, the tie-rod repair has structurally replaced core shroud horizontal welds H3 through H7. Therefore, in accordance with BWRVIP-07, Paragraph 4.4.1.1, and BWRVIP-76, Section 3.2, horizontal welds H3 through H7 do not require any further inspection. Welds H1 and H2 are considered design-reliant welds for the tie-rod repair. ER 2001-2481 (Reference 3.5.75) identified additional design-reliant welds for the shroud repair. The corrective action for this ER was to examine portions of H1, H2, and H3 in place of these other structures (which included the top guide support blocks). Accordingly, weld H3 will also be considered design-reliant. Technical Evaluation 2004-0018 (Reference 3.5.83) provides the basis for this decision. The reinspection frequency of "un-repaired" (design-reliant) horizontal welds is established in BWRVIP-76, Section 3.2, and Figure 3-1, which reference Figure 2-3 and Table 2-1. That frequency is ten years for welds that underwent UT and had minimal cracking (less than 10%), such as H1, H2, and H3. The NRC, in Reference 3.5.72, concurred with this determination. The next required examination would therefore be in RFO 24 (2004 (nine years later). Vermont Yankee has elected not to perform 100% of the accessible length of these welds in accordance with Technical Evaluation 2004-0018 (Reference 3.5.83) as would have been required. Appendix K of the FSAR will be revised accordingly. Vermont Yankee has also elected to perform these exams by EVT-1. Per BWRVIP-76, Figure 2-3, a full volumetric and/or two-sided surface technique is required. At VY, the inside of the shroud is not accessible at H1, H2, and H3 to perform an EVT-1. The core spray spargers cover H1 and H2 and because of the grating that covers the periphery of the top guide, access to the shroud ID would be through vacated fuel cells, and this would result in the camera being too distant from the inspection surfaces to perform an adequate EVT-1 of H1, H2, or H3. Technical Evaluation 2004-0018 (Reference 3.5.83) provides the basis for a one-sided EVT-1, as well. Following the RFO 24 (2004) inspection, horizontal welds will again require inspection in RFO 28 (2010), per Technical Evaluation 2004-0018 (Reference 3.5.83). Although no BWRVIP guidance is given for one-sided visual examinations of horizontal welds, this six-year inspection frequency follows the guidance for a one-sided EVT-1 of vertical welds per BWRVIP-76, Figure 3-3.

3.1.2. Core Shroud Vertical and Ring Segment Welds – The vertical and ring segment welds were inspected in RFO 19 (1996) in conjunction with the tie-rod installation. Option "A" of BWRVIP-07, Paragraphs 4.4.1.4 and 4.4.2.2, required that a minimum uncracked length be determined for both vertical and ring segment welds. For Vermont Yankee, MPR – the designer of the tie-rod repair – determined that this minimum length would be 41 inches (including allowance for crack growth rate). The RFO 19 (1996) inspection verified this minimum ligament for each vertical weld. Six ring segment welds received full coverage, except for one top guide ring segment weld (S3R3), which received 80% coverage due to a scanning limitation. No flaws were found. At the time the only document governing shroud vertical welds was BWRVIP-07, which followed the methodology used above for inspecting minimum ligaments for structural integrity. The Vermont Yankee RFO 19 (1996) vertical weld inspection met this criterion. Subsequently, BWRVIP-63 was issued, which required in Section 3.2, Option A, that 100% of the accessible length of all vertical welds (between H3 and H7 in Vermont Yankee's case) in repaired shrouds be inspected. BWRVIP-76, which was issued later, echoes this requirement. The reason that the vertical welds between H1 and H2 do not require inspection is that per BWRVIP-63, Section 3.1.1, or BWRVIP-76, Section 2.3.3.1, no inspection is required for vertical welds if the as found cracking in each horizontal weld at the ends of the vertical welds is less than 10% of the inspected length. This was as documented for the H1 and H2 welds in the 1995 inspection. Technical Evaluation 2004-0018 (Reference 3.5.83) provides the new basis for not inspecting the vertical welds between H1 and H2 going forward. Appendix K of the FSAR will be revised accordingly. The RFO 19 (1996) vertical weld inspection achieved 100% of all the accessible areas, with the exception of welds S5V1 and S5V2. Although more coverage could have been obtained on these welds, 56.5% and 68.3% was achieved, respectively. Vermont Yankee will comply with the BWRVIP coverage requirements in effect at the time of the next required reinspection. The reinspection frequency of vertical welds is found in BWRVIP-76, Figure 3-3. For vertical welds that were examined volumetrically and found to have no cracking, the inspection interval is ten years. Therefore, the next required inspection of the vertical welds would be in RFO 25 (2005 (nine years later). Technical Evaluation 2004-0018 provides the basis for only inspecting the OD of the vertical welds. During RFO 24, the vertical welds were inspected by EVT-1. Therefore, these welds shall be reinspected in RFO 28 (2010) (BWRVIP-76 Figure 3.3).

The RFO 19 (1996) examination included the six ring segment welds in the rings at the top guide and core plate. The three ring segment welds in the shroud flange were not examined based on the good results obtained on weld H1 in RFO 18 (1995). Weld H1 will be used to ensure that the shroud flange (top ring) segment welds have sufficient design reliant weld length; therefore, the top ring segment welds will not be inspected. BWRVIP-07 did not require inspection of the shroud flange ring segment welds. BWRVIP-76 states that the repair designer should establish the need to inspect ring segment welds; if the repair designer is able to demonstrate that the repair hardware does not rely on the integrity of particular ring segment welds in order for it to function properly, then no inspection is necessary. Technical Evaluation 2004-0018 (Reference 3.5.83) states that only the ring segment welds at the core plate will be required for future inspections. Welds H1, H2, and H3 will need to be inspected one cycle sooner than the ring segment and vertical welds, because welds H1 and H2 were examined in RFO 18 (1995); the other welds were examined in RFO 19 (1996). The core plate ring segment welds were inspected by EVT-1 during RFO 24 and shall be reinspected in RFO 28 (2010). (BWRVIP-76 Section 3.4)

3.1.3. Core Shroud Tie-Rod Repair - BWRVIP-07, Paragraph 4.2, contained requirements for inspection of repair components of core shrouds. It required a VT-3 of critical areas of 25% of the repair assemblies following the first operating cycle after repair installation and every ten years thereafter of all assemblies. The NRC requested that utilities perform this inspection of 100% of the assemblies following the first cycle of operation, in light of the Nine Mile Point 1 incident. The repair was installed in RFO 19 (1996) and Vermont Yankee satisfied the first-cycle inspection requirement in RFO 20 (1998). However, the tie-rods were retorqued to a higher value during that outage. Therefore, Vermont Yankee considered the repair a new installation and reinspected all four of the tie-rods again in RFO 21 (1999). If the tie-rods are to be retorqued again, a baseline inspection should be performed following that activity and the tie-rods should be examined again following one cycle of operation.

BWRVIP-76 has now replaced BWRVIP-07. BWRVIP-76, Paragraph 3.5, Option 1, which makes the best sense for Vermont Yankee, requires reinspection of repair component assemblies once every ten years after a first cycle inspection. Vermont Yankee decided the best way to comply was to perform inspection of two tie rods every three outages.

MPR, the designer of the tie-rod repair, has designated inspection requirements (all by the VT-3 method) for the tie-rods. The inservice inspection requirements were derived from the MPR installation (PSI) inspection requirements. (Reference 3.5.47)

3.2. ASME Section XI Applicability

The core shroud is part of the core support structure. Therefore the core shroud will be examined in accordance with ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.40, "Core Support Structure". Table IWB-2500-1 requires accessible surfaces to be visually inspected by the VT-3 method once per ten-year interval. This was conducted during RFO 23 (2002) for the third ten-year Section XI interval. VY has submitted a Relief Request (RI-01) for the fourth ten-year Section XI interval that would allow using the BWRVIP guidance rather than the Section XI Categories B-N-1 and B-N-2 requirements. VY will perform inspections accordingly, based on the outcome of the Relief Request.

3.3. Other Commitments – None.

3.4. Inspections for Risk to Generation Purposes – None.

3.5. References

3.5.1. GE RICSIL No. 054, Dated October 9, 1990, "Core Support Shroud Crack Indications"
3.5.2. Memorandum E. J. Betti to J. R. Hoffman, dated October 26, 1990, "RICSIL No. 054, Core Support Crack Indications"
3.5.3. Memorandum T. G. Stetson to C. B. Cameron, dated December 26, 1990, "Response to Commitment RICSIL0054, Core Support Shroud Crack Indications"
3.5.4. Memorandum T. G. Stetson to C. B. Cameron, dated February 15, 1991, "Delaying Commitment RICSIL0054RE1"
3.5.5. Memorandum T. G. Stetson to R. E. McCullough, dated July 17, 1991, "Response to Commitment RICSIL0054RE1"
3.5.6. Letter C. B. Cameron to A. D. Himle, dated July 22, 1991, regarding SIL 462, RICSIL054, RICSIL059 and GE support for possible inspection findings
3.5.7. OP 1428, Issued January 15, 1992, "Core Shroud Support Visual Inspection"
3.5.8. GE RICSIL No. 054 Revision 1, Dated July 21, 1993, "Core Support Shroud Crack Indications"
3.5.9. Memorandum T. G. Stetson to R. E. McCullough, dated August 9, 1993, "Response to Commitment RICSIL0054R1"
3.5.10. NRC Information Notice 93-79, dated September 30, 1993, "Core Shroud Cracking At Beltline Region Welds In Boiling Water Reactors"
3.5.11. Letter Michael E. Shepherd to D. A. Reid, dated October 4, 1993, "GE Nuclear Energy SIL No. 572"
3.5.12. GE SIL No. 572, dated October 1, 1993, "Core Shroud Cracks"
3.5.13. GE SIL No. 572, Revision 1, dated October 4, 1993, "Core Shroud Cracks"
3.5.14. Memorandum J. R. Hoffman to S. R. Miller, dated October 12, 1993, "Independent Assessment of R&CE Inspection of Vermont Yankee Core Shroud"
3.5.15. Memorandum W. F. Miller (GE) to M. P. Benoit, dated October 27, 1993, "Summary of Vermont Yankee Core Shroud Visual Examinations During RFO17"

3.5.16. Memorandum Jim Brooks to Rick McCullough, dated November 19, 1993, "Review of NRC Information Notice 93-79: Core Shroud Cracking At Beltline Region Welds In Boiling Water Reactors, and SEN 103, Circumferential Crack of a Boiling Water Reactor Core Shroud"

3.5.17. Memorandum T. G. Stetson to R. E. McCullough, dated February 14, 1994, "Response to Commitment SIL0572R1 On Core Shroud Cracks"

3.5.18. GE RICSIL No. 068, dated April 8, 1994, "Update On Core Shroud Cracking"

3.5.19. GE RICSIL No. 068, Revision 2, dated May 6, 1994, "Update On Core Shroud Cracking"

3.5.20. NRC Information Notice 94-42, dated June 7, 1994, "Cracking In the Lower Region of the Core Shroud In Boiling Water Reactors"

3.5.21. Memorandum T. G. Stetson to R. E. McCullough, dated July 11, 1994, "Response to Commitment RICSIL0054R1RE1"

3.5.22. Memorandum T. G. Stetson to R. E. McCullough, dated July 12, 1994, "Response to Commitment RICSIL068"

3.5.23. Letter BWROG to USNRC, dated July 13, 1994, "Response to NRC Request for Shroud Information"

3.5.24. NRC Information Notice 94-42, Supplement 1, dated July 19, 1994, "Cracking In the Lower Region of the Core Shroud In Boiling Water Reactors"

3.5.25. NRC Generic Letter 94-03, dated July 25, 1994, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors"

3.5.26. Letter BWROG to USNRC, dated August 5, 1994, "Revision 1 to BWROG Shroud Document"

3.5.27. Letter VY to USNRC, BVY 94-82, dated August 17, 1994, "Response to USNRC Generic Letter 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs"

3.5.28. BWRVIP-01, dated September 1994, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"

3.5.29. BWRVIP-02, Revision 1, dated September 1994, "BWR Core Shroud Repair Criteria"

3.5.30. Memorandum Bryan Croke to Mark Palionis, dated September 27, 1994, "NRC Information Notice 94-42, Cracking In the Lower Region of the Core Shroud In Boiling Water Reactors"

3.5.31. Memorandum Bryan Croke to Mark Palionis, dated September 27, 1994, "NRC Information Notice 94-42 Supplement 1, Cracking In the Lower Region of the Core Shroud In Boiling Water Reactors"

3.5.32. BWRVIP Core Shroud NDE Uncertainty & Procedure Standard, dated November 21, 1994

3.5.33. Letter VY to USNRC, BVY 94-125, dated December 15, 1994, "Vermont Yankee Plans to Inspect the Core Shroud Circumferential Welds During the Spring 1995 Refueling Outage"

3.5.34. Letter USNRC to VY, NVY 95-01, dated January 5, 1995, "Safety Evaluation for Vermont Yankee Nuclear Power Station Regarding Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds In Boiling Water Reactors"

3.5.35. Letter USNRC to VY, NVY 95-22, dated March 14, 1995, "Summary of March 9, 1995, Meeting with Representatives of Vermont Yankee Nuclear Power Corporation"

3.5.36. BWRVIP Report, dated April 1995, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines Revision 1"

3.5.37. Letter VY to USNRC, BVY 95-45, dated April 21, 1995, "Vermont Yankee 1995 Outage Core Shroud Inspection"

3.5.38. Letter USNRC to VY, NVY 95-52, dated April 25, 1995 "Generic Letter 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds In Boiling Water Reactors, Vermont Yankee Nuclear Power Station"

3.5.39. Memorandum D. C. Girroir to J. M. DeVincentis, dated April 26, 1995, "CAT A Item #GENLETR9403MEC3, re. Core Shroud Inspections"

3.5.40. B&W Nuclear Technologies Report dated April 27, 1995, "1995 Vermont Yankee Nuclear Power Corporation Project File Report for Core Shroud Examinations"

3.5.41. Letter USNRC to VY, NVY 95-55, dated April 27, 1995, "Core Shroud Inspection and Flaw Evaluation, Vermont Yankee Nuclear Power Station" (Safety Evaluation Attached)

3.5.42. Letter VY to USNRC, BVY 95-55, dated May 24, 1995, "Generic Letter 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds in BWRs"

3.5.43. BWRVIP-04, dated October 1995, "Guide for Format and Content of Core Shroud Repair Design Submittals"

3.5.44. BWRVIP-07, dated February 1996, "Guidelines for Reinspection of BWR Core Shrouds"

3.5.45. NUREG-1544, dated March 1996, "Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components"

3.5.46. Letter Vermont Yankee to USNRC, dated April 15, 1996, "Vermont Yankee Core Shroud Modification"

3.5.47. MPR-1730, Revision 0, dated April 1996, "Vermont Yankee Nuclear Power Station Core Shroud Repair – Design Report"

3.5.48. GE RICSIL No. 077, May 30, 1996, "Core Shroud Vertical Weld Cracking"

3.5.49. Operating Experience Information Review, dated June 10, 1996, "RICSIL077"

3.5.50. Memorandum D. C. Girroir to P. B. Corbett, July 10, 1996, "CAT A Item #GENLETR9403MEC4, Discuss Core Shroud with NRC"

3.5.51. VYS-046, Revision 2, dated July 26, 1996, "Design, Fabrication, and Installation Services for Reactor Pressure Vessel Core Shroud Repair at Vermont Yankee Nuclear Power Station"

3.5.52. Operating Experience Information Review, dated September 23, 1996, "Procedure Revisions To Reflect Core Shroud Modification"

3.5.53. BWRVIP-01, dated October 1996, "BWR Core Shroud Inspection and Flaw Evaluation Guideline, Revision 2"

3.5.54. Letter USNRC to Vermont Yankee, NVY 96-153, dated October 2, 1996, "Safety Evaluation Regarding the Vermont Yankee Core Shroud Repair (TAC No. M95207)"

3.5.55. Memorandum J. R. Hoffman to J. J. Duffy, dated October 8, 1996, "Review of USNRC SER for Shroud Repair"

3.5.56. Memorandum W. D. Fields to J. J. Duffy, dated November 25, 1996, "Core Shroud Repair Inspections"

3.5.57. Framatome Technologies Report dated December 18, 1996, "1996 Vermont Yankee Nuclear Power Corporation Project File Report for Core Shroud Examinations of the Vertical, Ring Segment, and H8/H9 Baffle Plate Welds"

3.5.58. Vermont Yankee Final Safety Analysis Report, Revision 14, dated November 5, 1997, Appendix K, "Core Shroud Repair"

3.5.59. Letter P. Butler (MPR) to D. Winterich (FTI), dated April 9, 1997, "BWR Core Shroud Repair Problems"

3.5.60. GE RICSIL No. 079, dated April 15, 1997, "Cracking of Vertical Welds On Core Shroud Outer Surface"

3.5.61. NRC Information Notice 97-17, dated April 17, 1997, "Cracking of Vertical Welds In the Core Shroud and Degraded Repair"

3.5.62. Memorandum D. C. Girroir to P. B. Corbett, dated May 18, 1997, "GE RICSIL 079 Response (AP 0028, CAT A Item)"

3.5.63. Memorandum D. C. Girroir to P. B. Corbett, dated June 18, 1997, "NRC Info Notice 97-17 (AP 0028 Item)"

3.5.64. Memorandum J. R. Hoffman to R. E. McCullough, dated September 5, 1997, "AP0028 for Core Shroud Tie Rod Retensioning"

3.5.65. Letter Vermont Yankee to NRC dated September 30, 1997, "Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals"

3.5.66. Letter NRC to Vermont Yankee dated March 25, 1998, "Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals – Vermont Yankee Nuclear Power Station"

3.5.67. BWRVIP-02, Revision 2, dated March 1999, "BWR Core Shroud Repair Criteria"

3.5.68. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"

3.5.69. BWRVIP-63, dated June 1999, "BWR Vessel and Internals Project Shroud Vertical Weld Inspection and Evaluation Guidelines"

3.5.70. Letter William Bateman, NRC, to Carl Terry, BWRVIP, dated August 13, 1999, "Staff Reevaluation of Table 1 in the BWRVIP-07 Report (TAC No. M94959)"

3.5.71. BWRVIP-76, dated November 1999, "BWR Vessel and Internals Project BWR Core Shroud Inspection and Evaluation Guidelines"

3.5.72. Letter NRC to BWRVIP, dated April 18, 2000, "Safety Evaluation of the "BWRVIP, Shroud Vertical Weld Inspection and Evaluation Guidelines (BWRVIP-63)," EPRI Report TR-113170, June 1999 (TAC No. MA6015)

3.5.73. BWRVIP-080, dated May 2000, "Evaluation of Crack Growth in BWR Shroud Vertical Welds"

3.5.74. Letter BWRVIP to all Committee Members, dated October 23, 2000, "Modification to Core Shroud I&E Guidelines (BWRVIP-63 and BWRVIP-76)"

3.5.75. Event Report 2001-2481, Dated December 3, 2001, "BWRVIP Documentation"

3.5.76. Action Item / Regulatory Commitment ER-2001-2481_01, dated January 9, 2002, "Investigate Options"

3.5.77. Action Item / Regulatory Commitment UND-2002-074_13, dated March 22, 2002, "Assess the need to inspect radial ring segment welds in the shroud flange and top guide support plates"

3.5.78. Action Item / Regulatory Commitment BWRVIP-04-A_01, dated June 6, 2002, "Evaluate BWRVIP-04-A and define solutions as required"

3.5.79. Action Item / Regulatory Commitment BWRVIP-04-A_02, dated June 20, 2002, "Revise PP 7027 – Shroud Repair Submittal Format"

3.5.80. Event Report 2003-0267, initiated January 31, 2003, "New RPV Internals Generated by GE for ARTS/MELLA Project that have Not Been Previously Considered by VY"

3.5.81. Letter USNRC to BWRVIP, dated February 19, 2003, "Safety Evaluation of 'BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Shroud Vertical Welds (BWRVIP-80)'"

3.5.82. Action Item / Regulatory Commitment BWRVIP-100_03, dated July 15, 2003

3.5.83. Technical Evaluation 2004-0018, "Justification to Inspect Portions of Shroud Horizontal Welds H1, H2, and H3 on the OD in Lieu of the Top Guide Spacer Block Welds, the Shroud Flange Ring Segment Welds, and the Top Guide Ring Segment Welds"

4.0 Core Shroud Support (Including Access Hole Cover)

4.1. BWRVIP Document Applicability

BWRVIP-38, published in September 1997, governs inspection of the core shroud support, with the exception of the access hole cover. Vermont Yankee has complied with this document as of its publication. The BWRVIP has not yet prepared an inspection and evaluation guideline that addresses the access hole cover.

In RFO 19 (1996) Vermont Yankee performed an inspection of welds H8 and H9 which meets the requirements of BWRVIP-38 for a baseline examination. The following describes the rationale for this statement. The baseline strategies for welds H8 and H9 are shown in Figures 3-4 and 3-5 of BWRVIP-38. The load multiplier is determined from Table 5-1. In Vermont Yankee's case this is 0.41. The flaw tolerance is determined from Figures 5-1 (for H8) and 5-2 (for H9) for plants with support legs. For both welds the flaw tolerance is 100%. The minimum examination coverage for a flaw tolerance of 100% is 10% for both H8 and H9. The coverage was 25% for weld H8 and 22% for weld H9 during the RFO 19 (1996) examination. No flaws were found. Therefore an adequate baseline of welds H8 and H9 was performed.

No welds other than H8 and H9 require examination in accordance with BWRVIP-38 for a plant with Vermont Yankee's core shroud support configuration.

The reinspection interval is established in BWRVIP-38, Paragraph 3.3.2, which states that, "if no flaws were found during the previous inspection, reinspections are performed on ten-year intervals if UT techniques were used..." The RFO 19 (1996) H8 and H9 examination was an ultrasonic test augmented with eddy current and no flaws were found. Therefore the reinspection interval is ten years if UT techniques are used, and six years if EVT-1 techniques are used (but see below). Accordingly, reinspection of H8 and H9 are due in RFO 25 (2005), nine years following the baseline exam. (Examination in RFO 26 (2007) would be six months late.)

BWRVIP-104 was issued in September 2002 to address extensive transverse cracking at the Tsuruga plant. BWRVIP-104, Section 9 revises the guidance of BWRVIP-38 for weld H9. Section 9.2.1 requires that an EVT-1 visual examination, or ultrasonic examination, of both top and bottom surfaces of the H9 weld. Therefore, the option in BWRVIP-38 to perform an EVT-1 examination of just the top surface at a six-year frequency is voided. Section 9.2.3 requires that the ultrasonic examination technique be capable of detection of both axial and circumferential flaws in the weld material and be able to determine whether the flaws have propagated into the RPV low alloy steel. This also effectively deletes the EVT-1 option. Section 9.2.5 further states, "An ultrasonic examination from the RPV ID is an acceptable alternative if OD access is limited (an OD exam is preferred). UT from the ID may require additional flaw evaluation or inspection sampling due to current limitations in flaw characterization in the low alloy steel." Per BWRVIP-94, new BWRVIP guidance is required to be implemented within two outages. Per PP 7027, Paragraph 4.2.1b, new BWRVIP guidance is required to be implemented within 16 months if it pertains to UT. This latter more restrictive requirement would require that VY implement the new BWRVIP-104 requirements in RFO 24(2004). However, no examination technique from the ID has been demonstrated to detect or size transverse flaws. Therefore, Technical Justification 2004-04 (Reference 4.5.39) was prepared because VY cannot meet BWRVIP-104. The BWRVIP has a 2004 budget and schedule item to attempt to demonstrate detection of transverse flaws from the vessel ID, but even if detection is demonstrated, it is highly unlikely that the technique will be able to determine if the flaws penetrate the RPV low alloy steel. Therefore, VY will postpone inspection of weld H9 until the originally scheduled time in RFO 25 (2005) and use the best demonstrated techniques available at that time. If a technique to detect transverse flaws from the ID becomes available, VY would reassess the feasibility of inspecting weld H9 accordingly at that time. Otherwise, the Technical Justification will remain in effect.

The BWRVIP stated in response to NRC SE Issue 3.2.2 (Reference 1.5.13) that when utilities have access to the lower plenum due to maintenance activities not related to the inspection recommendations of the BWRVIP, they will have the opportunity to perform a visual inspection of a portion of the lower plenum and that results of this inspection will be reported to the BWRVIP. This will be treated as a commitment for those items below the baffle plate listed in 4.2 below in the event that Relief Request RI-01 is accepted.

4.2. ASME Section XI Applicability

The core shroud support is part of the core support structure. Therefore the core shroud support will be examined in accordance with ASME Section XI, Table IWB-2500-1, Category B-N-2, Item B13.40, "Core Support Structure". Table IWB-2500-1 requires accessible surfaces to be visually inspected by the VT-3 method once per ten-year interval. This would normally include the upper side of the shroud support baffle plate and the shroud support shell course between welds H7 and H8. Such an examination was conducted during RFO 23 (2002) for the third ten-year Section XI interval. VY has submitted a Relief Request (RI-01) for the fourth ten-year Section XI interval that would allow using the BWRVIP guidance rather than the Section XI Categories B-N-1 and B-N-2 requirements. VY will perform inspections accordingly, based on the outcome of the Relief Request. The surfaces below the baffle plate are not accessible during the normal course of a refueling outage and would only be accessible if there were another reason to gain access below the core plate. The last time this area was accessible and, therefore, inspected was in 1983. Because this occurrence is so rare, any time that there is an opportunity for this Section XI inspection, it must be used. The under-core plate inspection would include accessible surfaces of the shroud support legs and their welds, and the underside of the shroud support baffle plate and its welds.

4.3. Other Commitments

SIL No. 462, Revision 1, (Reference 4.5.34 recommended ultrasonic examination of the access hole cover welds. SIL No. 462 Supplement 3 also recommended ultrasonic examination. A UT inspection has never been performed at Vermont Yankee; the oval shape of the weld does not lend itself to existing inspection tooling. In lieu of UT, Vermont Yankee has conducted visual inspections of the access hole cover welds every outage since at least 1993, with enhanced visual inspection performed in 1999. No relevant indications have ever been identified. An EVT-1 method should be specified, because the visual examination substitutes for what would normally be a UT examination.

Normally, VY would follow BWRVIP guidance for inspection of vessel internals components, with a nod to the guidance given in GE SILs. But the access hole cover is the only component important to safety for which the BWRVIP does not specify inspection requirements. In fact, BWRVIP-38 states that SIL 462 adequately addresses this area.

GE SIL 462, Revision 1, recommends the following inspection schedule:

- For a normal water chemistry (NWC) plant where the previous inspection was top surface VT-1 only and no crack indications were found, subsequent inspections, either top surface VT-1 or UT, should be performed during a refueling outage within 4 years of the previous inspection.

  For a NWC plant where the previous inspection was UT and no crack indications were found, subsequent inspections, either top surface VT-1 or UT, should be performed during a refueling outage within 6 years of the previous inspection.

- For a plant with an effective program of hydrogen water chemistry (HWC) or noble metal chemistry with HWC, a baseline UT inspection should be conducted according to the recommendation for subsequent inspections as noted above (dependent on the previous inspection method). Once the baseline has been established and no crack indications are found, future top surface VT-1 inspections should be conducted every 8 years and future UT inspection should be conducted every 12 years. (Effected HWC is then defined.)

If indications are found, the inspection frequency may change, depending upon structural analysis results.

It is notable that the SIL now gives guidance for plants that do not perform UT. It appears that under the GE recommendations for VY's circumstances, VY would have the following options:

- Perform the visual examination every other outage

- Perform a baseline UT and then a visual reexamination every eight years thereafter

- Perform a baseline UT and then a UT reexamination every twelve years thereafter

The position can be taken that ultrasonic examination of the access hole welds is not necessary. BWRVIP-38 allows an enhanced VT-1 examination of the H8 and H9 shroud support welds as an alternative to ultrasonic examination. It is evident that the H8 and H9 welds are more safety significant than the access hole cover welds, because they also provide structural support to the shroud – in addition to providing containment for 2/3 core height. The access hole cover welds only provide containment for 2/3 core height.

Vermont Yankee will adopt SIL 462, Revision 1, as the guidance for examination of the access hole cover welds. This will allow VY to perform visual examination of these welds every other refueling outage. However, VY will maintain the position that these welds should be examined by enhanced VT-1. This commitment will be honored until the BWRVIP issues guidance for the inspection of these welds.

4.4. Inspections for Risk to Generation Purposes

Vermont Yankee intends to continue to inspect the shroud support baffle plate (the annulus floor) for debris and loose parts. This is typically performed just prior to vessel reassembly near the end of each refueling outage. Although nonmandatory, this inspection provides a significant benefit with regard to assurance of fuel clad integrity.

The two shroud support vertical seam welds located between H7 and H8 will be examined by EVT-1 in RFO 25 (2005). Re-examination will be 100% each cycle, which means both seam welds will be inspected every 6 years (EOI 2011) as prescribed by BWRVIP-76, Section 2.3.3 for a one sided visual examination. This examination is being performed as a lesson learned from the JAF IVVI Program audits.

4.5. References

4.5.1. GE SIL No. 462, dated February 1, 1988, "Shroud Support Access Hole Cover Cracks"
4.5.2. Information Notice 88-03, dated February 2, 1988, "Cracks In Shroud Support Access Hole Cover"
4.5.3. Memorandum J. R. Hoffman to S. R. Miller OPVY 157/88, dated February 25, 1988, "NRC Meeting – Shroud Access Hole"
4.5.4. Memorandum J. R. Hoffman to S. R. Miller/D. A. Reid, March 7, 1988, "Reactor Pressure Vessel Shroud Access Hole Cover Cracking"
4.5.5. Letter T. Wilders (GE) to J. P. Pelletier, March 10, 1988, "Access Hole Cover Cracking"
4.5.6. Letter T. Wilders (GE) to J. C. Brooks, June 13, 1988, "In-vessel Inspections"
4.5.7. GE SIL No. 462, Supplement 1, dated February 22, 1989, "Shroud Support Access Hole Cover Cracks"
4.5.8. Memorandum C. B. Cameron to R. E. McCullough, dated March 7, 1989, "Preliminary Review to Supplement 1 of SIL-462"
4.5.9. Memorandum J. C. Brooks to R. D. Pagodin, May 12, 1989, "Review of SIL 462 Supplement 1"
4.5.10. GE SIL No. 462, Supplement 2, August 10, 1990, "Shroud Support Access Hole Cover Cracks"
4.5.11. Memorandum C. B. Cameron to B. R. Buteau, November 16, 1990, "Request for Extension to Commitment SIL0462S2"
4.5.12. GE SIL No. 462, Supplement 2, Revision 1, December 19, 1990, "Shroud Support Access Hole Cover Cracks"
4.5.13. Memorandum C. B. Cameron to R. D. Pagodin, February 4, 1991, "Commitment SIL0462S2"
4.5.14. Memorandum C. B. Cameron to D. C. Porter, March 1, 1991, "Service Request, Shroud Support Access Hole Covers"
4.5.15. Memorandum S. K. Naeck to S. R. Miller, April 2, 1991, "Service Request 91-21, Shroud Support Access Hole Covers"
4.5.16. Letter C. B. Cameron to A. D. Himle, dated July 22, 1991, regarding SIL 462, RICSIL054, RICSIL059 and GE support for possible inspection findings

4.5.17. Memorandum C. B. Cameron to R. E. McCullough, September 11, 1991, "Response to Commitment SIL0462S2RE1"

4.5.18. Calculation VYC-1021, Revision 0, dated November 15, 1991, "Loss of Shroud Access Hole Cover Analysis"

4.5.19. Letter A. D. Himle (GE) to C. B. Cameron, January 13, 1992, "Engineering Support of ISI Work at Vermont Yankee"

4.5.20. Memorandum C. B. Cameron to R. E. McCullough, January 30, 1992, "Response to Commitment SIL0462S2RE3"

4.5.21. GE SIL No. 462 Supplement 3, June 8, 1992, "Radial Cracking In Creviced Incone 600 Access Hole Cover Weldments"

4.5.22. NRC Information Notice 92-57, dated August 11, 1992, "Radial Cracking of Shroud Support Access Hole Cover Welds"

4.5.23. Memorandum C. B. Cameron to R. E. McCullough, September 16, 1992, "Commitment SIL0462S3"

4.5.24. BWROG Report GE-NE-523-107-0892, dated October 1992, "BWR Access Hole Cover Radial Cracking Evaluation"

4.5.25. Framatome Technologies Report dated December 18, 1996, "1996 Vermont Yankee Nuclear Power Corporation Project File Report for Core Shroud Examinations of the Vertical, Ring Segment, and H8/H9 Baffle Plate Welds"

4.5.26. BWRVIP-38, dated September 1997, "Shroud Support Inspection and Flaw Evaluation Guidelines"

4.5.27. Letter Vermont Yankee to NRC dated September 30, 1997, "Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals"

4.5.28. Letter NRC to Vermont Yankee dated March 25, 1998, "Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals – Vermont Yankee Nuclear Power Station"

4.5.29. BWRVIP-52, dated June 1998, "Shroud Support and Vessel Bracket Repair Design Criteria"

4.5.30. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"

4.5.31. GE SIL No. 624, March 24, 2000, "Stress Corrosion Cracking In Alloy 182 Welds In Shroud Support Structure"

4.5.32. Memorandum, C. B. Larsen to D. C. Girroir, dated May 11, 2000, "Response to Commitment SIL-0624_00"

4.5.33. Letter NRC to BWRVIP dated March 1, 2001, "Acceptance for Referencing of BWRVIP, Shroud Support Inspection and Flaw Evaluation Guidelines (BWRVIP-38), and Appendix B, Demonstration of Compliance with the Technical Information Requirements of the License Renewal Rule"

4.5.34. GE SIL No. 462, Revision 1, dated March 22, 2001, "Access Hole Cover Cracking"

4.5.35. Action Item/Regulatory Commitment SIL-0462R1_01, dated March 27, 2001, "Evaluate SIL No. 462, Rev. 1, 'Access Hole Cover Cracking'"

4.5.36. BWRVIP-104, dated September 2002, "BWRVIP Evaluation and Recommendations to Address Shroud Support Cracking in BWRs"

4.5.37. Action Item / Regulatory Commitment BWRVIP-104_01, initiated September 30, 2002, "Evaluate BWRVIP-104 for new commitments in accordance with BWRVIP-94"

4.5.38. Action Item / Regulatory Commitment BWRVIP-104_02, initiated November 20, 2002, "Revise PP 7027 to require examination of the shroud support H9 weld in accordance with the revised guidance of BWRVIP-104"

4.5.39. Technical Justification TJ-2004-04, dated March 26, 2004, "Justification to Defer Inspection for Detection of Transverse Flaws in Shroud Support Weld H9"

5.0 Core Spray Internal Piping and Spargers

5.1. BWRVIP Document Applicability

BWRVIP-18, published in July 1996, governs inspection of the core spray system internal to the vessel. Vermont Yankee has complied with this document as of its publication. Additionally, Vermont Yankee has committed to its use in References 5.5.18 and 5.5.20 as further described below. Per BWRVIP-94, letters from the BWRVIP Executives to the NRC are also considered mandatory. To that end, Vermont Yankee has also complied with the NRC's Final Safety Evaluation on BWRVIP-18 (Reference 5.5.22), with the exception that the core spray piping and sparger brackets were not inspected every two cycles per that letter. Event Report 2001-2480 (Reference 5.5.28) was initiated and the corrective action was to inspect these brackets RFO 23 (2002).

BWRVIP-48, published in February 1998, governs inspection of the core spray bracket attachment welds. Vermont Yankee has complied with this document as of its publication.

5.1.1. Thermal Sleeve Welds

These welds are currently inaccessible for inspection, but per BWRVIP-18, Paragraph 3.2.4, inspection is recommended when a technique becomes available. Because a technique still does not exist, VY has complied with this document as of its publication. Inspection of 100% of these welds would be required immediately upon development of a technique, considering scheduling as allowed under PP 7027.

Until such time as an inspection technique is available, BWRVIP-18, Section 3.2.4 "Hidden Welds", states..."a qualitative assessment of thermal sleeve integrity can be based on a plant-specific evaluation of similar core spray piping welds. If a plant has uncreviced thermal sleeve welds, the evaluation welds should be the junction box-to-pipe welds and the upper elbow welds. If the thermal sleeve welds are creviced, the evaluation welds should be the junction box cover plate weld, where applicable, the P1 weld in BWR/3-5 plants where accessible for inspection, and the downcomer sleeve welds." Regardless of whether VY's thermal sleeve welds are creviced, none of the above "evaluation welds" at VY show any indications of cracking. Therefore, the qualitative assessment of the core spray thermal sleeve welds is satisfactory (UND 2002-074_08). A Technical Justification in accordance with PP 7027, Paragraph 4.2.3 and BWRVIP-94 is in the course of preparation to defer examination of these welds until such time that tooling and an NDE technique become available.

BWRVIP-18, Paragraph 3.2.4 references Figure 3-1 for inspection requirements, which references the reinspection flowchart (Figure 3-3).

BWRVIP-18, Paragraph 2.2.1, states that most thermal sleeve welds are full penetration welds, but that some are creviced fillet welds, and at least one is a creviced partial penetration weld. Then from the way that is worded, full penetration thermal sleeve welds would be considered to be non-creviced. Vermont Yankee has three welds upstream of P1 in each of two nozzles that are full penetration butt welds:

- A tuning fork-to-10" schedule 40 pipe weld
- A 10" pipe-to-10" to 8" std. weight concentric reducer weld
- A 10" to 8" reducer-to-8" schedule 40 pipe weld

These six welds will be inspected as part of the 25% target weld sample on a rotating basis with the other 16 non-creviced welds. Therefore, of the 22 non-creviced welds, if only UT was used, five or six non-creviced welds would be inspected every other cycle, and the six hidden welds would be inspected all together every eighth cycle for convenience. The same will be true – inspection of the thermal sleeve welds every eight cycles – if the bulk of the core spray welds are inspected with the EVT-1 method every cycle (see below).

5.1.2. Internal Piping

A full baseline inspection of the core spray piping was performed in RFO 19 (1996) with the majority performed with the UT method. BWRVIP-18, Paragraph 3.3, specifies that certain target welds be reinspected. Target welds are defined as all of the creviced welds, the tee-box-to-pipe welds, and 25% of the non-creviced welds. For Vermont Yankee there are 24 creviced welds, four tee-box-to-pipe welds, and 16 non-creviced welds (22 non-creviced welds counting the thermal sleeves). The BWRVIP core spray piping reinspection frequency for ultrasonic inspection is two cycles, and for EVT-1 it is one cycle. Consequently, the welds that were examined visually in RFO 19 (1996) were reexamined visually in RFO 20 (1998). Of the 44 welds that were inspected ultrasonically and visually in RFO 19 (1996), 32 required reinspection in RFO 21 (1999); these examinations were performed visually. However, two creviced welds, AP1 and BP1, are essentially inaccessible for visual examination, so only a best effort inspection was performed.

In RFO 19 (1996), core spray piping welds 1P8b and 3P8b were found to be flawed by UT. Vermont Yankee received permission from the NRC to forgo UT reinspection of those welds in RFO 20 (1998) and RFO 21 (1999); however those welds were examined using EVT-1. The 1P9 and 3P9 welds are redundant to the two flawed welds, so in RFO 22 (2001) UT examination of all four P9 welds was attempted using UT. Welds 1P9, 2P9, and 3P9 were found to have indications, and were found to be acceptable for continued service (Reference 5.5.26), but the UT examination technique was subsequently disqualified by the BWRVIP in the spring of 2002 (References 5.5.36, 5.5.39, and 5.5.40). Experimentation at FRA-ANP using newly-built BWRVIP mockups determined that ultrasound was never entering the weld examination volume. Therefore, the RFO 22 (2001) P9 UT examination was ruled invalid; the P9 welds have been determined to never have undergone inspection to date; and the P9 welds are now assumed to be flaw-free (indications in the RFO 22 (2001) UT data are from component geometry or from some other non-flaw source). EPRI performed a comparison of the 1P8b and 3P8b UT data from RFO 19 (1996) and RFO 22 (2001) and it was determined that none of the indications on those two welds had changed over those three cycles (Reference 5.5.41). Vermont Yankee will examine the P9 welds when an examination technique becomes available, but until that time will perform EVT-1 examination of the P8b welds.

In the future, the creviced welds, the four tee-box-to-pipe welds, and 25% (five or six) of the 22 non-creviced welds will be inspected with either the EVT-1 method or the UT method. The inspection frequency will depend on the inspection method chosen: one cycle for EVT-1 or two cycles for UT.

5.1.3. Spargers

Vermont Yankee informed the NRC in References 5.5.18 and 5.5.20 below that VY will be following the BWRVIP-18 inspection guidelines rather than IE Bulletin 80-13 for the core spray spargers. In RFO 20 (1998), following the published BWRVIP-18 guidance for a geometry-tolerant plant, Vermont Yankee performed a modified VT-1 (with cleaning) of the core spray sparger circumferential welds and a VT-3 of the nozzles and brackets. No cracking was found. Since that time, the BWRVIP has agreed with the NRC to revise BWRVIP-18 and discontinue the designation of geometry-tolerant plant status. Therefore, in the future Vermont Yankee will perform EVT-1 (see reference 5.5.19) inspection of the sparger circumferential and bracket welds and a VT-1 inspection of the nozzle welds. In References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to inspect all the major circumferential welds and 50% of the nozzle welds in the core spray spargers every other refueling outage.

BWRVIP-18, in Figure 3-2, identifies the method of inspection for the large circumferential welds as CS VT-1. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the CS VT-1 and MVT-1 methods with the EVT-1 method. These welds were inspected with the MVT-1 method during RFO 20 (1998). BWRVIP-18, in Figure 3-2, identifies the method of inspection for the nozzle welds for geometry tolerant plants as VT-3. Because the BWRVIP dropped the distinction of geometry-tolerant plants, the BWRVIP committed to the NRC to inspect the nozzle welds by the VT-1 method. The nozzle welds were inspected with the VT-3 method in RFO 20 (1998). The large circumferential and nozzle welds were inspected in accordance with the BWRVIP document that was published at the time. In the future, these welds will be inspected in accordance with the revised philosophy.

A repair was installed on the sparger "C" tee-box during RFO 8 (1980) or RFO 9 (1981) to address cracking of the tee-box cover plate. BWRVIP-19, Section 10.2.3 states, "Inspections required for the entire repaired internal core spray piping and sparger assembly for the remaining life of the unit shall be specified commensurate with design considerations and Code requirements applicable to the specific design." Since the repair was installed prior to the existence of the BWRVIP, no ongoing inspection requirements were originally established. The tee-box repair has received a VT-3 inspection every refueling outage for the most part from its installation through RFO 21 (1999). However, BWRVIP-18, Section 3.3.3, under "Repairs", states, "For bolted repairs, reinspection should be with the same methods described for the baseline in Section 3.2.4". Since the spargers require VT-1, the repair will be inspected by VT-1. Section 3.3.3 also states, "Reinspection of bolted repairs should be every 2 cycles unless cracking or damage is found".

5.1.4. Piping and Sparger Brackets

Vermont Yankee has informed the NRC in References 5.5.18 and 5.5.20 below that VY will be following the BWRVIP-18 guidelines rather than IE Bulletin 80-13 for the core spray spargers and their brackets. The piping brackets were inspected in accordance with BWRVIP-18 during RFO 19 (1996), and no cracking was found. The sparger brackets were inspected in accordance with BWRVIP-18 during RFO 20 (1998), and no cracking was found. BWRVIP-18, Section 3.3.3, states that if there is no cracking, reinspection of piping and sparger brackets every four cycles is appropriate. However, in a response to the NRC Safety Evaluation on BWRVIP-18 (Reference 5.5.19), the BWRVIP states that the sparger brackets should be inspected every other cycle. Because the sparger brackets were not inspected after two cycles, Event Report 2001-2480 (Reference 5.5.28) was initiated and the corrective action was to inspect these brackets in RFO 23 (2002) and every other cycle in the future.

BWRVIP-48, Table 3-2, applies for the piping bracket vessel attachment welds. The inspection frequency for these welds is listed as every four cycles.

BWRVIP-18 identifies the method of inspection for the core spray piping brackets in Section 3.2.4 to be CS VT-1. BWRVIP-48, Table 3-2, identifies the method of inspection to be modified VT-1. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the CS VT-1 and MVT-1 methods with the EVT-1 method. In those same letters, the inspection method of core spray sparger brackets was changed to VT-1. Future inspections of the core spray piping brackets will be by the EVT-1 method. Future inspections of the core spray sparger brackets will be by the VT-1 method.

5.2. ASME Section XI Applicability

Inspection of the core spray piping bracket attachment welds is also governed by ASME Section XI, Table IWB-2500-1, Category B-N-2, Item No. B13.30, "Interior Attachments Beyond Beltline Region," which requires a VT-3 inspection once each ten-year interval, typically performed at the end of the interval. The method and frequency of inspections given above by the BWRVIP requirements exceed the ASME Section XI requirements. Therefore, this Program will consider ASME Section XI requirements to be satisfied by performing inspection of the core spray piping bracket attachment welds in accordance with BWRVIP requirements. Additionally, VY has submitted a Relief Request (RI-01) for the fourth ten-year Section XI interval that would allow using the BWRVIP guidance rather than the Section XI Categories B-N-1 and B-N-2 requirements.

5.3. Other Commitments

In Reference 5.5.13 below, Vermont Yankee committed to perform inspection of the core spray spargers during RFO 19 (1996) in accordance with Bulletin 80-13, however, it was indicated that Vermont Yankee intended, in the future, to perform such inspections in accordance with BWRVIP-18. In Reference 5.5.18 below, Vermont Yankee did indeed commit to follow the BWRVIP Guidelines for core spray spargers and their brackets. This commitment took effect with RFO 20 (1998).

5.4. Inspections for Risk to Generation Purposes – None.

5.5. References

5.5.1. GE SIL No. 289, Revision 0, dated February 1, 1979, "Core Spray Piping Visual Inspection"

5.5.2. NRC Bulletin 80-13, dated May 12, 1980, "Cracking In Core Spray Spargers"

5.5.3. EDCR 80-52, dated October 30, 1980 with Change No. 1 dated November 11, 1980, Change No. 2 dated December 12, 1980, and Change No. 3 dated March 4, 1982, "Design and Installation of Clamping Device for Core Spray Sparger Junction Box 'C'"

5.5.4. Letter USNRC to Vermont Yankee, NVY 80-qq, dated November 5, 1980, "Summary of Meeting Held On October 31, 1980 To Discuss Vermont Yankee Core Spray Sparger Cracking"

5.5.5. Letter Vermont Yankee to USNRC, WVY 80-164, dated December 1, 1980, Results of Core Spray Sparger Inspection"
5.5.6. GE SIL No. 289, Revision 1, Supplement 1, dated February 23, 1989, "Core Spray Piping Visual Inspection"
5.5.7. GE SIL No. 289, Revision 1, Supplement 1, Revision 1, dated March 15, 1989, "Core Spray Piping Visual Inspection"
5.5.8. Letter J. C. Brooks to R. D. Pagodin, dated May 3, 1989
5.5.9. GE RICSIL No. 074, dated November 1, 1995, "Cracking In Core Spray Piping"
5.5.10. GE SIL No. 289, Revision 1 Supplement 2, dated January 5, 1996, "Cracking In Core Spray Piping"
5.5.11. Memorandum D. C. Girroir to P. B. Corbett, May 1, 1996, "Core Spray Piping Weld, CAT A Items (SIL0289R1_S2, RICSIL074)"
5.5.12. BWRVIP-18, dated July 1996, "BWR Core Spray Inspection and Evaluation Guidelines"
5.5.13. Letter Vermont Yankee to USNRC, BVY 96-110, dated September 25, 1996, "Core Spray System Inspection at Vermont Yankee"
5.5.14. Letter USNRC to Vermont Yankee, NVY 96-176, dated November 20, 1996, "Review of Core Spray System Collar-to-Shroud Weld Flaw Evaluation and Core Spray System Inspection Plan at Vermont Yankee Nuclear Power Station (TAC Nos. M96671 and M96689)"
5.5.15. Memorandum J. R. Hoffman to J. J. Duffy, dated December 12, 1996, "Review of NRC SER for Core Spray Collar Cracking"
5.5.16. BWRVIP-16, dated March 1997, "Internal Core Spray Piping and Sparger Replacement Design Criteria"
5.5.17. Letter Vermont Yankee to USNRC, dated September 30, 1997, "Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals"
5.5.18. Letter USNRC to Vermont Yankee, dated March 25, 1998, "Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals – Vermont Yankee Nuclear Power Station"
5.5.19. Letter BWRVIP to USNRC, dated January 11, 1999, "BWRVIP Response to NRC Safety Evaluation of BWRVIP-18"
5.5.20. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"
5.5.21. Letter USNRC to BWRVIP, dated September 29, 1999, "Final Safety Evaluation of 'Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (BWRVIP-48)'"
5.5.22. Letter USNRC to BWRVIP, dated December 2, 1999, "Final Safety Evaluation of Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)"
5.5.23. Letter Vermont Yankee to USNRC, dated February 14, 2000, "Vermont Yankee's Plans for Reactor Vessel Internal Core Spray Piping"
5.5.24. Letter NRC to BWRVIP, dated December 7, 2000, "Acceptance for Referencing of BWRVIP, BWR Core Spray Internals Inspection and Evaluation Guidelines (BWRVIP-18) Report for Compliance with the License Renewal Rule (10 CFR Part 54)"
5.5.25. Framatome ANP UT Exam Report, dated May 11, 2001, "Core Spray Piping P9/P8b Weld Examination Field Report for Vermont Yankee", Revision 0

5.5.26. Technical Evaluation No. 2001-029, dated May 14, 2001, "Evaluation of Internal Core Spray Piping Flaws"

5.5.27. Event Report 2001-2479, initiated December 3, 2001, "BWRVIP Cleaning Requirements"

5.5.28. Event Report 2001-2480, initiated December 3, 2001, "Scheduling of BWRVIP Core Spray Piping Brackets"

5.5.29. Action Item / Regulatory Commitment ER-2001-2479_01, dated January 22, 2002, "Revise NE 8048"

5.5.30. Action Item / Regulatory Commitment ER-2001-2480_01, dated January 22, 2002, "Revise PP 7027 – revision applies to Core Spray inspection scope"

5.5.31. Action Item / Regulatory Commitment UND-2002-074_05, dated March 21, 2002, "Perform an EVT-1 of core spray piping bracket vessel attachment welds"

5.5.32. Action Item / Regulatory Commitment UND-2002-074_06, dated March 21, 2002, "Schedule an inspection of the core spray sparger bracket in accordance with BWRVIP-18"

5.5.33. Action Item / Regulatory Commitment UND-2002-074_08, dated March 21, 2002, "Provide a qualitative evaluation for inaccessible core spray welds in accordance with BWRVIP-18"

5.5.34. Action Item / Regulatory Commitment UND-2002-074_09, dated March 21, 2002, "Ensure that components with crud buildup are sufficiently cleaned"

5.5.35. Action Item / Regulatory Commitment UND-2002-074_10, dated March 21, 2002, "Improve the timeliness and review of vendor NDE activities during outage activities"

5.5.36. Memorandum D. C. Girroir to J. Dreyfuss, dated May 9, 2002, "Core Spray P9 Weld Status"

5.5.37. Action Item / Regulatory Commitment BWRVIP-006-A_01, dated June 6, 2002, "Evaluate BWRVIP-06-A and define solutions as required"

5.5.38. Action Item / Regulatory Commitment BWRVIP-006-A_02, dated June 20, 2002, "Reactor Internals Modifications prior to BWRVIP guidance"

5.5.39. Event Report 20022877, initiated December 3, 2002, "CS P9 Weld UT Technique Qualification Revocation"

5.5.40. BWRVIP-03, Revision 5, December 2002, "BWRVIP Examination Guidelines"

5.5.41. Letter EPRI to Vermont Yankee, dated December 15, 2003, "Review of Ultrasonic Inspection Information for Vermont Yankee Core Spray Internal Piping Welds 1P8b and 3P8b"

5.5.42. Technical Justification 2004-02, dated 03/26/04, "Justification for Deferral of Inspection of Inaccessible Welds"

6.0 Feedwater Spargers

6.1. BWRVIP Document Applicability

No BWRVIP Inspection and Evaluation document addresses the feedwater sparger, which is considered a non-safety related component, with the exception that BWRVIP-48, published in February 1998, governs inspection of the reactor vessel internal attachment welds. Vermont Yankee has complied with this document as of its publication. BWRVIP-48, Table 3-2, states that, "No additional inspections (for the feedwater sparger bracket attachments) are required above those specified in a plant's ASME Section XI program." One exception is listed in BWRVIP-48, Table 3-2, which requires that feedwater bracket attachment welds which use furnace-sensitized stainless steel or Alloy 182 material be examined by the modified VT-1 method. The inspection frequency is per ASME Section XI, Table IWB-2500-1, Category B-N-2, and this end-of-interval inspection will be performed in RFO 23 (2002). The reactor vessel was heat treated subsequent to welding of these attachment pads. There is no evidence at this time that the feedwater bracket attachment welds were not furnace-sensitized. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method. Therefore the feedwater bracket attachment welds were inspected with the EVT-1 method in RFO 23 (2002) for the Third Interval inspection, and will be examined likewise in the Fourth Interval.

6.2. ASME Section XI Applicability

Inspection of the feedwater sparger bracket welds is also governed by ASME Section XI, Table IWB-2500-1, Category B-N-2, Item No. B13.30, "Interior Attachments Beyond Beltline Region," which requires a VT-3 inspection once each ten-year interval, typically performed at the end of the interval. However, the BWRVIP requirement above exceeds this requirement. Therefore, this Program will consider ASME Section XI requirements to be satisfied by performing inspection of the feedwater bracket attachment welds in accordance with the BWRVIP. Additionally, VY has submitted a Relief Request (RI-01) for the fourth ten-year Section XI interval that would allow using the BWRVIP guidance rather than the Section XI Categories B-N-1 and B-N-2 requirements.

6.3. Other Commitments

In References 6.5.5, 6.5.6, and 6.5.7 below Vermont Yankee commits to continue to perform visual examinations of the feedwater spargers on a two-cycle frequency. The visual inspections are performed in accordance with NUREG-0619, which in Table 2 requires, for plants with interference fit spargers and cladding, a visual inspection of the flow holes and welds in sparger arms and sparger tees. It requires a VT-3 of the sparger piping, spacer brackets, and end brackets, and a VT-1 of the tee welds and end bracket-to-vessel weld. BWROG report GE-NE-523-A71-0594-A, Revision 1, (Reference 6.5.20) was issued in May 2000 to formalize substitution of UT for PT of the feedwater nozzle inner radius area. The use of that document by VY for nozzle inner radius examination does not alter VY's commitments for visual inspection of the spargers and brackets.

6.4. Inspection for Risk to Generation Purposes – None.

6.5. References

6.5.1. EDCR 75-30, dated June 28, 1976, "Feedwater Sparger Replacement"

6.5.2. NUREG-0619, dated November 1980, "BWR Feedwater Nozzle and CRD Return Line Nozzle Cracking"

6.5.3. Letter Vermont Yankee to USNRC, FVY 86-29, dated March 28, 1986, "Request for Revision of Routine Inspection Interval Guidance Provided by NUREG-0619, Based on Accumulated Plant-Specific Experience"

5.4. Letter USNRC to Vermont Yankee, NVY 86-73, dated April 18, 1986, "Alternate Inspection of Feedwater Nozzle for the 1986 Refueling Outage"

6.5.5. Letter Vermont Yankee to USNRC, FVY 87-02, dated January 5, 1987, "Request for Permanent Revision of Routine Inspection Interval Guidance Provided by NUREG-0619 for Feedwater Nozzle PT Examinations"

6.5.6. Letter Vermont Yankee to USNRC, FVY 87-60, dated June 2, 1987, "Request for Permanent Revision of Routine Inspection Interval Guidance Provided by NUREG-0619 for Feedwater Nozzle PT Examinations – Response to Request for Additional Information"

6.5.7. Letter Vermont Yankee to USNRC, BVY 94-07, dated February 11, 1994, "Request for Relief from NUREG-0619 Inspection Requirements"

6.5.8. Letter USNRC to Vermont Yankee, NVY 94-157, dated September 9, 1994, "Summary of August 30, 1994, Meeting with Representatives of Vermont Yankee Nuclear Power Corporation"

6.5.9. Letter Vermont Yankee to USNRC, BVY 94-110, dated November 8, 1994, "Feedwater Nozzle Inspection Relief Request"

6.5.10. Letter USNRC to Vermont Yankee, NVY 95-16, dated December 29, 1994, "Inspection Report No. 50-271/94-29"

6.5.11. Letter Vermont Yankee to USNRC, BVY 95-08, dated January 19, 1995, "Feedwater Nozzle Inspection Relief Request – Supplementary Information"

6.5.12. Letter USNRC to Vermont Yankee, NVY 95-02, dated February 6, 1995, "Evaluation of Request for Relief from NUREG-0619 for Vermont Yankee Nuclear Power Station"

6.5.13. Letter Vermont Yankee to USNRC, BVY 95-78, dated July 14, 1995, "Feedwater Nozzle Inspection Technique Qualification – Final Report"

6.5.14. Letter USNRC to Vermont Yankee, NVY 95-142, dated October 12, 1995, "Feedwater Nozzle Inspection Relief Request – Vermont Yankee Nuclear Power Station"

6.5.15. Letter USNRC to Vermont Yankee, NVY 96-182, dated December 5, 1996, "Erratum To the Safety Evaluation of Vermont Yankee Nuclear Power Corporation's Request for Relief from NUREG-0619 Feedwater Nozzle Inspection Requirements – Vermont Yankee Nuclear Power Station"

6.5.16. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"

6.5.17. Memorandum C. B. Larsen to D. C. Girroir, dated August 27, 1999, "Future Examinations of Feedwater Nozzle Inner Radii with Regard To Proposed BWROG NUREG 0619 Relief"

6.5.18. Letter USNRC to BWRVIP, dated September 29, 1999, "Final Safety Evaluation of 'Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (BWRVIP-48)'"

6.5.19. Letter USNRC to BWR Owners' Group, dated March 10, 2000, "Final Safety Evaluation of BWR Owner's Group Alternate Boiling Water Reactor (BWR) Feedwater Nozzle Inspection (TAC No. MA6787)"

6.5.20. BWR Owners' Group Report GE-NE-523-A71-0594-A, Revision 1, dated May 2000, "Alternate BWR Feedwater Nozzle Inspection Requirements"

6.5.21. Letter Vermont Yankee to USNRC, dated January 22, 2001, BVY 01-02, "Alternative Feedwater Nozzle Inspection"

7.0 Guide Rods

7.1. BWRVIP Document Applicability

No BWRVIP Inspection and Evaluation document addresses the guide rods, which are considered non-safety related components, with the exception that BWRVIP-48, published in February 1998, governs inspection of the reactor vessel internal attachment welds. Vermont Yankee has complied with this document as of its publication. The requirements for the guide rod attachment welds are found in the Miscellaneous Vessel Internal Attachments section.

7.2. ASME Section XI Applicability

Inspection of the guide rod attachment welds is also governed by ASME Section XI, Table IWB-2500-1, Category B-N-2, Item No. B13.30, "Interior Attachments Beyond Beltline Region." The requirements for the guide rod attachment welds are found in the Miscellaneous Vessel Internal Attachments section.

7.3. Other Commitments – None.

7.4. Inspection for Risk to Generation Purposes

Representatives from Reactor Engineering and Plant Engineering met on January 13, 1999, and agreed that these components are non-safety related. However, the group agreed that some inspection might be warranted for risk to generation reasons. The guide rods are used to position the shroud head and separator for reinstallation. If the guide rods were bent or broken off of their attachments, it would hamper this process. If the guide rods were damaged in this manner during operation, this damage would probably be detectable during disassembly. Nevertheless, a VT-3 inspection of the guide rods would not take considerable time and could be performed in conjunction with the vessel attachment welds. Therefore, this nonmandatory inspection would be performed at the same time as the ten-year vessel attachment weld ISI. Such an inspection was performed during RFO 23 (2002).

7.5. References – None.

8.0 Incore Flux Monitors (Including Housings, Guide Tubes, Dry Tubes)

8.1. BWRVIP Document Applicability

BWRVIP-47, published in December 1997, governs inspection of the incore flux monitor housing, guide tube, and dry tube assemblies. However, BWRVIP-47 considers the incore flux monitor housing, guide tube, and dry tube assemblies as non-safety related and does not identify any inspection for these components. Therefore, Vermont Yankee has complied with this document as of its publication.

However, the BWRVIP stated in response to NRC SE Issue 3.2.2 (Reference 1.5.13) that when utilities have access to the lower plenum due to maintenance activities not related to the inspection recommendations of the BWRVIP, they will have the opportunity to perform a visual inspection of a portion of the lower plenum and that results of this inspection will be reported to the BWRVIP. This will be treated as a commitment for those items listed in 8.4 below.

8.2. ASME Section XI Applicability – None.

8.3. Other Commitments

There are ten dry tubes at Vermont Yankee. Nine of the ten were replaced in RFO 12 (1986) and the remaining dry tube was replaced in RFO 18 (1995).

Reference 8.5.4 below consists of an internal commitment to inspect dry tubes following six refueling outages after their installation. The inspection was conducted in RFO 18 (1995) when three dry tubes were inspected. Reference 8.5.11 below consists of an internal commitment to perform inspection of three dry tubes every third outage. Representatives from Reactor Engineering and Plant Engineering met on January 13, 1999, and agreed that these components are non-safety related. However, the group decided to continue these inspections, but to decrease the population size. It was decided to perform inspection of two dry tubes every third outage from that date forward. Two dry tubes were inspected in RFO 21 (1999) so two dry tubes will again be inspected in RFO 24 (2004). SIL 409, Revision 2, (Reference 8.5.16) recommended that for dry tubes of the newer design with noncreviced welds and better material, the dry tubes be inspected at an increased frequency after they reach 20 years of age. Reference 8.5.18 confirmed that the dry tubes are of the newer design. Commitment SIL-409R2_02 (Reference 8.5.19) was generated to revise PP 7027 to perform inspection of dry tubes every other refueling cycle (50% every cycle) after they reach 20 years of service life. Starting in RFO 25 (2005), 50% of the dry tubes that are 20 years old will be inspected every refueling outage (five dry tubes one outage, four dry tubes the next, and so on until all dry tubes are 20 years old). These commitments are only internal commitments and could be changed or deleted in the future. Therefore, the inspections are only mandatory in that sense, although the dry tubes are considered a risk-to-generation component because they form the pressure boundary of the vessel.

The method of inspection is determined from GE SIL No. 409 (Reference 8.5.2). The top two feet of the dry tube assembly is inspected with the VT-1 method and the remainder of the dry tube assembly is inspected with the VT-3 method. For the VT-1 method, the dry tube is inspected from all four adjacent fuel bundle locations, because of the 30-degree rule. For the VT-3 method, the dry tube need only be inspected from two fuel bundle locations diagonally opposite from each other.

8.4. Inspection for Risk to Generation Purposes

At the same meeting mentioned above, incore instrumentation housing and guide tube inspection was discussed. It was agreed that these components are also non-safety related. Inspection of these components by themselves would be very costly and time consuming, because they are located below the core plate and core disassembly would be required. The group agreed to only perform inspection of these components if they were made accessible through other vessel activities. The last time this area was accessible and, therefore, inspected was in 1983. Because this occurrence is so rare, any time that there is an opportunity for inspection, it should be used. This nonmandatory under-core plate inspection would include accessible incore housing-to-vessel welds, incore housing-to-guide tube welds, and incore guide tube stabilizers.

8.5. References

8.5.1. GE SIL No. 406, February 24, 1984, "Incore Instrumentation Protection"
8.5.2. GE SIL No. 409, dated June 19, 1984, "Incore Dry Tube Cracks"
8.5.3. GE SIL No. 406, Revision 1, July 2, 1984, "Incore Instrumentation Protection"
8.5.4. Memorandum J. C. Brooks to B. R. Buteau, dated August 2, 1984, "Review of SIL 409 – Incore Dry Tube Cracks"
8.5.5. Memorandum B. R. Buteau to R. J. Wanczyk, dated August 4, 1984, "Review of SIL 409"
8.5.6. GE SIL No. 409, Revision 1, dated July 31, 1986, "Incore Dry Tube Cracks"
8.5.7. Memorandum D. E. LaBayer to D. A. Reid, dated August 15, 1986, "Incore Instrument Protection – SIL 406"
8.5.8. Memorandum J. C. Brooks to B. R. Buteau, dated September 9, 1986, "Review and Recommendation on SIL 409, Rev. 1"
8.5.9. GE RICSIL No. 073, dated May 12, 1995, "Cracking in Incore Dry Tube"
8.5.10. Memorandum T. G. Stetson to Outage 18 File, July 25, 1995, "Outage 18 Dry Tube Replacement"
8.5.11. Memorandum T. G. Stetson to R. E. McCullough, August 7, 1995, "Response to Commitment RICSIL073"
8.5.12. Memorandum E. J. Taintor to D. C. Girroir, dated October 20, 1995, "Inservice Inspection of Vessel Internal Items Located Below the Core Support Plate"
8.5.13. BWRVIP-47, dated December 1997, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
8.5.14. Letter USNRC to BWRVIP, dated October 13, 1999, "Final Safety Evaluation of 'BWRVIP, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47),' EPRI Report TR-108727, (TAC No. MA1102)"
8.5.15. Letter NRC to BWRVIP, dated December 7, 2000, "Acceptance for Referencing of BWRVIP, BWR Lower Plenum Inspection and Evaluation Guidelines (BWRVIP-47) for Compliance with the License Renewal Rule (10 CFR Part 54)"
8.5.16. GE SIL No. 409, Revision 2, dated February 8, 2002, "Incore Dry Tube Cracks"
8.5.17. Action Item / Regulatory Commitment SIL-409R2_01, dated February 8, 2002, "Incore Dry Tube Cracks"
8.5.18. Telex Warren Phelan (GE Reuter Stokes) to Carl Larsen, dated March 26, 2002, 1986 Dry Tube Replacement Design
8.5.19. Action Item / Regulatory Commitment SIL-409R2_02, dated April 1, 2002, "Revise PP 7027 to change the inspection frequency"

9.0 Instrument Penetrations

9.1. BWRVIP Document Applicability

BWRVIP-49, published in March 1998, governs inspection of the instrument penetrations. Section 3.2 of BWRVIP-49 states that no additional inspections (beyond the required ASME Section XI inspections) are recommended for any of these locations. Therefore, Vermont Yankee has complied with this document as of its publication.

9.2. ASME Section XI Applicability

ASME Section XI, Code Category B-P, Item B15.10, requires that a VT-2 be performed of the instrument penetrations each refueling outage. This requirement is addressed in PP 7034, the Inservice Inspection Pressure Test Program procedure. ASME Section XI, Code Category B-F, Items B5.20 and B5.30 require that a surface examination be performed of the nozzle-to-safe-end weld each ten-year interval. This requirement is addressed in PP 7015, the Inservice Inspection Program procedure. (Relief Request RI-01 does not include this scope.)

9.3. Other Commitments - None

9.4. Inspection for Risk to Generation Purposes - None

9.5. References

9.5.1. BWRVIP-49, dated March 1998, "BWRVIP Instrument Penetration Inspection and Flaw Evaluation Guidelines

9.5.2. Letter NRC to BWRVIP, dated August 4, 1998, "Safety Evaluation of 'BWRVIP Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49)'"

9.5.3. BWRVIP-57, dated December 1998, "BWRVIP Instrument Penetration Repair Design Criteria"

9.5.4. Letter NRC to BWRVIP, dated September 1, 1999, "Acceptance for Referencing of BWRVIP, 'BWRVIP Instrument Penetration Inspection and Flaw Evaluation Guidelines (BWRVIP-49),' for Compliance with the License Renewal Rule (10 CFR Part 54)"

9.5.5. BWRVIP-49-A, dated March 2002, "BWRVIP Instrument Penetration Inspection and Flaw Evaluation Guidelines

9.5.6. Action Item / Regulatory Commitment BWRVIP-049-A_01, dated June 6, 2002, "Evaluate BWRVIP-49-A and define solutions as required"

10.0 Jet Pumps

10.1. BWRVIP Document Applicability

BWRVIP-41, published in October 1997, governs inspection of the jet pumps. Vermont Yankee has complied with this document as of its publication, with the exception noted below for the diffuser/adapter circumferential welds below the diffuser shell. Those welds were, however, examined within two cycles of the publication of BWRVIP-41 in accordance with guidelines later published in BWRVIP-94. The inspection requirements for all of the jet pump subcomponents listed below are established in BWRVIP-41, Table 3.3-1. The document establishes six-year inspection intervals for specific inspections described below. Vermont Yankee defines the first six-year interval to include RFO 20 (1998), RFO 21 (1999), RFO 22 (2001), and RFO 23 (2002). The second six-year interval will include RFO 24 (2004), RFO 25 (2005), RFO 26 (2007), and RFO 27 (2008). The third six-year interval will begin with RFO 28 (2010) and RFO 29 (2011).

BWRVIP-48, published in February 1998, governs inspection of the jet pump riser brace attachment welds. Vermont Yankee has complied with this document as of its publication.

10.1.1. Beams

No inspection is required during the first ten years of service. After ten years of service, inspection of 50% of the beams is required every six years. After 20 years of service, inspection of 100% of the beams is required every six years. The beams were replaced in RFO 9, (November/December 1981). In RFO 20 (1998) all 20 beams were ultrasonically inspected for the first time. One beam bolt (no. 7) was replaced as a result of that inspection. It was determined (Reference 10.5.67) that it is unlikely that the UT indication, which instigated replacement of that beam, was from a service-related flaw. In RFO 21 (1999) beams 1 through 10 (50%) were inspected. In RFO 23 (2002) the beams were over 20 years old and 100% will require inspection in each six-year interval. All 20 beams were ultrasonically tested in RFO 23 (2002). In response to GE RICSIL No. 086 (Quad Cities beam failure) the beam transition regions were also inspected in RFO 23 (2002) by VT-1. The inspection frequency is determined from BWRVIP-41, Table 3.3-1.

BWRVIP-41 identifies the method of inspection in Table 3.3-1 to be UT or by other NDE technique. Currently, ultrasonic techniques are the only method of qualifying on the Inspection Committee mockups in accordance with BWRVIP-03.

Vermont Yankee, through RFO 20 (1998), visually inspected the jet pump beams in accordance with SIL 330, Supplement 2 and RICSIL 065 (References 10.5.26 and 10.5.28) per commitments in References 10.5.33 and 10.5.35 below in order to address GE RICSIL No. 065; GE SIL No. 330, Supplement 2; and NUREG/CR-3052. This required that the beams be visually inspected in one loop every refueling outage on an alternating basis. Because the ultrasonic method is much more capable of detecting flaws in the relevant areas of the beam bolt than the visual method, the BWRVIP methodology will be adopted. The aforementioned internal commitment is considered revised accordingly, with the issuance of this document.

During RFO 23 (2002), all beams were also visually inspected in the transition region to address RICSIL 086 and the beam failure at Quad Cities (see References 10.5.75 and 10.5.76).

10.1.2. Riser Thermal Sleeve Welds

These welds are currently inaccessible for inspection, and per BWRVIP-41, Table 3.3-1, "Inspection is recommended when the technique becomes available." Because a technique still does not exist, VY has complied with this document as of its publication. Inspection of 50% of these welds would be required within the first six-year interval and the other 50% within the six-year interval following that. After those first twelve years, inspection of 25% of these welds within each subsequent six-year interval would be required. This inspection frequency is determined from BWRVIP-41, Table 3.3-1. BWRVIP-41 is not clear when the six or twelve years begins for these hidden welds; additionally, the BWRVIP Assessment Committee is currently evaluating the necessity of performing these examinations.

BWRVIP-41, in Table 3.3-1, identifies the method of inspection for the riser thermal sleeve welds as modified VT-1. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method. However, visual inspection of these inaccessible welds will probably not be possible, and ultrasonic testing will most likely be required. BWRVIP-41, Paragraph 3.2.4, states that, "In all cases where a (visual) inspection is recommended...a suitable NDE examination technique meeting the requirements of BWRVIP-03 may be substituted."

A Technical Justification in accordance with PP 7027, Paragraph 4.2.3 and BWRVIP-94 is in the course of preparation to defer examination of these welds until such time that tooling and an NDE technique become available.

10.1.3. Riser Welds

An ultrasonic baseline inspection of 26 of these 30 welds (three per riser) was performed in RFO 20 (1998). The remaining four welds received a modified VT-1 (with cleaning performed) inspection. The ultrasonic inspection identified indications on four thermal sleeve-to-elbow welds (N2B-RS-1, N2C-RS-1, N2H-RS-1, and N2K-RS-1). Vermont Yankee received an SER from the NRC (Reference 10.5.61 below) to allow deferral of inspection for these four welds with UT indications until RFO 22 (2001). During RFO 22 these four riser welds were reinspected by UT with the result that two of the previous indications were found to be liftoff of the transducers, and therefore nonrelevant. The indications in the remaining two welds (N2H-RS-1 and N2K-RS-1) had not grown. One of the welds was inspected visually in the area of the UT indications and no cracking was seen. Technical Evaluation No. TE-2003-0021 (Reference 10.5.82) was prepared in order to allow these welds to be inspected by EVT-1 rather than by UT going forward. Per TE-2003-0021, these welds are to be inspected every two cycles. Welds N2H-RS-1 and N2K-RS-1 were inspected by EVT-1 during RFO 24 (2004) with no indications identified. If after three successive inspections with no recorded indications of cracks, TE-2003-0021 states that VY will revert to the six-year inspection interval specified in BWRVIP-41.

After a baseline inspection has been completed within the first six-year interval, inspection of 50% of the riser welds is required within each subsequent six-year interval. This inspection frequency is determined from BWRVIP-41, Table 3.3-1. The second 6 year baseline of 50% of the riser welds were completed in RFO 24 (2004).

BWRVIP-41, in Table 3.3-1, identifies the method of inspection for the riser welds as modified VT-1. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method. BWRVIP-41, Paragraph 3.2.4, states that, "In all cases where a (visual) inspection is recommended...a suitable NDE examination technique meeting the requirements of BWRVIP-03 may be substituted." Therefore, for these welds an EVT-1 or a UT technique is acceptable.

The above BWRVIP methodology exceeds the commitment in References 10.5.47 and 9.5.49 below, which was generated in order to address GE SIL No. 605. This would have required that the two elbow riser welds be visually inspected in one loop every refueling outage on an alternating basis. Because the scope has been expanded and the inspection methods have been upgraded, the BWRVIP methodology will be adopted. The aforementioned internal commitment is considered revised accordingly, with the issuance of this document.

10.1.4. Riser-to-Restrainer and Riser-to-Brace Welds

A modified VT-1 baseline inspection of 50% these welds was performed in the first six year interval during RFO 20 (1998). The EVT-1 baseline inspection of the other 50% required within the second six-year interval was performed in RFO 24 (2004). After those first twelve years ending with RFO 27 (2008), inspection of 25% of these welds within each subsequent six-year interval would be required. This inspection frequency is determined from BWRVIP-41, Table 3.3-1.

BWRVIP-41, in Table 3.3-1, identifies the method of inspection for the riser-to-restrainer welds as modified VT-1. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method. In the future, these welds will be examined with the EVT-1 method.

10.1.5. Riser Braces

A modified VT-1 baseline inspection was performed in the first six year interval on 50% of these welds during RFO 20 (1998). The EVT-1 baseline inspection of the other 50% required within the second six-year interval was performed in RFO 24 (2004). After those first twelve years ending with RFO 27 (2008), inspection of 25% of these welds within each subsequent six-year interval would be required. This inspection frequency is determined from BWRVIP-41, Table 3.3-1 and BWRVIP-48, Table 3-2.

BWRVIP-41, in Table 3.3-1 and BWRVIP-48, in Table 3-2 identify the method of inspection for the riser brace and vessel attachment welds as modified VT-1. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method. In the future, these welds will be examined with the EVT-1 method.

Vermont Yankee, through RFO 20 (1998), visually inspected the jet pump riser brace welds in accordance with References 10.5.19 and 10.5.31 below in order to address GE RICSIL No. 045 and GE SIL No. 551. This requires that the riser brace welds be inspected in one loop every refueling outage on an alternating basis. All jet pump riser brace welds have been inspected and no findings have been reported. This internal commitment is considered revised by the above BWRVIP inspection methodology with the issuance of this document.

10.1.6. Inlet Clamp Bolts

A VT-3 baseline inspection of 50% the inlet clamp bolted connections was performed in RFO 20 (1998). No degradation has ever been identified. A VT-3 50% baseline inspection on the balance of inlet clamp bolted connections was performed in RFO 24 (2004) for the second 6 year inspection interval. After those first twelve years, ending in RFO 27 (2008), inspection of 25% of the bolted connections within each subsequent six-year interval would be required. This inspection frequency is determined from BWRVIP-41, Table 3.3-1.

BWRVIP-41, in Table 3.3-1, identifies the method of inspection for the inlet clamp bolts as VT-3.

10.1.7. Restrainer Assemblies

A modified VT-1 baseline inspection of 50% of the restrainer wedges was performed in RFO 20 (1998). No movement or wear of the wedges has ever been identified. Per BWRVIP-41 – after a baseline inspection of 50% of the wedges is performed in the first six-year interval – inspection of the other 50% is required within the second six-year interval. After those first twelve years, inspection of 25% of the wedges within each subsequent six-year interval would be required. This inspection frequency is determined from BWRVIP-41, Table 3.3-1.

Representatives from Reactor Engineering and Plant Engineering met on January 13, 1999, and decided to increase this inspection frequency in order to be conservative and to also address the risk to generation consequences of restrainer failure. Vermont Yankee intends to perform inspection of the restrainer wedges in one loop every other outage. Therefore, only every other inspection performed on the restrainer wedges would be mandatory.

BWRVIP-41, in Table 3.3-1, identifies the method of inspection for the restrainer wedges as VT-1. If movement or wear of the wedge bearing surface is detected, inspection of the other restrainer components and locations, such as the bracket welds and the adjusting set screws, is required to assess the cause of movement.

Vermont Yankee, through RFO 20 (1998), visually inspected the jet pump adjusting screws (sometimes called the set screw or restrainer stop) in accordance with References 10.5.38, 10.5.43, and 10.5.51 below in order to address GE SIL No. 574 and GE RICSIL No. 078. This requires that setscrew gaps and the two tack welds on each of the two setscrews per jet pump be inspected. One loop has been performed each refueling outage on an alternating basis. All setscrews have been inspected and no findings were reported.

Representatives from Reactor Engineering and Plant Engineering met on January 13, 1999, and agreed that even though the set screws were non-safety related, that in order to address the economic consequences of restrainer failure, the setscrew inspections should continue to be performed, but on a reduced frequency. Vermont Yankee will perform inspection of the setscrews in one loop every other outage. These inspections will be VT-3s. The above internal commitment is considered revised by the above BWRVIP inspection methodology with the issuance of this document. This commitment is only an internal commitment and could be changed or deleted in the future. Therefore, the inspections are considered nonmandatory. BWRVIP 41-A criteria for inspecting jet pump wedge assembly set-screw has been revised by letter (Reference 10.5.87) to require set-screw inspection only after wedge assembly surface wear has been identified. Therefore, the above outlined VT-3, non-mandatory, set-screw inspection will not be performed unless wedge assembly surface wear is identified.

BWRVIP 41-A has been issued to the Executive Committee, which changed restrainer bracket assembly inspection recommendations. The purpose of inspecting the restrainer bracket assembly is to detect wear. The causes for wedge wear are related, but not limited to, increase in jet pump drive flow and/or core flow, set screw gaps and slip joint differential pressure which can increase vibration loads. If wear is detected, inspection of the other restrainer components/location such as bracket weld locations, the adjusting screw, wedge rod, not weld, etc., as applicable, should be performed during the same outage when wedge wear was detected to assess the cause of wear.

The baseline inspection of the wedge and bearing surfaces is required over the next two outages with 50% being inspected in the next refueling outage (Reference 10.5.87). The re-inspection is 25% each inspection cycle. If wedge wear is detected, then no wedge re-inspection shall exceed 6 years.

Since RFO 20 (1998) VY has visually examined 50% of the jet pump wedges every other outage. During RFO 22 (2001) and RFO 24 (2004) 100% of the jet pump wedge assemblies were examined by VT-1 with no wear indicated. Inspections over the next inspection cycle of six (6) years, RFO 25 (2005), RFO 26 (2007), RFO 27 (2008), and RFO 28 (2010) five (5) jet pump wedge assemblies will be inspected and then 25% more over the next 6 years and so forth.

This inspection strategy satisfies the 100% baseline examination requirement using the prescribed inspection method. The power up-rate has been considered in this inspection strategy. VY has not commenced with up-rated power conditions as of the publication date 11/4/04, but is scheduled to commence during cycle 24 (2005). However core flow is only increasing a small amount (Reference Email 10.5.88) and remains within the original licensed limit, therefore increased jet pump vibration is not anticipated (Reference 10.5.89). VY complies with the requirements of BWRVIP 41 as amended by the 2004 letter (Reference 10.5.87). This inspection strategy supercedes the internal commitment outlined above for inspecting 50% of the jet pump wedge assemblies every other outage.

GE SIL 629 also addresses jet pump restrainer wedges and set screws. This SIL has no impact on the conduct of jet pump restrainer inspections at Vermont Yankee, as outlined above and in Reference 10.5.71.

10.1.8. Mixer Inlet

GE SIL No. 465, Supplement 1, recommended visual inspection of the mixer inlets for detection of deleterious crud deposits. Reference 10.5.44 below provides Vermont Yankee's latest response to this GE SIL. It recommends that Vermont Yankee continue to monitor jet pump performance via the Reactor Engineering Jet Pump Performance Monitoring Program, which trends various critical parameters important for tracking jet pump efficiency. This reference also recommends making no plans to perform additional jet pump internal visual inspections, unless it is deemed necessary from indications of degraded performance from the trended data. It may, however, be advisable in the future to perform this inspection (and/or to perform mixer inlet cleaning) if jet pump performance drops below a critical level.

10.1.9. Mixer/Diffuser Circumferential Welds above Diffuser Shell

A baseline inspection of a minimum of 50% of these welds was required within the current six-year interval, which began December 1997 when BWRVIP-41 was first published. Because 100% were examined in RFO 21 (1999), these welds do not require reinspection until the next 12-year interval. After the first twelve-year interval, inspection of 25% of these welds within each subsequent six-year interval would be required. This inspection frequency is determined from BWRVIP-41, Table 3.3-1.

BWRVIP-41, in Table 3.3-1, identifies the method of inspection for the mixer and diffuser circumferential welds as modified VT-1. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method. BWRVIP-41, Paragraph 3.2.4, states that, "In all cases where a (visual) inspection is recommended...a suitable NDE examination technique meeting the requirements of BWRVIP-03 may be substituted." Therefore, for these welds an EVT-1 or a UT technique is acceptable.

10.1.10. Diffuser/Adapter Circumferential Welds below Diffuser Shell

A baseline inspection of 50% of these welds was required at the next refueling outage following publication of BWRVIP-41. For Vermont Yankee this would have been during RFO 20 (1998). Baseline inspection of the other 50% of these welds was required within the first six-year interval. Instead of the above guidance, Vermont Yankee elected to perform 100% of these welds in RFO 21 (1999) using a UT technique. Therefore, Vermont Yankee did not comply with BWRVIP-41 as of its publication for these particular welds. However, those welds were examined within two cycles of the publication of BWRVIP-41 in accordance with guidelines later published in BWRVIP-94. After a baseline inspection has been completed within the first six-year interval, inspection of 50% of these welds is required within each subsequent six-year interval. This inspection frequency is determined from BWRVIP-41, Table 3.3-1.

The RFO 21 (1999) ultrasonic inspection identified indications in four diffuser welds (2-DF-2, 3-DF-3, 9-DF-2, and 10-DF-2). Vermont Yankee performed an analysis (Reference 10.5.69 below) to allow deferral of inspection for the most limiting of these four welds until RFO 23 (2002). The RFO 23 (2002) UT measured flaw lengths were the same as found in RFO 21 (1999) within the documented NDE accuracy. These welds were inspected visually from the ID of the jet pump and no cracking was seen (one weld was also inspected on the OD). Technical Evaluation No. TE-2003-0021 (Reference 10.5.82) was prepared in order to allow these welds to be inspected by EVT-1 rather than by UT going forward. Per TE-2003-0021, these welds are to be inspected every two cycles. If after three successive inspections with no recorded indications of cracks, TE-2003-0021 states that VY will revert to the six-year inspection interval specified in BWRVIP-41.

BWRVIP-41, in Table 3.3-1, identifies the method of inspection for the diffuser and adapter circumferential welds as modified VT-1. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method. BWRVIP-41, Paragraph 3.2.4, states that, "In all cases where a (visual) inspection is recommended...a suitable NDE examination technique meeting the requirements of BWRVIP-03 may be substituted." Therefore, for these welds an EVT-1 or a UT technique is acceptable.

10.1.11. Sensing Lines

Representatives from Reactor Engineering and Plant Engineering met on January 13, 1999, and agreed to inspect jet pump sensing lines and their brackets in order to address the economic consequences of sensing line failure. Vermont Yankee intends to perform these nonmandatory inspections of the sensing lines in one loop every other outage. These inspections will be VT-3s.

10.2. ASME Section XI Applicability

Inspection of the jet pump riser braces is also governed by ASME Section XI, Table IWB-2500-1, Category B-N-2, Item No. B13.20, "Interior Attachments Within Beltline," which requires a VT-1 inspection once each ten-year interval, typically performed at the end of the interval. The inspection method given above by the BWRVIP requirements (EVT-1) exceeds the ASME Section XI requirements (VT-1). However, the inspection frequency would be less conservative – 100% in the first twelve-year BWRVIP interval instead of 100% in the Section XI ten-year interval – and 50% in subsequent twelve-year BWRVIP intervals thereafter. VY has submitted a Relief Request (RI-01) for the fourth ten-year Section XI interval that would allow using the BWRVIP guidance rather than the Section XI Categories B-N-1 and B-N-2 requirements. VY will perform inspections accordingly, based on the outcome of the Relief Request.

10.3. Other Commitments

This Program supersedes various internal commitments. They are discussed above with regard to the jet pump beams, riser circumferential welds, riser brace welds, and the restrainer setscrews.

10.4. Inspections for Risk to Generation Purposes

There are two jet pump components that are intended to be inspected solely for risk to generation purposes. These are the restrainer set screws and the sensing lines. Current BWRVIP guidance (Reference 10.5.87) no longer requires set screw inspections, instead wedge surface inspections are performed, resultant wear is a good indication of vibration which would require set screw inspections. Therefore set screw inspection will not be performed unless surface wear is detected. There is also one case noted above for the mixer inlets where inspections may be indicated, based on operational performance.

10.5. References

10.5.1. NRC IE Bulletin 80-07, dated April 4, 1980, "BWR Jet Pump Assembly Failure"
10.5.2. Memorandum L. H. Heider to B. H. Grier, dated May 8, 1980, "Response to IE Bulletin; BWR Jet Pump Assembly Failure"
10.5.3. NRC IE Bulletin 80-07, Supplement 1, dated May 13, 1980, "BWR Jet Pump Assembly Failure"
10.5.4. GE SIL No. 330, dated June 9, 1980, "Jet Pump Beam Cracks"
10.5.5. GE SIL No. 330, Supplement 1, dated February 1981, "Jet Pump Beam Cracks"

10.5.6. Memorandum B. R. Buteau to R. E. Kenney, dated May 27, 1981, "Vermont Yankee Implementation of SIL No. 330 Regarding Jet Pump Beam Cracking"

10.5.7. Letter R. F. Thibault (GE) to W. P Murphy, dated May 14, 1982, "Jet Pump Beam Test Results"

10.5.8. Memorandum B. R. Buteau to VY PORC, dated September 28, 1982, "Closeout of PORC Follow Item 81-55-01, Evaluation of GE Jet Pump Beam Test Results and Recommended Surveillance Program"

10.5.9. NUREG/CR-3052, dated November 1984, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure"

10.5.10. GE SIL No. 420, dated March 28, 1985, "Inspection of Jet Pump Sensing Lines"

10.5.11. Memorandum G. A. "Englesson to L. P. Lopriore, dated September 20, 1985, "Inspection of Jet Pump Sensing Lines SIL 420"

10.5.12. Memorandum G. A. "Englesson to D. D. Bauer, dated September 23, 1985, "Response to MD Commitment Item SIL 420"

10.5.13. Memorandum B. R. Buteau to R. J. Wanczyk, dated October 18, 1985, "Response to Commitment Item Regarding Review of GE SIL 420"

10.5.14. GE SIL No. 465, dated May 17, 1988, "Jet Pump Mixer Unusual Surface Observations"

10.5.15. Memorandum J. C. Brooks to B. R. Buteau, dated August 16, 1988, "Review of SIL0465"

10.5.16. Memorandum J. C. Brooks to B. R. Buteau, dated March 14, 1989, "Inspection of One Loop of Jet Pump Inlet Mixers"

10.5.17. GE RICSIL No. 045, dated October 16, 1989, "Jet Pump Riser Brace Crack"

10.5.18. Memorandum C. B. Cameron to R. E. McCullough, dated October 20, 1989, "Preliminary Review to RICSIL045"

10.5.19. Memorandum J. C. Brooks to B. R. Buteau, dated November 2, 1989, "Review of RICSIL No. 045, Jet Pump Riser Brace Crack"

10.5.20. Memorandum R. A. Current to T. A. Watson, dated September 2, 1992, "Recirc "A" Flow Anomalies"

10.5.21. Memorandum T. G. Stetson to R. E. McCullough, dated February 11, 1993, "Response to Commitment RCE9214 On Inspecting Jet Pump Throat Area"

10.5.22. GE SIL No. 551, dated February 26, 1993, "Jet Pump Riser Brace Cracking"

10.5.23. GE SIL No. 465, Supplement 1, dated April 30, 1993, "Surface Observations on Jet Pump Mixers"

10.5.24. GE SIL No. 573, dated October 5, 1993, "Jet Pump Nozzle Plug Modification"

10.5.25. GE SIL No. 574, dated October 5, 1993, "Jet Pump Adjusting Screw Tack Weld Failures"

10.5.26. GE SIL No. 330, Supplement 2, dated October 27, 1993, "Jet Pump Beam Cracks"

10.5.27. INPO Significant Event Notification, dated November 9, 1993, "Unanticipated Reactor Recirculation Jet Pump Failure During Power Operation"

10.5.28. GE RICSIL No. 065, dated December 3, 1993, "Jet Pump Beam Cracking"

10.5.29. NRC Information Notice 93-01, dated December 17, 1993, "Jet Pump Hold-down Beam Failure"

10.5.30. Operating Experience Review Report, dated December 27, 1993, "Jet Pump Hold Down Beam Failure"

10.5.31. Memorandum T. G. Stetson to R. E. McCullough, dated January 27, 1994, "Response to SIL0551, Jet Pump Riser Brace Cracking"
10.5.32. Memorandum T. G. Stetson to D. C. Porter, dated January 27, 1994, "Service Request – Effect of Increased Recirc Pump Speed on Reactor Internals"
10.5.33. Memorandum T. G. Stetson to R. E. McCullough, dated February 18, 1994, "Response to Commitment RICSIL065, Jet Pump Beam Cracking"
10.5.34. Memorandum T. G. Stetson to R. E. McCullough, dated February 18, 1994, "Response to Commitment SEN105 on Jet Pump Failure"
10.5.35. Memorandum T. G. Stetson to R. E. McCullough, dated February 18, 1994, "Response to Commitment SIL0330S2"
10.5.36. Letter M. O. Lenz (GE) to M. E. Shepherd (GE), dated March 4, 1994, "Vermont Yankee Jet Pump Beam-Bolt Assemblies"
10.5.37. Memorandum J. Cihak to M. P. Benoit, dated March 4, 1994, "Commitment SIL0573"
10.5.38. Memorandum T. G. Stetson to R. E. McCullough, dated April 19, 1994, "Response to Commitment SIL0574"
10.5.39. Memorandum T. G. Stetson to R. E. McCullough, dated May 9, 1994, "Response to Commitment SIL0465S1, Surface Observations on Jet Pump Mixers"
10.5.40. Memorandum T. G. Stetson to R. E. McCullough, dated May 19, 1994, "Response to Commitment UND94007, Review of NUREG/CR-3052 on Jet Pump Beams"
10.5.41. Memorandum T. G. Stetson to F. J. Helin, dated May 22, 1995, "Response to Commitment SIL0465S1RE1"
10.5.42. GE RICSIL No. 078, dated June 3, 1996, "Jet Pump Restrainer Bracket Set Screw Gaps"
10.5.43. Memorandum T. G. Stetson to R. E. McCullough, dated July 11, 1996, "Response to Commitment RICSIL078"
10.5.44. Memorandum T. G. Stetson to R. E. McCullough, dated October 25, 1996, "Response to Commitment SIL0465S1RE2"
10.5.45. GE SIL No. 605, dated December 6, 1996, "Jet Pump Riser Pipe Cracking"
10.5.46. NRC Information Notice 97-02, dated February 6, 1997, "Cracks Found In Jet Pump Riser Assembly Elbows at Boiling Water Reactors"
10.5.47. Memorandum T. G. Stetson to R. E. McCullough, dated February 7, 1997, "Response to Commitment SIL0605"
10.5.48. GE SIL No. 605, Revision 1, dated February 25, 1997, "Jet Pump Riser Pipe Cracking"
10.5.49. Memorandum T. G. Stetson to R. E. McCullough, dated April 9, 1997, "Response to Commitment SIL0605R1"
10.5.50. Memorandum T. G. Stetson to R. E. McCullough, dated April 9, 1997, "Response to Commitment INF97002"
10.5.51. Memorandum D. C. Girroir to P. B. Corbett, dated August 19, 1997, "AP 0028 CAT A Item # OE8428"
10.5.52. BWRVIP-41, dated October 1997, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"
10.5.53. GE Nuclear Energy Final Report No. 1HQXE, Rev. 0, dated April 1998, "Vermont Yankee Nuclear Plant Unit 1 Recirculation Inlet Riser Ultrasonic Examination"
10.5.54. BWRVIP-51, dated May 1998, "Jet Pump Repair Design Criteria"

10.5.55. Letter VYNPC to USNRC, dated May 4, 1998, BVY 98-67, "Jet Pump Riser Circumferential Weld Inspections"

10.5.56. Letter USNRC to VYNPC, dated June 3, 1998, NVY 98-77, "Request for Additional Information Regarding Jet Pump Riser Circumferential Weld Inspections at Vermont Yankee Nuclear Power Station (TAC No. MA 1681)"

10.5.57. GE-NE-B13-01935-02, Revision 1, dated July 1998, "Jet Pump Assembly Welds Flaw Evaluation Handbook for Vermont Yankee"

10.5.58. Letter VYNPC to USNRC, dated July 30, 1998, BVY 98-112, "Response To Request for Additional Information Regarding Jet Pump Riser Circumferential Weld Inspections"

10.5.59. Letter USNRC to VYNPC, dated October 26, 1998, NVY 98-153, "Jet Pump Riser Circumferential Weld Inspections at Vermont Yankee Nuclear Power Station (TAC No. MA 1681)" (includes original one-cycle SER)

10.5.60. Letter VYNPC to USNRC, dated March 29, 1999, BVY 99-43, "Jet Pump Riser Circumferential Weld Inspections and Flaw Evaluation"

10.5.61. Letter USNRC to VYNPC, dated April 29, 1999, NVY 99-46, "Jet Pump Riser Circumferential Weld Inspections at Vermont Yankee Nuclear Power Station (TAC No. MA5109)" (includes two-cycle SER)

10.5.62. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"

10.5.63. GE Nuclear Energy Report No. GE-NE-B13-01935, Revision 2, dated July 1999, "Jet Pump Assembly Welds Flaw Evaluation Handbook for Vermont Yankee"

10.5.64. M[illegible]randum John Hoffman to Tom Silko, dated July 30, 1999, "Jet Pump Riser [illegible]r Cycle 21 Operation"

[illegible] [U]SNRC to BWRVIP, dated September 29, 1999, "Final Safety Evaluation of [illegible] Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (BWRVIP-48)"

10.5.66. Memorandum J. M. Abdelghany to John Hoffman, dated October 22, 1999, "Vermont Ya[illegible]e Allowable Jet Pump Weld Leakage Rates for LOCA and Recirculation Pump [illegible]nance"

10.5.67. [illegible]t Yankee Report by David B. King, dated October 12, 1999, "Study to Dete[illegible]ne Cause for Rejection of Jet Pump Beams at Vermont Yankee and Monticello"

10.5.68. Memorandum C. B. Larsen to John Hoffman, dated November 15, 1999, "Application of Uncertainty to Jet Pump Diffuser UT Indications"

10.5.69. Memorandum John Hoffman to D. C. Girroir, dated November 26, 1999, "Jet Pump Assembly Inspection Discrepancy Report Evaluation"

10.5.70. GE SIL No. 629, dated July 11, 2000, "Inlet-mixer Wedge Damage In BWR Jet Pump Assemblies"

10.5.71. Action Item / Regulatory Commitment SIL-0629_00, initiated August 16, 2000, "Inlet-mixer Wedge Damage In BWR Jet Pump Assemblies"

10.5.72. Letter USNRC to BWRVIP, dated February 4, 2001, "Final Safety Evaluation of the BWRVIP, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)"

10.5.73. GE Report JXOAL, Revision 0, dated May 2001, "Vermont Yankee Recirculation Inlet Riser Ultrasonic Examination"

10.5.74. Memorandum C. B. Larsen to John Hoffman, dated May 9, 2001, Jet Pump NDE Uncertainty"

10.5.75. Technical Evaluation No. 2001-030, dated May 14, 2001, "Evaluation of Jet Pump Riser Flaws"

10.5.76. GE RICSIL No. 086, dated January 28, 2002, "Cracking in the Transition Region of a Jet Pump Beam"

10.5.77. Action Item / Regulatory Commitment RICSIL-086_01, dated January 29, 2002, "Cracking in the transition region of a jet pump beam"

10.5.78. Action Item / Regulatory Commitment BWRVIP-028-A_01, dated June 6, 2002, "Evaluate BWRVIP-28-A and define solutions as required"

10.5.79. Action Item / Regulatory Commitment BWRVIP-028-A_02, dated July 8, 2002, "Generate ERFIS Point IDs for JP M Ratio"

10.5.80. Action Item / Regulatory Commitment BWRVIP-028-A_03, dated July 8, 2002, "Revise DP 0455 Jet Pump M Ratio Startup Checklist"

10.5.81. Action Item / Regulatory Commitment BWRVIP-028-A_04, dated July 8, 2002, "Revise OP 4110 – Jet Pump M Ratio Required Action"

10.5.82. FRA-ANP Final UT Report, dated December 9, 2002, "In-Vessel Ultrasonic Examination Final Report"

10.5.83. Technical Evaluation No. TE-2003-0021, dated April 3, 2003, "Justification to Revert to EVT-1 Inspection of Jet Pump Circumferential Welds with UT Indications"

10.5.84. GE RICSIL No. 088, dated April 4, 2003, "Jet Pump Beam Records"

10.5.85. Letter Carl Terry (BWRVIP Executive Chairman) to all BWRVIP Committee members, dated May 2, 2003, "Recommended Actions for GENE RISIL 088"

10.5.86. Technical Justification 2004-02, dated 03/26/04, "Justification for Deferral of Inspection of Inaccessible Welds"

10.5.87. BWRVIP Letter 2004-047, R. Dyle/T. Mulford to the Assessment and Integration Committee Members, Re: Request for Review & Approval to Transmit Revised Jet Pump Wedge Inspection Guidance for BWRVIP 41-A to the Executive Committee, dated February 2, 2004.

10.5.88. Email from Robert Vita to J. Lafferty, Re: Flow Induced Vibration, dated October 14, 2004.

10.5.89. General Electric "Safety Analysis Report for VY Nuclear Power Station Constant Pressure Power Uprate," NEDC-33090P, Revision 0, dated September 2003.

11.0 Lower Plenum

11.1. Components in the lower plenum (areas below the core plate) are discussed in other Paragraphs of this Appendix, as referenced below:

- CRD housings, CRD housing caps, CRD stub tubes – Paragraph 1.0
- Core plate beam fillet welds, core plate rim hold-down bolts, core plate alignment hardware – Paragraph 2.0
- Core shroud support legs, core shroud support baffle plate underside – Paragraph 4.0
- Incore flux monitor housings, incore flux monitor guide tubes, incore flux monitor guide tube stabilizers – Paragraph 8.0
- SLC and core plate ΔP lines – Paragraph 15.0

11.2. In addition, for risk to generation purposes, if access is gained to the lower plenum, the vessel bottom head and the bottom head drain should be inspected for debris or crud buildup. Debris and crud are foreign material concerns for the fuel cladding and the bottom head drain line.

12.0 Miscellaneous Vessel Internal Attachments (Including Steam Dryer, Specimen Holder, Guide Rod)

12.1. BWRVIP Document Applicability

Inspection requirements for the core spray, feedwater sparger, and jet pump vessel attachments are found in other sections of this document. This section will address inspection requirements for all other vessel internal bracket attachments. BWRVIP-48, published in February 1998, governs inspection of the reactor vessel internal attachment welds. Vermont Yankee has complied with this document as of its publication.

However, BWRVIP-48, Table 3-2, states that, "No additional inspections (for the steam dryer support and hold-down, guide rod, and surveillance specimen holder attachment welds) are required above those specified in a plant's ASME Section XI program." The inspection frequency is per ASME Section XI, Table IWB-2500-1, Category B-N-2 (once in each ten-year interval). One exception is listed in BWRVIP-48, Table 3-2, which requires that steam dryer support attachment welds that use furnace-sensitized stainless steel or Alloy 182 material be examined by the modified VT-1 method. The reactor vessel was heat treated subsequent to welding of these attachment pads. There is no evidence at this time that the steam dryer support attachment welds were not furnace-sensitized. Per References 5.5.19 and 5.5.22, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method. Therefore the steam dryer support attachment welds were ~~will be~~ inspected with the EVT-1 method in RFO 23 (2002).

12.2. ASME Section XI Applicability

The vessel internal attachment welds are examined in accordance with ASME Section XI, Table IWB-2500-1, Category B-N-2, Item Nos. B13.20 and B13.30. Table IWB-2500-1, Item No. B13.20, "Interior Attachments within Beltline Region" requires accessible welds to be visually inspected by the VT-1 method once per ten-year interval. The only interior attachment welds within the beltline region are the jet pump riser brace attachment welds and the lower surveillance specimen bracket attachment welds. (Inspection of jet pump riser brace attachment welds is specified in Appendix J, "Jet Pumps".) Table IWB-2500-1, Item No. B13.20, "Interior Attachments within Beltline Region" requires accessible welds to be visually inspected by the VT-1 method once per ten-year interval. Table IWB-2500-1, Item No. B13.30, "Interior Attachments Beyond Beltline Region", requires accessible welds to be visually inspected by the VT-3 method once per ten-year interval. Therefore, the surveillance specimen holder bracket attachment welds will be inspected with the VT-1 method (the upper specimen holder bracket will be upgraded from a VT-3 method to a VT-1 method), and the steam dryer hold-down brackets and guide rod brackets will be inspected with the VT-3 method. The steam dryer support brackets will be upgraded to EVT-1 as per the above BWRVIP direction. (Inspection of core spray piping bracket attachment welds are specified in the "Core Spray Internal Piping and Spargers" Section. Inspection of feedwater sparger bracket attachment welds is specified in the "Feedwater Spargers" Section.) All bracket attachment weld examinations for the third Section XI ten-year interval were conducted during RFO 23 (2002). VY has submitted a Relief Request (RI-01) for the fourth ten-year Section XI interval that would allow using the BWRVIP guidance rather than the Section XI Categories B-N-1 and B-N-2 requirements. VY will perform inspections accordingly, based on the outcome of the Relief Request. However, since the BWRVIP references ASME Section XI for the brackets, these inspections would be performed exactly as stated above.

12.3. Other Commitments

In RFO 16 (1992), cracking in cladding in the vessel head and shell interior was discovered at Vermont Yankee. The inspection was initiated in response to GE RICSIL No. 050 and GE SIL No. 539. One of these cracks was adjacent to the dryer support bracket at 215 degrees. It was determined through ultrasonic manual sizing from the outside of the reactor vessel at this location that the crack did not propagate into the vessel base-material pressure boundary. BWRVIP-48, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines," requires in Table 3-2, Note (1), that for indications that are detected visually, ultrasonic inspections should be performed to determine if the indication has propagated into the reactor vessel base material. Paragraph 3.2.1 states, "For any flaws which are found to have propagated into the base material, an evaluation should be performed in accordance with the requirements of ASME Section XI, Paragraph IWB-3600." Vermont Yankee's commitment to the NRC in References 12.5.2 and 12.5.9 below follows this logic.

Even though the flaw did not propagate into the reactor pressure vessel boundary, Vermont Yankee committed to the NRC (References 12.5.2 and 12.5.9 below) to perform successive examinations similar to ASME Section XI, IWB-2420(b), of this clad crack. In this way, this clad crack would be treated as if it were indeed a defect that exceeded the ASME Section XI acceptance criteria, even though it did not. Paragraph IWB-2420(b) requires that areas containing flaw indications that have been accepted analytically be reexamined during the next three inspection periods.

RFO 16 (1992) fell in the third period of the second interval. The dryer support bracket flaw was visually and ultrasonically reexamined in RFO 17 (1993), which fell in the first period of the third interval. In RFO 20 (1998), this examination was repeated, which satisfied the second successive reexamination (second period, third interval) for this flaw. During RFO 22 (2001), the third successive re-examination was completed (third period of the third interval).

12.4. <u>Inspections for Risk to Generation Purposes</u> – None.

12.5. <u>References</u>

12.5.1. Letter Arthur Shepard to J. J. Cihi (GE), dated September 22, 1970, "Overlay of RPV Internals Bracket and Pad Areas in Accordance with GE FDI #78"

12.5.2. Letter Vermont Yankee to USNRC, BVY 92-055, dated April 5, 1992, "Proposed Alternative for Compliance with 10CFR50.55a Regarding RPV Cladding Indications"

12.5.3. Letter Vermont Yankee to USNRC, dated April 10, 1992, "Supplemental Information Regarding Proposed Alternative for Compliance with 10CFR50.55a Regarding RPV Cladding Indications"

12.5.4. Letter USNRC to Vermont Yankee, dated April 17, 1992, "Meeting Summary of April 8, 1992 Meeting To Discuss Vermont Yankee Reactor Vessel Flaws"

12.5.5. Letter USNRC to Vermont Yankee, dated April 17, 1992, "Disposition of Reactor Vessel Cladding Indications Discovered During the March 1992 Refueling Outage At Vermont Yankee Nuclear Power Station"

12.5.6. Memorandum F. J. Helin to PORC, dated April 17, 1992, "Clad Indications Found During 1992 Refueling Outage"

12.5.7. EPRI TR-101971, dated February 1993, "Evaluation of Reactor Pressure Vessel Head Cracking in Two Domestic BWRs"

12.5.8. Letter Vermont Yankee to USNRC, dated July 1, 1993, "1993 Refueling Outage Vessel Clad Inspection Plans"

12.5.9. Letter Vermont Yankee to USNRC, dated October 6, 1993, "Reactor Vessel Clad Inspection during the 1993 Refueling Outage"

12.5.10. BWRVIP-48, dated February 1998, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"

12.5.11. BWRVIP-52, dated June 1998, "Shroud Support and Vessel Bracket Repair Design Criteria"

12.5.12. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"

12.5.13. Letter USNRC to BWRVIP, dated September 29, 1999, "Final Safety Evaluation of 'Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines, (BWRVIP-48)'"

12.5.14. Action Item / Regulatory Commitment BWRVIP-048-A_01, dated August 5, 2002, "Evaluate BWRVIP-48-A: Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"

13.0 Orificed Fuel Support Castings

13.1. BWRVIP Document Applicability

BWRVIP-47, published in December 1997, governs inspection of the orificed fuel support castings. However, BWRVIP-47 does not establish any inspection requirements for the orificed fuel support. Therefore, Vermont Yankee has complied with this document as of its publication.

13.2. ASME Section XI Applicability

The orificed fuel support castings are part of the core support structure; however, they are not integrally welded as stated in the title of ASME Section XI, Table IWB-2500-1, Category B-N-2. Therefore the orificed fuel support castings are not subject to ASME Section XI. See Reference 13.5.2 below.

13.3. Other Commitments – None.

13.4. Inspections for Risk to Generation Purposes – None.

13.5. References

13.5.1. BWRVIP-47, dated December 1997, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"

13.5.2. Memorandum C. B. Larsen to D. C. Girroir, dated May 13, 1999, "Definition of Core Support Structures (ASME Section XI, Category B-N-2)"

13.5.3. Letter USNRC to BWRVIP, dated October 13, 1999, "Final Safety Evaluation of 'BWRVIP, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines (BWRVIP-47),' EPRI Report TR-108727, (TAC No. MA1102)"

13.5.4. Letter NRC to BWRVIP, dated December 7, 2000, "Acceptance for Referencing of BWRVIP, BWR Lower Plenum Inspection and Evaluation Guidelines (BWRVIP-47) for Compliance with the License Renewal Rule (10 CFR Part 54)"

14.0 Specimen Holders

14.1. BWRVIP Document Applicability

No BWRVIP Inspection and Evaluation document addresses the specimen holders, which are considered non-safety related components, with the exception that BWRVIP-48, published in February 1998, which governs inspection of the reactor vessel internal attachment welds. Vermont Yankee has complied with this document as of its publication. The requirements for the specimen holder attachment welds are found in the Miscellaneous Vessel Internal Attachments section. Per BWRVIP-102, Vermont Yankee is obligated to inform the BWRVIP if it intends to withdraw any of the surveillance specimen coupons twelve months prior to their planned withdrawal.

14.2. ASME Section XI Applicability

Inspection of the specimen holder attachment welds is also governed by ASME Section XI, Table IWB-2500-1, Category B-N-2, Item No. B13.20, "Interior Attachments within Beltline Region." These requirements are also found in the Miscellaneous Vessel Internal Attachments section.

14.3. Other Commitments – None.

14.4. Inspections for Risk to Generation Purposes – Vermont Yankee has determined that inspection of the surveillance specimen holders should be performed for loose part issues and to assure that these assets are preserved. This nonmandatory inspection would coincide with the ten-year interval bracket inspection. Such an inspection was performed in RFO 23 (2003).

14.5. References

14.5.1. BWRVIP-86, dated December 2000, "BWR Integrated Surveillance Program Implementation Plan

14.5.2. BWRVIP-102, dated June 2002, "BWR Integrated Surveillance Program Implementation Guidelines

15.0 Standby Liquid Control/Core Plate Delta Pressure

15.1. BWRVIP Document Applicability

BWRVIP-27, published in October 1997, governs inspection of the SLC and core plate ΔP system. BWRVIP-27-A was issued August 2003. Vermont Yankee has complied with this document as of its publication.

BWRVIP-27-A asserts that the only safety critical welds in the SLC/Core Plate ΔP system within the scope of the BWRVIP are the welds outside the reactor vessel which connect the SLC system piping to the vessel. BWRVIP-27-A, Paragraph 2.1.5 and Figure 2-5 describe the Vermont Yankee configuration, which is a stainless steel safe-end welded to a carbon steel forged nozzle and fabricated by CB&I. VY Drawing 5920-358 shows this configuration and Drawing 5920-5266 shows the replacement safe-end of improved material installed shortly before initial start-up. The safe-end thickness on both drawings is 7/8". BWRVIP-27-A, Paragraphs 3.3.1 and 3.4.1 state the requirements for the Vermont Yankee configuration; it requires that the nozzle-to-safe end weld and the safe-end extension be examined volumetrically.

BWRVIP-03 through Revision 5 (December 2002), in Sections 11.4.2 and 11.4.3, contained two qualifications of UT techniques performed by EPRI that are applicable to the SLC safe-end. However, those two qualifications were performed on safe-ends that were ½" thick, and neither qualification applied to a safe-end that is 7/8" thick. Therefore, a volumetric examination technique had not been demonstrated for this configuration to that date.

BWRVIP-27, Paragraphs 3.3.1 and 3.4.1 also stated that, "until such time as a qualified volumetric examination is available, enhanced leakage inspection during each Category B-P pressure boundary leak test should be performed." An enhanced leakage test is defined as requiring a view of this joint specifically, rather than as would normally be required by ASME Section XI, which would be an examination for leakage in the general area. Per BWRVIP-27-A, insulation removal is required. This was not clarified until BWRVIP-27-A was issued as a draft in July 2002. Until that time the need for insulation removal was not explicitly stated (in BWRVIP-27) and VY did not do such in RFO 20 (1998), RFO 21 (1999), and RFO 22 (2001).

Per BWRVIP-27-A, Paragraphs 3.3.1 and 3.4.1, a surface examination performed every other refueling outage may be substituted for the enhanced leakage inspection. VY followed this option for the SLC nozzle-to-safe end weld and the safe end extension in RFO 23 (2002).

BWRVIP-03, Revision 6, Standard 2.6, Section 3.3, states that personnel performing analysis of dissimilar-metal weld UT data for the SLC system shall be qualified per ASME Section XI, Appendix VIII, Supplement 10. Personnel have qualified under the detection requirements. Technical Justification TJ-2004-05 (Reference 15.5.13) was prepared to allow continuation of surface examinations every other refueling outage because qualifications for sizing have not yet been determined.

BWRVIP-03, Standard 2.6, Section 3.3 states, "Personnel performing final analysis and review of examinations of dissimilar metal welds in the standby liquid control system shall have current qualification for crack detection, length sizing, and/or depth sizing, as appropriate, in accordance with ASME Code, Section XI, Appendix VIII, Supplement 10. During late 2002 and 2003 personnel began qualifying for examination of dissimilar metal welds in accordance with Appendix VIII, Supplement 10. Qualifications of those personnel covered the range of thicknesses and diameters of the SLC nozzle welds. Therefore, UT of these welds became mandatory for RFO 24. The only exception to this requirement is that the welds must be ground flush in accordance with the PDI dissimilar weld UT procedure. The BWRVIP Assessment Committee has provided an interpretation that if the SLC welds are not ground flush, then the plant is not obligated to either grind the welds *or* perform a UT – and may continue doing either EVT-2 or a PT. However, the recollection is that Vermont Yankee SLC nozzle welds are ground flush (this will be verified during RFO 24 (2004) and therefore, UT is mandatory. During RFO 24 (2004) the nozzle to safe-end weld was visually observed and determined inconclusive if UT could be performed. The weld was not profiled, instead a PT of the weld was performed with no relevant indications detected. A work tracking LO-VTYLO-2004-00541 has been issued to profile the weld and ultrasonically examine it during RFO 25.

Per BWRVIP-27-A, Paragraphs 3.3.1 and 3.4.1, the desired frequency for ultrasonic examination is once every 10 years. For scheduling purposes, the ultrasonic examinations performed per BWRVIP guidance should coincide with the surface examinations required by ASME Section XI below.

The BWRVIP stated in response to NRC SE Issue 3.2.2 (Reference 1.5.13) that when utilities have access to the lower plenum due to maintenance activities not related to the inspection recommendations of the BWRVIP, they will have the opportunity to perform a visual inspection of a portion of the lower plenum and that results of this inspection will be reported to the BWRVIP. This will be treated as a commitment for those items listed in 15.4 below.

15.2. <u>ASME Section XI Applicability</u>

Inspection of the SLC nozzle-to-safe-end weld is also governed by ASME Section XI, Table IWB-2500-1, Category B-F, Item No. B5.20, "Reactor Vessel Nozzle-To-Safe End Butt Welds Less than NPS 4." A surface examination is required once per ten-year interval. This weld and the requirements for its inspection are also included in the Vermont Yankee Inservice Inspection Program, PP 7015. (Relief Request RI-01 does not include this scope.)

15.3. <u>Other Commitments</u> – None.

15.4. Inspection for Risk to Generation Purposes

Representatives from Reactor Engineering and Plant Engineering met on January 13, 1999, and agreed that the SLC and core plate ΔP lines are non-safety related. In addition, inspection of these lines by themselves would be very costly and time consuming, because they are located below the core plate and core disassembly would be required. However, in order to address the economic consequences of failure, the group agreed to perform inspection of these components, but only if they were made accessible through other vessel activities. These are recommended inspections and are considered nonmandatory.

15.5. References

15.5.1. BWRVIP-27, dated April 1997, "BWR Standby Liquid Control System/Core Plate Delta P Inspection Criteria and Flaw Evaluation Guidelines"

15.5.2. BWRVIP-53, dated July 1998, "Standby Liquid Control Line Repair Design Criteria"

15.5.3. Letter USNRC to BWRVIP, dated April 27, 1999, "Safety Evaluation of the BWRVIP, 'BWR Standby Liquid Control System / Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-27),' EPRI Report TR-107236 (TAC No. M98708)"

15.5.4. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"

15.5.5. Memorandum M. P. Dugan to D. C. Girroir, dated June 29, 1999, "SLC Weld # N10-SE, Nozzle to Safe-end Examination"

15.5.6. Letter USNRC to BWRVIP, dated December 20, 1999, "Acceptance for Referencing of Report, 'BWRVIP, BWR Standby Liquid Control System / Core Plate Inspection and Flaw Evaluation Guidelines (BWRVIP-27),' for Compliance with the License Renewal Rule"

15.5.7. Memorandum C. B. Larsen to D. C. Girroir, dated June 29, 1999, "Status Of SLC Safe-End Examination Technique and Prognosis for Examination In RFO 23 (Rev. 1)"

15.5.8. Action Item / Regulatory Commitment UND-2002-074_07, dated March 21, 2002, "Standby liquid control nozzle-to-safe-end weld should be inspected in accordance with current industry guidance"

15.5.9. Action Item / Regulatory Commitment BWRVIP-027_01, dated June 12, 2002, "Evaluate BWRVIP-27 Revised Inspection Recommendations for SLC Penetrations"

15.5.10. Draft BWRVIP-27-A, dated July 2002, "BWR Standby Liquid Control System/Core Plate Delta P Inspection Criteria and Flaw Evaluation Guidelines"

15.5.11. Action Item / Regulatory Commitment BWRVIP-027-A_01, dated August 5, 2002, "Evaluate BWRVIP-27-A 'BWR SLC/Core Plate Delta P Inspection & Flaw Evaluations Guidelines"

15.5.12. BWRVIP-27-A, dated August 2003, "BWR Standby Liquid Control System/Core Plate Delta P Inspection Criteria and Flaw Evaluation Guidelines"

15.5.13. Technical Justification TJ-2004-05, dated March 26, 2004, "Justification for Deferral for UT of SLC Safe-end"

16.0 Steam Dryer

16.1. BWRVIP Document Applicability

No BWRVIP Inspection and Evaluation document addresses the steam dryer, which is considered a non-safety related component with the exception that BWRVIP-48, published in February 1998, governs inspection of the reactor vessel internal attachment welds. Vermont Yankee has complied with this document as of its publication. The requirements for the steam dryer support and hold-down attachment welds are found in the Miscellaneous Vessel Internal Attachments section of this Program.

16.2. ASME Section XI Applicability

Inspection of the steam dryer support and [illegible]ld-down attachment welds is also governed by ASME Section XI, Table IWB-2500-1, C[illegible]gory B-N-2, Item No. B13.30, "Interior Attachments Beyond Beltline Region." These require[illegible]nts are found in the Miscellaneous Vessel Internal Attachments section of this Program.

16.3. Other Commitments – GE SIL 644 (Refe[illegible]nce 16.5.7) was issued in August 2002 to address the Quad Cities steam dryer cover plate failu[illegible] following power uprate. VY inspected the dryer cover plate welds during RFO 23 (2002) [illegible] accordance with the SIL's recommendations because of the planned uprate following RFO 24. [illegible] addition, GE recommended looking at the steam dryer sta[illegible] [illegible]entation penetration [illegible]nd this examination was also performed in RFO 23 [illegible] [illegible]pplement 1 (Refe[illegible]nce 16.5.15) was issued in September 2003 to [illegible] [illegible]ities steam dryer [illegible]ailure following power uprate – this time in the outer [illegible] vertical hood pl[illegible]e. Extended Power Uprate Task T0305 [illegible] which addresses flo[illegible] induced vibration of the vessel internals, in Section 4.4.1 recommended modification of the st[illegible]am dryer hood vertical plates and the outer cover [illegible] [illegible]o inspections perform[illegible]d in accordance with SIL 644, Supplement 1. [illegible]ned these modifi[illegible]ations and inspections in RFO 24 (2004) in accordance [illegible]eference 6.5.18).

The modi[illegible]cations and repairs consisted [illegible]: 1) cutting out the existing ½" vertical and horizontal plates on each of the two Outer Hoods an[illegible] replacing them with 1" thick places; 2) removing the [illegible]r diagonal braces inside the Outer Ho[illegible]ls; 3) replacing the ¼" thick horizontal cover plates that are adjacent to the steam outlet nozz[illegible]s; 4) installing three gussets on the lower section of each Outer Hood; 5) removing the old T[illegible]-Bars and installing eight mitigation Tie-Bars with support gussets on the outer Tie-Bars; 6[illegible] repairing crack indications at weld location V-02-90 and V-02-270; 7) installing reinforceme[illegible] hardware in the areas behind the lifting lugs near the outer plenum vertical welds; and 8) addi[illegible]g new tack welds to the four leveling screws. Vermont Yankee committed to performing a detai[illegible]ed inspection of the steam dryer during the next refueling outage RFO 25 (2005) and duri[illegible]g the two subsequent refueling outages RFO 26 (2007) and RFO 27 (2008) in accordance with GE-SIL-644 Revision 1 as part of the EPU (Reference 16.5.19).

In parallel with the steam dryer modification activities GE began In Vessel Visual Inspection of the steam dryer in accordance with the recommendations of SIL 644 Supplement 1. The inspection included a VT-1 and VT-3 on the interior and exterior of the steam dryer according to GE Procedure GE-VT-203 Version 9. The results of these inspections are documented in Steam Dryer IVVI Final Report VYR24-04-MJ525 (Reference 16.5.20). One crack indication was found in welds OP-V19-180 and VO2-270. These welds are located at the 215 azimuth behind lifting lug "C."

16.4. Inspection for Risk to Generation Purposes

Representatives from Reactor Engineering and Plant Engineering met on January 13, 1999, and agreed that the steam dryer (other than the support and hold-down attachment welds) is non-safety related. However, the group agreed that some inspection may be warranted for risk to generation reasons. The inspection may be performed off of critical path in the equipment pool. Therefore, this nonmandatory inspection of the lifting lugs and associated hardware is intended to be performed every fourth refueling outage.

During RFO 20 (1998) several cracked tack welds on the steam dryer jacking bolts for the lifting eyes were discovered. These particular tack welds were reexamined in RFO 21 (1999). No changes from the previous examination were noted, so it was determined per Reference 16.5.6 that no further inspections of these tack welds are recommended until its next regularly scheduled inspection.

16.5. References

16.5.1. GE SIL No. 474, dated October 26, 1988, "Steam Dryer Drain Channel Cracking"

16.5.2. Memorandum C. B. Cameron to R. E. McCullough, dated December 8, 1988, "Preliminary Review to SIL-474

16.5.3. Memorandum R. P. Lopriore to W. L. Wittmer, dated June 19, 1989, "Commitment Item SIL474"

16.5.4. GE SIL No. 558, dated April 22, 1993, "Steam Dryer Damage Prevention"

16.5.5. Memorandum D. J. Rollins/M. Selling to M. P. Dugan, Revision 1, dated April 17, 1998, "Inservice Discrepancy Report 98-004, 98-005, 98-006 – Inservice Inspection of Tack Welds on Steam Dryer Lifting Lug Assemblies"

16.5.6. Memorandum D. J. Rollins/M. Selling to S. D. Goodwin, dated November 23, 1999, "Inservice Discrepancy Report 99-019 – Inservice Inspection of Steam Dryer"

16.5.7. GE SIL 644, dated August 21, 2002, "BWR/3 Steam Dryer Failure"

16.5.8. Action Item / Regulatory Commitment SIL-0644_01, dated August 26, 2002, "Evaluate GE SIL 644: BWR/3 Steam Dryer Failure"

16.5.9. NRC Information Notice 2002-26, dated September 11, 2002, "Failure of Steam Dryer Cover Plate after a Recent Power Uprate"

16.5.10. Action Item / Regulatory Commitment INF-2002-026_01, dated September 16, 2002, "Failure of Steam Dryer Cover Plate after a Recent Power Uprate"

16.5.11. OE16492 - PRELIMINARY REPORT, dated June 30, 2003, "Reactor Vessel Steam Dryer Structural Steel Bracing Degraded"

16.5.12. Action Item / Regulatory Commitment OE-16492_01, initiated July 8, 2003, "Reactor Vessel Steam Dryer Structural Steel Bracing Degraded"
16.5.13. NRC Information Notice 2002-26, Supplement 1, dated July 21, 2003, "Additional Failure of Steam Dryer After a Recent Power Uprate"
16.5.14. Action Item / Regulatory Commitment INF-2002-026 S1_01, dated August 6, 2003, "Additional Failure of Steam Dryer After a Recent Power Uprate"
16.5.15. GE SIL-644, Supplement 1, dated September 5, 2003, "BWR Steam Dryer Integrity"
16.5.16. Action Item / Regulatory Commitment SIL-0644S1_01, initiated September 8, 2003, "BWR Steam Dryer Integrity"
16.5.17. Extended Power Uprate Task T0305, GE-NE-0000-0016-4161-01, dated December 2003, "RPV Flow Induced Vibration"
16.5.18. VYDC 2003-12, dated January 2004, "Steam Dryer Strengthening"
16.5.19. Attachment 4,Vermont Yankee Nuclear Power Station Proposed Technical Specification Change No. 263 - Supplement No. 13 "Extended Power Uprate Response to Steam Dryer Action Items Commitments," BVY 04-097.
16.5.20. Vermont Yankee Nuclear Power Station Steam Dryer Modifications and Repairs, VY-RFO 24 Refueling Outage, April 2004, Report Number VYR24-04-MJ525.
16.5.21. GE-SIL-644 Revision 1, dated 11/04 "BWR Steam Dryer Failure"

17.0 Steam Separator/Shroud Head (Including Hold-down Bolts)

17.1. BWRVIP Document Applicability

No BWRVIP Inspection and Evaluation document addresses the steam separator, shroud head, or shroud head hold-down bolts. These are considered non-safety-related components.

17.2. ASME Section XI Applicability

There are no ASME Section XI inspection requirements that apply to the steam separator, shroud head, or shroud head hold-down bolts.

17.3. Other Commitments – None.

17.4. Inspection for Risk to Generation Purposes

Representatives from Reactor Engineering and Plant Engineering met on January 13, 1999, and agreed that the steam separator/shroud head is non-safety related. However, the group also agreed that some inspection may be warranted for risk to generation reasons. This inspection may be performed off of critical path in the equipment pool. Therefore, this nonmandatory inspection of the lifting lugs and associated hardware is intended to be performed every fourth refueling outage.

The shroud head hold-down bolts are considered non-safety related. These bolts were replaced as part of the shroud tie-rod repair in RFO 19 (1996). The replacement bolts were of a new design. There have been no materials problems associated with the new design of shroud head hold-down bolts and no inspections are recommended at this point.

17.5. References

17.5.1. GE Report MDE #293-1285, Revision 1, DRF #B11-00337, dated January, 1986, "Shroud, Shroud Head Sealing Surfaces, Alignment Pins and Guide Rod Evaluation"

17.5.2. GE SIL No. 433, dated February 7, 1986, "Shroud Head Bolt Cracks"

17.5.3. Memorandum R. P. Lopriore to R. E. McCullough, dated March 22, 1986, Response to MD Commitment Item SIL 433"

17.5.4. GE SIL No. 433, Supplement 1, dated September 15, 1993, "Shroud Head Bolt Failures"

17.5.5. Memorandum S. K Naeck to T. A. Watson, dated October 5, 1993, "GE SIL No. 433 Supplement 1 Shroud Head Bolt Failures"

17.5.6. Event Report ER#95-0267, dated April 1995, "Shroud Head Bolt Cracking"

17.5.7. Letter D. B. Drendel (GE) to B. R. Buteau, dated April 13, 1995, "Operation with Less Than the Full Complement of Shroud Head Bolts"

17.5.8. Memorandum B. R. Buteau to W. King, dated April 13, 1995, "Shroud Head Bolt Corrective Actions"

17.5.9. Memorandum B. R. Buteau to W. King, dated April 14, 1995, "Shroud Head Bolt Recommended Options"

17.5.10. Memorandum H. Ely to R. E. McCullough, dated April 15, 1995, "AP0028 Commitments"

17.5.11. Letter D. B. Drendel (GE) to B. R. Buteau, dated April 18, 1995, "Operation with Less Than the Full Complement of Shroud Head Bolts"

17.5.12. Memorandum T. A. Watson to R. E. McCullough, April 25, 1995, "Re: Shroud Head Bolt Failures, Perform Ultrasonic Testing of Shroud Head Bolts During 1995 Refueling Outage"

17.5.13. Memorandum T. R. Osterhoudt to PORC, dated April 26, "Operation with Degraded Shroud Head Bolts"

17.5.14. Memorandum J. T. Meyer to D. A. Reid, dated August 16, 1995, "Review of LERs, ERs"

17.5.15. Memorandum T. A. Watson to R. E. McCullough, dated September 25, 1995, "Shroud Head Bolts, UND95019_01"

17.5.16. Memorandum T. A. Watson to R. E. McCullough, dated May 10, 1996, "Shroud Head Bolts Unlatching, ER950267_02"

17.5.17. ER 20022538, dated October 10, 2002, "Two shroud head bolts had spring loaded keeper retainer nuts that would not spring up into place around the tensioning nut"

18.0 Top Guide

18.1. BWRVIP Document Applicability

BWRVIP-26, published in December 1996, governs inspection of the top guide. Vermont Yankee was not able to immediately comply with the inspection method as specified in BWRVIP-26 as of its publication. However, Vermont Yankee began examination in accordance with this document (with the exception of access, as described below) as of RFO 21 (1999) – within two cycles of the publication of BWRVIP-26 in accordance with guidelines later published in BWRVIP-94.

BWRVIP-26, Table 3-2, requires inspection of three components for BWR/4 plants without installed wedges: aligner pin assemblies, hold-down assemblies, and the top guide rim weld. The top guide rim weld does not exist at Vermont Yankee and is therefore exempt.

According to BWRVIP-26, Table 3-2, welds in two adjacent aligner pin assemblies are to be inspected every other refueling outage with the VT-1 method, unless a plant-specific analysis is performed to show that less than 20% of the weld is required. Prior to RFO 23 (2002), this analysis was performed and documented in VYC-2218 (Reference 18.5.39). Therefore, no inspection of the top guide aligner pin assemblies is required. Prior to RFO 23 (2002), a best effort VT-1 of the aligners was performed every other refueling outage. Such an examination was performed during RFO 19 (1996) on the aligner assemblies at 162 and 252 degrees, and again during RFO 21 (1999) on the aligner assemblies at 72 and 162 degrees. (If inspection of the top guide aligner assemblies ever becomes necessary again, there is sufficient weld length accessible for a VT-1 inspection in the aligner socket that is welded to the shroud ledge. On the other hand, the welds in the aligner socket that is welded into the top guide are not easily accessible for inspection; however, it could be argued that the two abutting aligner "Lego" blocks perform the same function as the aligner socket that is welded into the top guide. The aligner "Lego" blocks were verified to be in position with the VT-3 inspection method. Also, see Reference 18.5.32.)

According to BWRVIP-26, Table 3-2, a VT-1 inspection of two hold-down assemblies 180 degrees apart, where the hold-down latches to the shroud, are to be inspected every other refueling outage. Such an examination was performed in RFO 21 (1999) on the hold-down assemblies at 18° and 198° and in RFO 23 (2002) on the hold-down assemblies at 108° and 288°.

Vermont Yankee is not now planning to install top guide wedges. (There are no inspection requirements identified in Table 3-2 of BWRVIP-25 for BWR/4 plants with installed wedges.) If Vermont Yankee ever does install wedges they may require some periodic inspection. BWRVIP-50, Paragraph 10.2, states, "Inspections required for the entire repaired top guide/core plate structures for the remaining life of the unit, shall be specified commensurate with design considerations and code requirements applicable to the specific design. This shall include inspections of the repair hardware and inspection of the reactor internal components utilized for repair anchorage." These inspection requirements would be delivered as a piece of the wedge design scope. Barring any guidance, the new wedges would all be reinspected after one cycle of operation. Thereafter, two wedges would be alternately inspected every third outage. This would ensure that all four top guide wedges are inspected every ten years.

Internal commitments in References 18.5.17 and 18.5.19 below to address GE SIL No. 588 will no longer be applicable with the BWRVIP inspection strategy adopted herein. These commitments are considered revised accordingly, with the issuance of this document.

Cracking in grid beams has been discovered at Oyster Creek. BWRVIP-26, Paragraph 3.2.2, states that, for now, no inspection is required for the grid beams, but that this recommendation will be reevaluated after the Oyster Creek inspection and sample results have been evaluated by the BWRVIP. EPRI issued a report (Reference 18.5.37) that characterizes these cracked specimens. The report does not categorically state that the cause of the cracking in the top guide beam was IASCC, but it would be difficult to conclude otherwise given the reported data. Consequently, the BWRVIP will have to revisit BWRVIP-26 for top guide beam cracking and inspection recommendations. The subject EPRI report does not contain any recommendations or guidance, so no action is necessary at this time. (See commitment to GE SIL No. 554 below for further discussion of top guide grid beam inspection.)

18.2. <u>ASME Section XI Applicability</u>

The top guide is part of the core support structure; however, the top guide is not integrally welded as stated in the title of ASME Section XI, Table IWB-2500-1, Category B-N-2. Therefore the top guide is not subject to ASME Section XI. See Reference 18.5.28 below.

18.3. Other Commitments

In References 18.5.6 and 18.5.22 below, Vermont Yankee committed to perform examinations as recommended by GE SIL No. 554. This GE SIL recommends inspection of top guide grid beams, as they become accessible during the normal course of refueling outages. In Reference 18.5.25 below, Vermont Yankee stated that following RFO 21 (1999), inspection of the top guide grid beams will revert to that recommended by the BWRVIP. However, Reference 18.5.30 stated that top guide grid beams in four cells will be inspected until further notice. A change to that commitment was forwarded to the NRC in Reference 18.5.34. BWRVIP guidance governed inspection of the top guide grid beams until January 2004. As part of the power uprate approval process, VY committed in BVY 04-008 (Reference 18.5.40) to perform inspections of the top guide grid beams in accordance with SIL 554 requirements. VY committed to perform inspection of top guide components in the refueling outage following power uprate. Enhanced visual testing (EVT-1) of top guide grid beams will be performed in accordance with SIL 554 following sample selection and inspection frequency of BWRVIP-47 for the CRD guide tubes. In other words, VY will perform inspection of 10% of the total population of cells within twelve years, w... -half (5%) to be completed within six years. The six-year intervals at Vermont Yankee w... defined to be the same as those for the CRD guide tubes. The first top guide grid beam 6 ye... erval aligns with the CRD second six-year interval and is defined as RFO 24 (200... (2005), RFO 26 (2007), and RFO 27 (2008). The second top guide grid beam ... will coincide with the CRD third 6-year interval which begins with RFO 28 (2010) and includ... RFO 29 (2011). The sample is chosen from the cell locations where control blades will be ... Selection of the cells will also be biased to the highest fluence areas in the top gui... ...ever, Vermont Yankee reserves the right to modify the above inspection ... BWRVIP-26 be revised in the future.

18.4. ...pect...s for Ris... ...eration Purposes

...e are two top g... ...subcomponents that are to be inspected solely for risk to generation ... of one quadrant of the top guide rim bolts and the cover sheet bolts is ... every fourth refueling outage on a rotating basis. These components are no... ...ese nonmandatory inspections would only be performed for loose parts concern...

18.5. R...s

18.5.1. GE ... SIL No. 059, Revision 0, dated May 31, 1991, "Top Guide Crack Indication"

18.5.2. Letter C. B. Cameron to A. D. Himle, dated July 22, 1991, regarding SIL 462, RICSIL054, RICSIL059 and GE support for possible inspection findings

18.5.3. Memorandum C. B. Cameron to R. E. McCullough, dated August 5, 1991

18.5.4. GE SIL No. 554, dated April 6, 1993, "Top Guide Cracking"

18.5.5. Memorandum R. A. Woehlke/K. B. Spinney to T. G. Stetson, dated May 24, 1993, "Application of SIL No. 554 to VY"

18.5.6. Memorandum T. G. Stetson to R. E. McCullough, dated June 14, 1993, "Top Guide Cracking"

18.5.7. Memorandum T. G. Stetson to C. B. Cameron, dated July 19, 1993, "Delaying Commitment SIL0554RE1"

18.5.8. Service Request T. G. Stetson to D. C. Porter, dated February 15, 1994, "Service Request to Determine Radial Flux Profile On Vermont Yankee Top Guide" (Later Canceled)

18.5.9. Letter R. C. Hooper (GE) to F. J. Helin, dated May 4, 1994, "Follow-up Questions Asked During Our April 11, 1994, Meeting"

18.5.10. Memorandum F. J. Helin to D. C. Porter, dated May 10, 1994, "Cancel Service Request (94-18); Radial Flux Profile On Top Guide Evaluation"

18.5.11. GE RICSIL No. 071, Revision 0, dated November 22, 1994, "Top Guide and Core Plate Cracking"

18.5.12. Letter from BWRVIP to USNRC, dated January 3, 1995, "Request for Information Regarding the Impact of BWR Core Plate and Top Guide Ring Cracking"

18.5.13. GE SIL No. 588, dated February 17, 1995, "Top Guide and Core Plate Cracking"

18.5.14. NRC Information Notice 95-17, dated March 10, 1995, "Reactor Vessel Top Guide and Core Plate Cracking"

18.5.15. GE SIL No. 588, Revision 1, dated May 18, 1995, "Top Guide and Core Plate Cracking"

18.5.16. Memorandum T. G. Stetson to R. E. McCullough, dated February 5, 1996, "Response to Commitment RICSIL071 on Top Guide and Core Plate Cracking"

18.5.17. Memorandum T. G. Stetson to R. E. McCullough, dated February 5, 1996, "Response to Commitment SIL0588 on Top Guide and Core Plate Cracking"

18.5.18. Memorandum T. G. Stetson to R. E. McCullough, dated February 5, 1996, "Response to Commitment INF95017 on Top Guide and Core Plate Cracking"

18.5.19. Memorandum T. G. Stetson to R. E. McCullough, dated July 11, 1996, "Response to Commitment SIL0588_01"

18.5.20. Memorandum T. G. Stetson to F. J. Helin, dated September 18, 1996, "Recommendations for Remaining Top Guide Beam Inspections"

18.5.21. BWRVIP-26, dated December 1996, "BWR Top Guide Inspection and Flaw Evaluation Guidelines"

18.5.22. Letter Vermont Yankee to NRC dated September 30, 1997, "Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals"

18.5.23. Letter NRC to Vermont Yankee dated March 25, 1998, "Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals – Vermont Yankee Nuclear Power Station"

18.5.24. BWRVIP-50, dated May 1998, "Top Guide/Core Plate Repair Design Criteria"

18.5.25. Memorandum T. G. Stetson to F. J. Helin, dated September 18, 1998, "Recommendation for Remaining Top Guide Beam Inspections"

18.5.26. Memorandum T. G. Stetson to R. E. McCullough, dated December 23, 1998, "Response to Commitment UND96055"

18.5.27. Memorandum E. J. Taintor to D. C. Girroir, dated April 23, 1999, "Accessibility Following Installation of Proposed Top Guide and Core Support Assemblies"

18.5.28. Memorandum C. B. Larsen to D. C. Girroir, dated May 13, 1999, "Definition of Core Support Structures (ASME Section XI, Category B-N-2)"

18.5.29. Memorandum C. B. Larsen to D. C. Girroir, dated May 13, 1999, "1999 Top Guide Grid Inspection Plans"

18.5.30. Letter Vermont Yankee to USNRC, dated May 27, 1999, BVY 99-73, "Reactor Vessel Internal Plans for the 1999 and 2001 Refueling Outages"

18.5.31. Letter USNRC to BWRVIP, dated September 29, 1999, "Final Safety Evaluation of 'BWR Top Guide Inspection and Flaw Evaluation Guidelines (BWRVIP-26)'"

18.5.32. Letter Vermont Yankee to USNRC, dated October 29, 1999, BVY 99-137, "Deferral of Top Guide and Core Plate Wedge Installation"

18.5.33. Memorandum D. C. Girroir to P. B. Corbett, dated May 9, 2000, "Cost of Top Guide Inspections"

18.5.34. Letter Vermont Yankee to USNRC, dated June 6, 2000, BVY 00-56, "Change in Inspection Plans for the Top Guide Grid Beams"

18.5.35. Letter Vermont Yankee to USNRC, dated September 26, 2000, BVY 00-89, "Cancellation of Top Guide and Core Plate Wedge Installation"

18.5.36. Letter NRC to BWRVIP, dated December 7, 2000, "Acceptance for Referencing of BWRVIP, BWR Top Guide Inspection and Evaluation Guidelines (BWRVIP-26) Report for Compliance with the License Renewal Rule (10 CFR Part 54)"

18.5.37. EPRI Report 1003422, dated May 2002, "Analytical Transmission Electron Microscopy (ATEM) Characterization of Stress Corrosion Cracks in LWR-Irradiated Austenitic Stainless Steel Core Components"

18.5.38. Action Item / Regulatory Commitment BWRVIP-26-A_01, dated August 5, 2002, "Evaluate BWRVIP-26-A: BWR Top Guide Inspection and Flaw Evaluation Guidelines"

18.5.39. VY Calculation VYC-2218, Revision 0, dated November 25, 2002, "Structural Evaluation of RPV Top Guide Aligner"

18.5.40. BVY 04-008 Attachment 1-CPPU Submitted RAI Response, dated January 31, 2004.

19.0 Vessel Cladding

19.1. BWRVIP Document Applicability

BWRVIP documents do not contain any specific inspection requirements for vessel cladding.

19.2. ASME Section XI Applicability

The cladding is outside the scope of ASME Section XI. The examination volumes shown in Figures IWB-2500-1 specifically exclude the cladding.

19.3. Other Commitments

In RFO 16 (1992), cracking in cladding in the vessel head and shell interior was discovered at Vermont Yankee. The inspection was initiated in response to GE RICSIL No. 050 and GE SIL No. 539. A large sample of these clad cracks was ultrasonically sized. It was determined through this ultrasonic manual sizing, in conjunction with a statistical analysis, that none of the cracks propagated into the vessel base-material pressure boundary. Even so, Vermont Yankee committed to the NRC in References 19.5.6 and 19.5.14 below to perform successive examinations similar to ASME Section XI, IWB-2420(b), of a sample of clad cracks. In this way, the clad cracking would be treated as if these were indeed defects that exceeded the ASME Section XI acceptance criteria, even though they do not. Paragraph IWB-2420(b) requires that areas containing flaw indications that have been accepted analytically be reexamined during the next three inspection periods.

RFO 16 (1992) fell in the third period of the second interval. The vessel cladding was visually and ultrasonically reexamined in RFO 17 (1993), which fell in the first period of the third interval. Reference 19.5.14 below contains a commitment to the NRC to perform a similar clad inspection to the RFO 17 (1993) reexamination once each period for the next two periods. In RFO 19 (1996), the reactor vessel shell welds were examined using an automated ultrasonic technique (Reference 19.5.20). This constituted a very large sample of the vessel interior surface and was used to also serve as the second successive reexamination (second period, third interval) of the cladding flaws. During RFO 23 (2002), the third successive re-examination was completed (third period of the third interval). Reference 19.5.21 documented closure of the Reference 19.5.14 commitment; summarized the four inspections of the vessel clad cracking; and concluded that there was no evidence that clad cracks have penetrated into the base material.

Clad cracking will continue to be monitored through the following mechanism. Approximately every ten years, the vessel shell welds will be examined in accordance with ASME Section XI and BWRVIP-05. This is next scheduled to occur in RFO 24 (2004). Most of the vessel clad cracking was found to be located in areas of manually applied cladding. The manually applied cladding coincides with the vessel weld locations because of the original vessel fabrication sequence. Therefore, a large sample of clad cracked areas will be examined every ten years. This will give a very good indication of whether the clad cracks are likely to become a problem.

19.4. Inspections for Risk to Generation Purposes – None

19.5. References

19.5.1. GE RICSIL No. 050, dated April 23, 1990, "Reactor Vessel Head Clad Cracking"
19.5.2. Memorandum C. B. Cameron to R. E. McCullough, dated April 24, 1990, "Preliminary Review to RICSIL-050"
19.5.3. NRC Information Notice No. 90-29, dated April 30, 1990, "Cracking of Cladding and Its Heat-Affected Zone In the Base Metal of a Reactor Vessel Head"
19.5.4. Memorandum J. R. Hoffman to D. C. Girroir, dated June 8, 1990, "Input for AP0028 Close-out of NRC I. N. 90-29 and 90-32"
19.5.5. GE SIL No. 539, dated November 5, 1991, "RPV Head Clad Cracking"

19.5.6. Letter Vermont Yankee to USNRC, BVY 92-055, dated April 5, 1992, "Proposed Alternative for Compliance with 10CFR50.55a Regarding RPV Cladding Indications"

19.5.7. Letter Vermont Yankee to USNRC, BVY 92-056, dated April 10, 1992, "Supplemental Information Regarding Proposed Alternative for Compliance with 10CFR50.55a Regarding RPV Cladding Indications"

19.5.8. Memorandum F. J. Helin to PORC, dated April 17, 1992, "Clad Indications Found During 1992 Refueling Outage"

19.5.9. Letter USNRC to Vermont Yankee, dated April 17, 1992, "Meeting Summary of April 8, 1992 Meeting To Discuss Vermont Yankee Reactor Vessel Flaws"

19.5.10. Letter USNRC to Vermont Yankee, dated April 17, 1992, "Disposition of Reactor Vessel Cladding Indications Discovered During the March 1992 Refueling Outage At Vermont Yankee Nuclear Power Station"

19.5.11. EPRI TR-101971, dated February 1993, "Evaluation of Reactor Pressure Vessel Head Cracking in Two Domestic BWRs"

19.5.12. Letter Vermont Yankee to USNRC, dated July 1, 1993, "1993 Refueling Outage Vessel Clad Inspection Plans"

19.5.13. Memorandum C. B. Larsen to J. R. Hoffman, dated September 9, 1993, "RPV Clad Crack Investigation"

19.5.14. Letter Vermont Yankee to USNRC, BVY 93-112, dated October 6, 1993, "Reactor Vessel Clad Inspection during the 1993 Refueling Outage CAR92016MEC3"

19.5.15. Memorandum T. G. Stetson to R. E. McCullough, dated October 12, 1993, "Response to Commitment CAR92016RE1"

19.5.16. D. C. Girroir to G. Cappuccio, dated May 26, 1994, "CAT A Item: Dryer Support Bracket Inspections"

19.5.17. Memorandum G. A. Wallin to R. E. McCullough, dated April 18, 1995, "Response to Commitment CAR92016RE2"

19.5.18. Memorandum T. G. Stetson to R. E. McCullough, dated July 20, 1995, "Canceling Commitments CAR92016RE3 and CAR92016RE4"

19.5.19. Letter VYNPC to USNRC, BVY 96-105, dated September 10, 1996, "Augmented Examination of the Vermont Yankee Reactor Pressure Vessel Shell Welds"

19.5.20. SwRI Final Report, dated December 1996, Reactor Vessel Shell Weld Inspection Report

19.5.21. Memorandum C. B. Larsen to D. C. Girroir, dated October 21, 2002, "Evaluation of Clad Crack Indications under the Reactor Head and in the Vessel"

APPENDIX C

Technical Justifications

Table 1 – Technical Justification Index

No.	ID	Approved	Title	BWRVIP Reference
1	TE-2003-0021	04/09/03	Justification to Revert to EVT-1 Inspection of Jet Pump Circumferential Welds with UT Indications	BWRVIP-41, Section 3.2.4, Table 3.3-1
2	TE-2003-0023	07/07/03	Technical Assessment For Delaying Hydrogen Injection Into The Reactor Core	BWRVIP-79, Table 4-5a
3	TJ 2003-03	08/18/03	Justification to Perform Less Than 5% of CRD Guide Tube Weld Exams within the First Six-Year Interval	BWRVIP-47-A, Table 3.2-1
4	TJ 2003-04	08/18/03	Continued Operation without a Feedwater Zinc Injection System	BWRVIP-79, Section 3.2.3.3, BWRVIP-107, Section 5.1
5	TJ 2003-05	12/17/03	Feedwater Copper Concentrations Above Recommended Limits	BWRVIP-79, Table 4-6, Note c
6	TJ-2004-01	03/26/04	Justification for Alternative Inspection of Core Plate Rim Hold-down Bolts	BWRVIP-25, Section 3.2.2.2, Table 3-2
7	TJ-2004-02	03/26/04	Justification for Deferral of Inspection of Inaccessible Welds	BWRVIP-18, Section 3.2.4, BWRVIP-41, Table 3.3-1
8	TE-2004-0018	In review	Justification to Inspect Portions of Shroud Horizontal Welds H1, H2, and H3 On the OD In Lieu of the Top Guide Spacer Block Welds, the Shroud Flange Ring Segment Welds, and the Top Guide Ring Segment Welds	BWRVIP-76, Section 3.2, Section 2.2.1, Section 2.2.2, Figure 2-3
9	TJ-2004-04	03/26/04	Justification to Defer Inspection for Detection of Transverse Flaws In Shroud Support Weld H9	BWRVIP-104, Section 9.2
10	TJ-2004-05	03/26/04	Justification for Deferral for UT of SLC Safe-end	BWRVIP-27-A, Sections 3.3.1 and 3.4.1

04/03/03

Technical Evaluation No. 2003-0021

Technical Evaluation No. TE-2003-0021

Title: Justification to Revert to EVT-1 Inspection of Jet Pump Circumferential Welds with UT Indications

■ QA (Safety Class, OQA, or Vital Fire) ☐ Non QA (Non-Safety) (Check One)

Background *(Enter a concise summary of the condition or reason for the requested TE stating the existing condition and the desired results. State the scope of the requested TE.)*

Vermont Yankee desires to use EVT-1 (enhanced visual testing) for subsequent inspections of six jet pump welds with reported UT (ultrasonic testing) indications, rather than continue reexamination by UT of those welds. This TE justifies this decision.

The BWRVIP was initially formed out of the BWROG to address inter-granular stress corrosion cracking (IGSCC) of reactor internals. Vermont Yankee is bound by certain commitments to follow the guidance of the BWRVIP (References 1A through 1G). BWRVIP-94 (Reference 1F), Appendix A states that a technical justification shall be required when utility methodology is inconsistent with the intent of the supporting BWRVIP guidelines.

BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines" (Reference 2A), in Table 3.3-1 requires examination of 50% of RS-1 (thermal sleeve-to-elbow) welds, DF-2 (diffuser shell-to-tailpipe) welds, and DF-3 (diffuser tailpipe-to-adapter) welds (among many others) every inspection cycle. "Inspection cycle" is defined in BWRVIP-41, Section 3.2.1, as six years. The method of inspection that is required per BWRVIP-41 is MVT-1 (modified visual testing). Per Reference 2C, the BWRVIP committed to the NRC to replace the MVT-1 method with the EVT-1 method (enhanced visual testing). The NRC final SER of BWRVIP-18 (Reference 2D), reflects this agreement. BWRVIP-41, Paragraph 3.2.4, states that, "In all cases where a [visual] inspection is recommended...a suitable NDE examination technique meeting the requirements of BWRVIP-03 may be substituted." Therefore, for these welds, EVT-1 or UT (ultrasonic testing) is acceptable.

Vermont Yankee elected to perform inspection of the RS-1 welds (among others) using UT in RFO 20 (1998). The ultrasonic inspection identified indications on four RS-1 welds: weld numbers N2B-RS-1, N2C-RS-1, N2H-RS-1, and N2K-RS-1 (Reference 3A). BVY-98-67 (Reference 4A) was prepared and submitted to the NRC. Following an RAI (NVY 98-77, Reference 4B) and subsequent reply, (BVY 98-112, Reference 4C), Vermont Yankee received an SER from the NRC (NVY 98-153, Reference 4D) to allow deferral of inspection for these four welds until RFO 22 (2001). Letter BVY 99-43 to the NRC (Reference 4E) sought to extend the inspection interval from one to two cycles. VY received an SER for this in letter NVY 99-46 (Reference 4F). During RFO 22 these four riser welds were reinspected (Reference 3B) with the result that two of the previous indications (in welds N2B-RS-1 and N2C-RS-1) were found to be liftoff of the transducers, and therefore nonrelevant. The indications in the remaining two welds (N2H-RS-1 and N2K-RS-1) were the same as found in RFO 20 (1998) within the documented NDE accuracy (Reference 4I and BWRVIP-03, Sections 10.4.3 and 10.6.1, Reference 2B). An

Technical Evaluation No. 2003-0021

indication was discovered in weld N2K-RS-1 during RFO 22 (2001) that had not been recorded in RFO 20 (1998), but it was determined that this indication was indeed in the UT data from RFO 20 (1998), so in fact it was not a new indication (see evaluation in GE report, Reference 3B). TE No. 2001-030 was generated to justify operation for two cycles based on the reported indications. In addition, EVT-1 was performed during RFO 22 (2001) of weld N2K-RS-1 in the area of the reported indications (Reference 3C). No indications of cracking were observed.

Vermont Yankee elected to perform inspection of the DF-2 and DF-3 welds (among others) using UT in RFO 21 (1999). The ultrasonic inspection identified indications in four diffuser welds: weld numbers 2-DF-2, 3-DF-3, 9-DF-2, and 10-DF-2 (Reference 5A). Vermont Yankee performed an evaluation (Reference 6B) to allow deferral of inspection for the most limiting of these four welds until RFO 23 (2002). The RFO 23 (2002) UT measured flaw lengths (Reference 5B) were the same as found in RFO 21 (1999) within the documented NDE accuracy (Reference 6A and BWRVIP-03, Sections 10.4.6 and 10.6.3, Reference 2B). Because the indication lengths were the same as before, a flaw evaluation was not necessary. In addition, a visual inspection was performed of the OD and ID surfaces of weld 2-DF-2 and the ID surfaces of welds 9-DF-2 and 10-DF-2 during RFO 21 (1999) (Reference 5A). Again during RFO 23 (2002), visual inspections were performed of the ID of welds 2-DF-2, 3-DF-3, 9-DF-2, and 10-DF-2 (Reference 5B). All of these visual inspections were performed primarily to determine if there were geometric features in the welds – none were noted, but no cracks were detected either.

To summarize, there are six welds with recorded UT indications in the jet pumps. None of these indications were confirmed by visual inspection. Two welds are riser RS-1 welds and the other four are circumferential diffuser welds. All six welds were reinspected by UT after two cycles of operation and there were no new indications or indication growth (within the NDE uncertainty error).

In this instance, it is not clear that VY does not intend to follow BWRVIP guidance because BWRVIP-41 states that both EVT-1 and UT are acceptable. However, it is common practice throughout industry that when flaws are identified to repeat the examination using the same method that originally found the flaw. Additionally, INPO has identified recommendations in this area in related cases. See Reference 2L, an email which documents a conference call organized by INPO – with Monticello, BWRVIP Assessment Committee, INPO, and EPRI representatives – to determine whether Monticello was outside the bounds of BWRVIP guidance when they reverted to EVT-1 after identifying indications with UT. Monticello was criticized during their INPO Review Visit for reverting to EVT-1 without establishing that there was no growth by UT. Like VY's indications, Monticello's were not apparent visually. However, the general feeling was that it would have been okay to revert to EVT-1 in a case where UT verified that there was no flaw growth. This general agreement was endorsed by the BWRVIP during the January 2003 Assessment Committee meeting. The consensus there was that if a plant performed a reinspection with UT (as VY has completed), then it would be acceptable to implement EVT-1 in subsequent inspections. Monticello has argued that per BWRVIP guidance, EVT-1 and UT are considered interchangeable, and that they shouldn't be penalized for choosing

Technical Evaluation No. 2003-0021

UT to perform their baseline. Most of the phone conference participants did not agree with this reasoning and felt that Monticello should conduct further UT. This TE is prepared in order to formalize the Assessment Committee consensus and to differentiate VY's circumstances from Monticello.

Discussion ***(Record the evaluation considerations and the results of the evaluation. Describe any features that required special attention during the TE process. Document and validate any assumptions made during the evaluation.)***

When BWRVIP-41 was developed, the BWRVIP focus group responsible for jet pump inspection and evaluation determined that visual inspection of the riser and diffuser welds was adequate. (As background information, in austenitic [stainless] steel materials, IGSCC initiates in the heat affected zone [not in the weld itself] and must start on either the ID or OD surface; it does not initiate mid-wall.) The adequacy of a visual inspection is explained in BWRVIP-41, Section 3.2.4 (Reference 2A). It states, "A visual examination was determined to be technically justified for the jet pump assembly welds. These welds are typically uncreviced and the wall is thin as compared to the circumference of the component. With this configuration, it is the expectation based on field experience...that the outer diameter (OD) and inner diameter (ID) cracking would not be significantly different. Also, research shows that for conventional groove welds in piping less than 10 inches in diameter, the residual stress is expected to exhibit variation in amplitude around the circumference. Since IGSCC initiation is dependent upon tensile stress magnitude, residual stress variation leads to a greater variability in crack initiation time, and the likelihood that a single crack will grow through-wall before extensive circumferential cracking occurs...Even if cracking is initiated on the ID, the thin walled configuration would promote a crack to propagate through wall such that VT inspection from the OD is appropriate." Restated, the thin wall of the jet pumps is an advantage in that cracking would manifest itself visually on the inspection surface (either OD or ID) before its length approaches the structural integrity limit.

The table below (derived from References 3B and 5B) provides the indication length data:

	Indication Lengths			
Riser Weld	**First UT (1998)**	**Second UT (2001)**	**Delta**	**NDE Uncertainty**
N2H-RS-1, Indication 1	9° (0.84")	9.8° (0.91")	+ 0.8° (0.07")	6° (0.56")
N2K-RS-1, Indication 1	30° (2.8")	30° (2.8")	0.0° (0.00")	6° (0.56")
N2K-RS-1, Indication 2	20° (1.9")	16° (1.5")	- 4.0° (0.37")	6° (0.56")
Diffuser Welds	**First UT (1999)**	**Second UT (2002)**	**Delta**	**NDE Uncertainty**
2-DF-2, Indication 1	10.2° (1.26")	9.5° (1.17")	- 0.7° (- 0.09")	2.6° (0.32")
3-DF-3, Indication 1	4° (0.51")	3.5° (0.45")	- 0.5° (- 0.06")	2.6° (0.32")
3-DF-3, Indication 2	8° (1.03")	7.2° (0.93")	- 0.8° (- 0.09")	2.6° (0.32")
3-DF-3, Indication 3	6° (0.77")	5.6° (0.72")	- 0.4° (- 0.09")	2.6° (0.32")
9-DF-2, Indication 1	11° (1.40")	11° (1.36")	0° (0.00")	2.6° (0.32")
10-DF-2, Indication 1	11° (1.40")	12.5° (1.54")	+ 1.5° (+ 0.14")	2.6° (0.32")
10-DF-2, Indication 2	6° (0.77")	8.4° (1.0")	+ 2.4° (+ 0.23")	2.6° (0.32")

Technical Evaluation No. 2003-0021

In addition to the ultrasonic data, visual inspections were conducted of certain of these welds as summarized in the table below (derived from References 3C, 5A, and 5B).

Visual Inspections			
Weld	**Outage Performed**	**Method and Scope**	**Findings**
Riser Weld			
N2K-RS-1	RFO 22 (2001)	EVT-1 in area of UT indications	None
Diffuser Welds			
2-DF-2	RFO 21 (1999)	VT of OD and ID	None
9-DF-2	RFO 21 (1999)	VT of ID	None
10-DF-2	RFO 21 (1999)	VT of ID	None
2-DF-2	RFO 23 (2002)	VT of ID	None
3-DF-3	RFO 23 (2002)	VT of ID	None
9-DF-2	RFO 23 (2002)	VT of ID	None
10-DF-2	RFO 23 (2002)	VT of ID	None

There are three possible explanations for the ultrasonic indications' lack of growth and no detection visually of cracking:

1) The indications are not cracks and are instead nonrelevant. Nonrelevant means that the indications are not from flaws. For ultrasonic testing, possible sources of nonrelevant indications are part/weld geometry, metallurgical interfaces (but unlikely in these welds), or transducer liftoff – among others. Nonrelevant ultrasonic indications from geometry can be caused by weld drop-through, steeply edged weld crown, weld ripples, joint mismatch, or undercut. The riser welds were scanned from the OD of the riser elbow, and the ID was not accessible for visual confirmation of possible geometric conditions that might have caused ultrasonic reflectors. Conversely, the diffuser welds were scanned from the ID, and the jet pump exterior was not always accessible for visual confirmation of possible geometric conditions. It is also possible that indications could have been caused by transducer liftoff. For example, two indications identified in the 1998 riser examination were later identified to be liftoff. For some reason, liftoff signals in UT data from underwater contact testing appear to behave more like flaws (as opposed to liftoff signals from more conventional UT contact testing in air).

2) The indications are not cracks and instead originate from fabrication flaws, such as lack-of-fusion or incomplete penetration. For the riser welds, the original nondestructive test specified was a penetrant test (PT) on the root pass and final surface (References 2J and 2K). Such penetrant tests may not detect all lack-of-fusion or incomplete penetration in these welds, as would an ultrasonic test. It is not known what NDE was specified for the diffuser welds, because these were shop welds. It is unlikely, though, that a more rigorous NDE would be specified for the shop welds (e.g. UT) than for the field welds (i.e. PT).

Technical Evaluation No. 2003-0021

3) The indications are indeed cracks, but the cracks are not actively growing. The currently accepted conservative crack growth rate established by the BWRVIP in BWRVIP-14 (Reference 2E) and agreed to by the NRC (Reference 2F) is 5E-5 inches per hour. The flaw evaluations (References 4J and 6B) performed to assess the structural integrity of each jet pump circumferential weld with flaws assumed this crack growth rate. Because the observed delta between the first and second ultrasonic tests is within the NDE technique uncertainty, no flaw growth must be assumed, and it would not make sense to compute a flaw growth rate (especially considering the apparent negative flaw growth in half the cases). Nonetheless, the NDE uncertainty for the riser welds would equate to a growth rate of 2.1E-5 inches per hour applied over two cycles, if applied in the positive direction. The NDE uncertainty for the diffuser welds would equate to a growth rate of 1.2E-5 inches per hour over two cycles, if applied in the positive direction. Clearly, neither the observed data, nor the NDE uncertainty applied over two cycles, approaches 5E-5 inches per hour.

In summary, it can be concluded that either: 1) the UT indications are nonrelevant (e.g., geometry); 2) the UT indications are fabrication flaws; or 3) the UT indications are from cracks, but they are not actively growing.

The visual inspections performed also support this conclusion. No indication of cracking has been detected in any of the visual inspections performed. Any of the three possible scenarios is supported by this observation. If the UT indications are nonrelevant or fabrication flaws, then of course no cracking would be observed; and if the UT indications are inactive cracks, then they may not have propagated through-wall to the surface where they would have been visible. On the contrary, if the indications were cracks and the cracks were active, then a visual confirmation of the cracking would be expected following two cycles – given the thin wall of the jet pumps. This statement must be tempered by the fact that for the risers only one of two welds was visually inspected, and for the diffusers the method of visual was not EVT-1, and therefore may not have detected fine tight IGSCC.

Assuming the worst case – that the UT indications are inactive or slow-growing cracks, an EVT-1 quality inspection of either the OD or ID surface will reveal this cracking prior to encroachment on the weld structural integrity limit as determined in References 4J and 6B.

A further assurance of safe operation is established in Technical Specification 4.6.F (Reference 2M), which requires that jet pump integrity and operability be checked daily by monitoring jet pump flow characteristics. In addition, AP 0028 commitment BWRVIP-028-A_04 (Reference 2G) instituted an LPC to OP 4110 to monitor jet pump M ratio, a more sensitive indicator of possible jet pump leakage (see discussion in Section 4 of BWRVIP-028-A, Reference 2G). Section 2.6.2 of BWRVIP-06-A (Reference 2H) contains further discussion of how failures of various jet pump locations could be detected during operation.

Even though it has been demonstrated that there is no active cracking in these welds, VY will maintain the two-cycle inspection frequency determined from the flaw evaluations, which used the

Technical Evaluation No. 2003-0021

UT indication data (References 4J and 6B). If after three successive inspections with no recorded indications of cracks, VY will revert to the six-year inspection interval specified in BWRVIP-41 for these welds. This is similar philosophically to ASME Section XI requirements for successive inspections of welds with rejectable indications.

Assumptions/Open Items *(List any assumptions used in the TE and provide a basis for each. List any open items requiring additional action prior to closure of the TE.)*

None

Material Requirements/Implementation Instructions *(List any identified specifications for equipment, materials, or services needed to implement the recommendations of the TE. Specify any special implementation instructions or cautions, such as field testing requirements or system interface requirements during implementation.)*

None

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

Based on the above analysis, this TE recommends the following actions:

1. Revise PP 7027 and the Bases Document to specify EVT-1 for the subject welds. Maintain the two-cycle (three-year) inspection frequency through RFO 28 for the riser welds and RFO 29 for the diffuser welds. Then revert to the six-year inspection interval specified in BWRVIP-41 if no indications are recorded.
   Responsible Department – System Engineering, Code Programs
   Due Date – 02/26/04

2. Per Reference 1F requirements, include this Technical Evaluation as part of the BWRVIP program (reference in PP 7027 and Bases Document).
   Responsible Department – System Engineering, Code Programs
   Due Date – 02/26/04

Technical Evaluation No. 2003-0021

Approvals/Closeout ***(Print name and provide signature/date.)***

[signature] 1/4/3/03 **Reactor Internals Management Program Coordinator (CE)**

[signature] 1/4/7/03 **Independent Reviewer (Mechanical/Structural Design)**

[signature] 1/4/3/03 **Independent Reviewer (Code Programs)**

Dennis Girroir/[signature] 4/9/03 **Code Programs ~~Manager~~** Supervisor

(signature) **(date)**

Closeout ***(All actions that were recommended by the TE and accepted by management have been initiated and any identified open items have been dispositioned.)***

__________________ / __________ **CE**

(signature) **(date)**

Attachments ***(Provide a list of all forms, document markups, etc. provided as part of the TE package.)*** **– Please see the Technical Evaluation Data Base which lists all references used, and notes which references are attached.**

VYAPF 6045.02
AP 6045 Original
Page 7 of 10

Technical Evaluation No. 2003-0021

TECHNICAL EVALUATION DATABASE INPUT

TE No.: 2003-0021

TE Title: Justification to Revert to EVT-1 Inspection for the Jet Pump Riser and Diffuser Circumferential Welds with UT Indications

Keywords: Jet Pump, Reactor Internals, Ultrasonic Testing, BWRVIP

Design Input Documents - The following documents provide design input to this TE.

#	Document Title (including Rev. No. and Date, if applicable)
1	**Vermont Yankee's explicit and implicit commitment to the guidance of the BWRVIP.**
1A	PP 7027, "Reactor Vessel Internals Management Program"
1B	Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC) dated May 30, 1997, "BWR Utility Commitments to the BWRVIP"
1C	BVY 97-123, dated September 30, 1997, "Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals"
1D	Letter Brian Sheron (USNRC) to Carl Terry (BWRVIP Executive Chairman), dated July 29, 1997, "BWR Utility Commitments to the BWRVIP"
1E	Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC) dated October 30, 1997, "BWR Utility Commitments to the BWRVIP"
1F	BWRVIP-94, dated August 2001, "BWRIP Program Implementation Guide" EPRI TR 1006288
1G	Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC), dated April 16, 2002, "Project No. 704 – Utility Implementation of BWRVIP Products"
2	**Miscellaneous Background Input for this Evaluation**
2A	BWRVIP-41, dated October 1997, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines" EPRI TR-108728
2B	BWRVIP-03, Revision 5, dated December 2002, "Reactor Pressure Vessel and Internals Examination Guidelines"
2C	Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC) dated May 30, 1997, "BWR Utility Commitments to the BWRVIP"
2D	Letter Jack Strosnider (USNRC) to Carl Terry (BWRVIP Executive Chairman), dated January 11, 1999, "BWRVIP Response to NRC Safety Evaluation of BWRVIP-18"
2E	BWRVIP-14, dated March 1996, "Evaluation of Crack Growth in BWR Stainless Steel RPV Internals" EPRI TR-105873
2F	Letter Jack Strosnider (USNRC) to Carl Terry (BWRVIP Executive Chairman), dated December 3, 1999, "Final Safety Evaluation of Proprietary Report TR 105873 'BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals (BWRVIP-14)' (TAC No. M94975)"
2G	BWRVIP-28-A, dated April 2002, "Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking"
2H	BWRVIP-06-A, dated March 2002, "Safety Assessment of BWR Reactor Internals"

Technical Evaluation No. 2003-0021

2I	GE Nuclear Energy Report No. GE-NE-B13-01935, Revision 2, dated July 1999, "Jet Pump Assembly Welds Flaw Evaluation Handbook for Vermont Yankee"
2J	GE Quality Assurance Inspection Instruction ISE-WIS-301, dated June 15, 1970
2K	Installation and Service Weld Inspection Specification ISE-WIS-301, dated August 9, 1971
2L	Email Carl Larsen to Dennis Girroir, Summary of BWRVIP Conference Call, dated December 16, 2002
2M	Technical Specification 4.6.F
2N	AP 0028 Commitment Item BWRVIP-028-A_04, closed November 15, 2002, "Revise OP 4110 – Jet Pump M Ratio Required Action"
3	RFOs 20 and 22 NDE Reports for Jet Pump Riser Weld (RS-1) Indications
3A	GE Nuclear Energy Report No. 1HQXE, Revision 1, dated April 1998, "Vermont Yankee Nuclear Plant Recirculation Inlet Riser Ultrasonic Examination"
3B	GE Nuclear Energy Report No. JXOAL, Revision 0, dated May 2001, "Vermont Yankee Nuclear Plant Recirculation Inlet Riser Ultrasonic Examination"
3C	Framatome ANP Report No. 1600515, dated May 13, 2001, "2001 RFO-22 Reactor, In-Vessel Services Final Report"
4	Evaluations and Correspondence regarding Jet Pump Riser (RS-1) Welds
4A	Letter VYNPC to USNRC, dated May 4, 1998, BVY 98-67, "Jet Pump Riser Circumferential Weld Inspections"
4B	Letter USNRC to VYNPC, dated June 3, 1998, NVY 98-77, "Request for Additional Information Regarding Jet Pump Riser Circumferential Weld Inspections at Vermont Yankee Nuclear Power Station (TAC No. MA 1681)"
4C	Letter VYNPC to USNRC, dated July 30, 1998, BVY 98-112, "Response To Request for Additional Information Regarding Jet Pump Riser Circumferential Weld Inspections"
4D	Letter USNRC to VYNPC, dated October 26, 1998, NVY 98-153, "Jet Pump Riser Circumferential Weld Inspections at Vermont Yankee Nuclear Power Station (TAC No. MA 1681)" [includes original one-cycle SER]
4E	Letter VYNPC to USNRC, dated March 29, 1999, BVY 99-43, "Jet Pump Riser Circumferential Weld Inspections and Flaw Evaluation"
4F	Letter USNRC to VYNPC, dated April 29, 1999, NVY 99-46, "Jet Pump Riser Circumferential Weld Inspections at Vermont Yankee Nuclear Power Station (TAC No. MA5109)" [includes two-cycle SER]
4G	Memorandum John Hoffman to Tom Silko, dated July 30, 1999, "Jet Pump Riser Flaws for Cycle 21 Operation"
4H	Memorandum J. M. Abdelghany to John Hoffman, dated October 22, 1999, "Vermont Yankee Allowable Jet Pump Weld Leakage Rates for LOCA and Recirculation Pump Performance"
4I	Memorandum C. B. Larsen to John Hoffman, dated May 9, 2001, "Jet Pump NDE Uncertainty"
4J	Technical Evaluation No. 2001-030, dated May 14, 2001, "Evaluation of Jet Pump Riser Flaws"
5	RFOs 21 and 23 NDE Reports for Jet Pump Diffuser Weld (DF-2 and DF-3) Indications
5A	Framatome Technologies Report for Job 1220685, Revision 0, dated December 9, 1999,

Technical Evaluation No. 2003-0021

	"BWR Reactor Vessel Internals Inspection Ultrasonic Examination Report for Vermont Yankee Nuclear Power Station"
5B	Framatome ANP Report for Job 1220825, Revision 0, dated December 9, 2002, "BWR Reactor Vessel Internals Inspection Ultrasonic Examination Report for Entergy Nuclear Northeast Vermont Yankee"
6	**Evaluations and Correspondence regarding Jet Pump Diffuser (DF-2 and DF-3) Welds**
6A	Memorandum C. B. Larsen to John Hoffman, dated November 15, 1999, "Application of Uncertainty to Jet Pump Diffuser UT Indications"
6B	Memorandum John Hoffman to D. C. Girroir, VYM 99/134, dated November 26, 1999, "Jet Pump Assembly Inspection Discrepancy Report Evaluation"

Design Output Documents - The following documents are impacted by this TE.

#	Document Title
PP 7027	Reactor Vessel Internals Management Program

General References

#	Reference Title (including Rev. No. and Date, if applicable)
	None

Data Entered into Database ____________ / ________ **Entry Verified** ____________ / ________

Signature **Date** **Signature** **Date**

VYAPF 6045.02
AP 6045 Original
Page 10 of 10

Technical Evaluation No. 2003-023

Title: BWRVIP-79 Technical Justification For Delaying Hydrogen Injection Into The Reactor Core

_ QA (Safety Class, OQA, or Vital Fire) ■ Non QA (Non-Safety) (Check One)

Background *(Enter a concise summary of the condition or reason for the requested TE stating the existing condition and the desired results. State the scope of the requested TE.)*

Per Reference 1 and 2, VY installed a hydrogen water chemistry (HWC) system and injected noble metals to minimize risk of reactor vessel inter-granular stress corrosion cracking (IGSCC). This HWC system met Boiling Water Reactor Vessel Internal Project (BWRVIP) guidance (Reference 3 and sub-references), and therefore met VY's commitment to the NRC to follow this guidance (Reference 4A through 4D). Just prior to HWC online acceptance testing, however, VY developed fuel-cladding perforations (Reference 5). Subsequently, this condition degraded further and VY performed a mid-cycle outage to remove both the perforated-clad fuel and fuel with similar material/exposure configuration. Although subsequent detailed inspection of this fuel condition with team collaborators (Reference 6) identified potential causes for fuel failure (Reference 7, and recently Reference 23), it did not decisively identify the root cause.

After completion of initial HWC startup testing, GE recommended per Reference 8 that hydrogen injection not be implemented until the damaged fuel was inspected. Concurrently, VY management determined that hydrogen injection (a potential additional stressor to the fuel) should not proceed until investigators identified the fuel failure root cause. Per Reference 9, GE subsequently issued a draft letter recommending that hydrogen be injected at low levels, but was silent on resolving root cause for fuel failures. VY then sought further consultation (Reference 10) and evaluation, and determined that hydrogen injection starting November 2003 would meet optimum prudence for protecting both the fuel and reactor vessel components. This "delay" decision, however, varies from the BWR Water Chemistry Guidelines guidance (e.g., BWRVIP-79- Reference 3A) which states, "if HWC is not incorporated into the station chemistry program, an engineering evaluation should be performed to support the decision using an assessment methodology similar to that presented in Appendix A."

This TE meets BWRVIP-94 (Reference 3B) justification requirements to vary from BWRVIP-79, Table 4-5a, Note a, by:

a) Providing the technical rationale for delaying hydrogen injection,
b) Showing that the level of conservatism for preserving both fuel and vessel integrity meets the intent of the BWRVIP guidelines, particularly on a short term basis,
c) Identifying that alternate partial mitigation is already being provided,
d) Stating that the TE will be in effect until November 2003.

This TE also looks forward, and makes recommendations on addressing HWC for a) extended power up-rate needs and b) future installation of noble metals.

Discussion *(Record the evaluation considerations and the results of the evaluation. Describe any features that required special attention during the TE process. Document and validate any assumptions made during the evaluation.)*

Chemistry, Reactor Engineering, Codes Programs, and Design Engineering identified the following key areas requiring risk assessment to properly determine when hydrogen should be prudently injected into the reactor system:

1. Fuel Performance: After Reactor Engineering (RE) identified the fuel-cladding defects, VY Management decided to have RE and GNF conservatively design Cycle 22B to replace 44 fuel assemblies (including 5 damaged pins in 4 assemblies, 35 assemblies that could potentially sustain damage, and 4 additional assemblies – Reference 24). This strategy ensured that Cycle 23 could be reached without further interruption, and avoided the Brown's Ferry 2 configuration where 25 + fuel-cladding perforations eventually developed. VY then had GNF, EPRI, and an independent expert inspect the 4 damaged assemblies to establish the root cause. Per Reference 7 and 23, GNF determined that although the fuel was exposed to several stressors (such as the presence of copper, low alloy content of some fuel clad, noble metals, fuel power history), no root cause failure could be determined. The GNF inspection team chemically analyzed the fuel crud scrapes and used visual examination to make their conclusions. Based on this information, VY Management determined that no additional stressors should be placed on the fuel at this time until either more root cause information became available and/or before more fuel performance history (e.g. operation without cladding perforations) could be demonstrated.

Technical Evaluation No. 2003-023

2. Offsite Dose Impact: Based on initial noble metals injection dose rates and tight site boundary dose limits (e.g. per Reference 11, 20 mrem/year), Design Engineering initially identified that HWC dose performance could exceed site boundary dose limits. Since that time, Duke Engineering (now Framatome Engineering, Reference 12) has refined VY dose assessment methods. Design Engineering has noted per Reference 13, that additional dose from HWC operation should be manageable. VY can achieve the following controls by starting the HWC system late in the year (e.g. November 2003):
   a) the actual margins can be best quantified
   b) the HWC system can be shut down with a minimum of plant impact, and
   c) higher-than-predicted dose projections (if they occur) can be more easily managed relative to site boundary dose limits.
3. Noble Metal Protection: Per dissolved reactor water oxygen measurements (Reference 14A) and initial ECP probe testing (Reference 14B), Chemistry confirmed that VY received significant chemical protection from the presence of noble metals alone. Dissolved oxygen levels decreased from 200 ppb to 100 ppb. Both platinum and iron electro-chemical potential (ECP) electrodes indicated –40 millivolts prior to HWC testing without hydrogen injection. This protection value was among the lowest achieved in the industry when hydrogen was not being injected. Noting that many plants have an initial ECP value of between +100 to +200 millivolts without noble metals, VY's reduction in projected crack growth rate per Reference 15 is estimated to have been reduced by a factor of about 2. This is a significant reduction in crack growth and is a theoretical measure of reduced IGSCC activity.
4. Reactor Vessel Internals Assessment: Based on the above, Code Programs reviewed the reactor vessel health report (Reference 16), to determine whether identified vessel internal indications had grown and required immediate mitigation. Based on recent outage inspections, all confirmed existing indications have remained the same with no identified growth. Based on:
   a) the overall stable condition of vessel indications,
   b) the work towards assuring that there are two methods to mitigate IGSCC for most austentic piping welds regardless of water chemistry (Reference 22)
   c) the additional protection being received by noble metals alone,

   Code Programs has judged that HWC startup in November 2003 to be acceptable.
5. BWRVIP-79 Appendix A Evaluation: During the HWC scope phase, HWC Project Management performed a BWRVIP-79 Appendix A evaluation to determine HWC cost effectiveness. The evaluation showed that HWC would be cost effective, based on potential IGSCC damage to key vessel internals. This evaluation, however, did not evaluate the potential offsite dose or economic factors associated with potential failed fuel cladding, and the necessity to remove fuel. Based on:
   - internal vessel performance to date (e.g. no current areas of concern, pre-emptive repairs made)
   - Cycle 22B outage (e.g. outage time, fuel replacement costs)

   Management determined that a short delay to determine the fuel cladding failure root cause, was more cost effective than going forward and potentially creating a reactor vessel environment with HWC that caused cladding failures and required a second mid-cycle outage to mitigate consequences.
6. Reactor Vessel Chemistry: Since noble metals application, the reactor vessel has become a more reducing (versus oxidizing) environment, and has increased IGSCC resistance. Chemistry has identified that vessel chemistry has been in transition since noble metal application in May 2001. This transition is similar to plants that have injected hydrogen, with or without noble metals, but length of transition is significantly longer. VY is in a "slow motion" transition because vessel ECP (-40 millivolts to +55 millivolts) is less than standard HWC vessel ECP (nominally –230 to –500 millivolts). Although vessel chemistry has not reached steady-state conditions, the extended transition is further proof that VY is receiving significant IGSCC protection from noble metals application alone, and that overall, is in a desirable configuration. Chemistry has noted the following:
   a) Moderator metal ion inventory - After initial noble metals application and various system upsets, the inventory of metal ions has generally trended downward in the reactor moderator over time (Reference 17). This phenomenon is expected in a reducing environment and is generally considered beneficial (e.g. VY has seen a distinct reduction in copper). However, this phenomenon has also reduced beneficial ions (e.g. zinc) and per fuel scraping measurements (Reference 7) may have resulted in more copper and zinc deposits on fuel. Although the ion configuration is still changing and is sensitive to system variables (e.g. circulation water temperature, power changes, resin changes, etc.), key ion ratios (e.g. zinc to cobalt, iron to nickel) have remained near recommended values to date. These values will require close monitoring to determine if zinc injection will be required in future (particularly if the main condenser is re-tubed or replaced) and copper deposition is a continuing issue.

Technical Evaluation No. 2003-023

b) "Rust" findings – During recent refurbishment, Maintenance identified loose adhering iron oxide in both Reactor Water Cleanup and Recirc Pump seals. Again, based on peer plant and GE (Reference 9) feedback, reducing environments will cause release of soluble and insoluble iron oxide as reactor film surfaces convert from a loose Fe_2O_3 oxide film to a desirable, tight Fe_3O_4 oxide film. Although loose oxide could impact performance (e.g. valve seat leakage) and vessel chemistry has not reached steady state conditions, this conversion to a more durable oxide film on vessel components is still viewed as desirable.

c) Copper – Copper is generally considered a detrimental ion, because it can initiate corrosion on piping, cause crud induced localized corrosion (CILC) on fuel, and encourage formation of N-16 laden ammonia in non-noble metal environments. However, per Reference 18, copper acts as a catalyst to break up hydrogen peroxide, which is the most significant oxidant in the reactor. VY has seen this effect based on HWC startup tests (Reference 14B). In these tests, VY needed approximately 45% less hydrogen then peer plants to achieve full ECP protection (Reference 9). Based on the above:
   - Noble metals presence has established a more reducing vessel environment than originally present, and has reduced the coolant concentration of metal ions, including copper, which aids in slowing down vessel/component and fuel corrosion.
   - A small copper residual in FW (within BWRVIP-79 limits) remains and appears to help catalyze vessel hydrogen peroxide. Therefore, with VY's current noble-metal-without-hydrogen configuration, low copper concentrations are considered beneficial, and need to be closely monitored to confirm that they remain beneficial (e.g. ion concentration, continued ECP benefit).
   - By keeping noble metal concentration high enough (e.g. per Reference 3D above 0.1 micrograms/cm**2 to ensure adequate metal surface reduction reactions), and copper concentration at low but adequate catalytic levels (to ensure hydrogen peroxide breakdown in the bulk fluid), VY can receive adequate interim protection.

d) Crack flanking – Per Reference 19, crack propagation can occur even in a reducing environment when either noble metals or noble-metals/ hydrogen combination are in low concentrations. For VY's current noble metal configuration, it is important to ensure that VY maintain a relatively high concentration of noble metals, so that these materials can migrate into existing cracks, and stop potential crack flanking phenomena. Per Reference 20, VY has maintained a relatively high concentration of noble metals in the vessel to date. However, continued monitoring (which is currently scheduled) is needed to optimize planning for re-injection.

In summary, this TE recommends deferring HWC startup until November 2003, based on the following rationale:

1. Per References 7 and 23, VY will not be able to determine the fuel cladding root cause failure in the short term. Therefore, continued operation without hydrogen injection (until November 2003) will demonstrate acceptable fuel performance. This will additionally provide a contingent outage window to address any potential fuel cladding failures that might develop between November 2003 and April 2004 and will allow long term assessment of HWC performance relative to extended power up-rate. HWC is considered a short-term fuel risk in that copper metal can plate-out on fuel during injection. However, for the long term HWC will reduce the copper concentration in the moderator, and therefore tend to reduce copper plate-out on the fuel and the risk of CILC induced failures. Reactor Engineering, as part of their normal work scope, will continue to monitor industry fuel performance.
2. Based on Code Programs health report (Reference 16) and current noble metals protection, the theoretical risk of vessel internal crack growth is currently deemed less than the observed risk for fuel cladding failures that might develop and impact reactor operation. The November 2003 HWC startup date will optimize the overall risk between vessel internals and fuel health.
3. Based on HWC startup testing, noble metals and low copper concentration are currently providing a good interim method for controlling IGSCC, and meeting Reference 3B technical assessment requirements. Chemistry plans to monitor MMS coupons for noble metals reduction and provide timely recommendations to either inject HWC or perform noble metals re-injection so that the potential for crack flanking can be minimized.

Technical Evaluation No. 2003-023

<u>Assumptions/Open Items</u> *(List any assumptions used in the TE and provide a basis for each. List any open items requiring additional action prior to closure of the TE.)*

Plant and Engineering departments completed the above assessments based on information available on 03/17/03. Although there are no particular assumptions or open items that require tracking per se, the above evaluation should be periodically reviewed over the next year, to ensure that new information does not change the conclusions made.

<u>Material Requirements/Implementation Instructions</u> *(List any identified specifications for equipment, materials, or services needed to implement the recommendations of the TE. Specify any special implementation instructions or cautions, such as field testing requirements or system interface requirements during implementation.)*

Reference 13 provides the material requirements and references the implementation instructions that need to be completed before the HWC system can be restarted.

<u>Recommendations</u> *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

Based on the above analysis, this TE recommends deferral of HWC system startup until at least November 2003. To accomplish this task, this TE recommends the following actions:

1. Per Reference 3B requirements, include this TE as part of the BWRVIP program
   Responsible Department – Code Programs
   AP 0028 UND-2003-174_01 Due Date – 06/15/03

2. Ensure that the HWC startup schedule (e.g. Reference 13) is included in the 13-week schedule.
   Responsible Department – Work Management
   AP 0028 UND-2003-174_02 Due Date – 6/15/03

3. In light of new information that may emerge (e.g. root cause for fuel cladding failures, vessel chemistry change, new protection methods, etc.), re-examine this TE in 6 months, and:
   - confirm that starting HWC in November 2003 is still acceptable
   - provide additional documented justification if further delay in system startup is deemed necessary,

   Responsible Department – Technical Services Manager
   AP 0028 UND-2003-174_03 Due Date – 9/15/03

4. Provide recommendation for re-injecting noble metals based on available plant data (e.g. ECP millivolt readings, MMS coupon monitoring, vessel chemistry, etc.). Note that an acceptable initial response may include a status, with a follow-up commitment for future assessment.
   Responsible Department – Chemistry Superintendent
   AP 0028 UND-2003-174_04 Due Date – 06/15/04

5. Based on HWC re-start performance, project impact of site boundary dose when extended power up-rate is implemented.
   Responsible Department – EPU Project
   AP 0028 UND-2003-174_05 Due Date – 06/15/04

Technical Evaluation No. 2003-023

6. Determine strategy to achieve NRC goal to have two mitigation methods to combat IGSCC for Class 1 similar-metal piping welds (Category B-J) in light of the decision to delay hydrogen water chemistry (HWC).
   Responsible Department – Code Programs
   AP 0028 UND-2002-183_02 Due Date – 03/25/04

7. Resolve INPO inspection items on preparation and contingencies for hydrogen injection
   Responsible Departments – Code Programs
   AP 0028 UND-2002-074_12 Due Date – 11/01/03

Approvals/Closeout *(Print name and provide signature/date.)*

H.M. Metell H. M. Metell 6/24/03 CE (TE initiated 03/03)

P.B. Perez / 6/24/03 Independent Reviewer

D.C. Girroir / 6/26/03 Code Programs Supervisor

D.J. Mannai / 6/26/03 Reactor Engineering Superintendent

Sam Wender / 6/30/03 Chemistry Superintendent

/ 6-25-03 System Engineering Manager

Callaghan / 7/7/03 Design Engineering Manager

Michael / 6/26/03 Technical Services Manager

(Signature) (Date)

Closeout *(All actions that were recommended by the TE and accepted by management have been initiated and any identified open items have been dispositioned.)*

H. M. Metell / 7/8/03 CE
(Signature) (Date)

Attachments *(Provide a list of all forms, document markups, etc. provided as part of the TE package.)* – Please see the Technical Evaluation Data Base which lists all references used, and notes which references are attached.

Technical Evaluation No. 2003-023

TECHNICAL EVALUATION DATABASE INPUT

TE No.: 2003-023

TE Title: Technical Assessment For Delaying Hydrogen Injection Into The Reactor Core

Keywords: hydrogen, license renewal, IGSCC mitigation, offsite dose, BWRVIP

Design Input Documents - The following documents provide design input to this TE.

#	Document Title (including Rev. No. and Date, if applicable)
	None

Design Output Documents - The following documents are impacted by this TE.

#	Document Title
	None

General References

#	Reference Title (including Rev. No. and Date, if applicable)
1	VYDC 2000-006 " Installation of the Hydrogen Water Chemistry System,"
2	VYDC 2000-021 "Deposition of Noble Metals in the RCA,"
3	Boiling Water Reactor Vessel and Internals Project (BWRVIP), Reports 1 through 111. References 3A to 3I provide the key information for this evaluation.
3A	"BWR Water Chemistry Guidelines - 2000 Revision", BWRVIP-79, EPRI Report TR-103515-R2, March 02, (applicable pages attached).
3B	"BWRVIP-94: BWR Vessel and Internals Project, Program Implementation Guide", Appendix A, EPRI Technical Report 1006288, August 2001, (applicable pages attached).
3C	"BWR Vessel and Internals Project, Noble Metal Chemical Application (NMCA) Materials Surveillance Program at Duane Arnold Energy Center: Second Surveillance Report (1997-1998) (BWRVIP-68)", EPRI Report TR-112869, June 1999.
3D	"BWRVIP-92: BWR Vessel and Internals Project, NMCA Experience Report and Applications Guidelines", EPRI Technical Report 1003022, September 2001
3E	"Post NMCA Fuel Surveillance Program at the Duane Arnold Energy Center; RFO 15 Fuel Surveillance Program," BWRVIP-69
3F	"Post NMCA Fuel Surveillance Program at the Duane Arnold Energy Center; RFO 16 Fuel Surveillance Program," BWRVIP-82
3G	"Post NMCA Fuel Surveillance Program at Peach Bottom 2 After One Cycle EOC13" BWRVIP-93
3H	"BWR Vessel and Internals Project, Induction Heating Stress Improvement Effectiveness on Crack Growth in Operating Plants," BWRVIP-61
3I	BWR Vessel and Internals Project, "Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection," BWRVIP-62
	Vermont Yankee's explicit and implicit commitment to the guidance of the BWRVIP. (See next 4 references)
4A	BVY 97-123, dated September 30, 1997. "Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals" (applicable pages attached).
4B	Letter Carl Terry (Niagara Mohawk Power Company – BWRVIP Executive Chairman) to Brian Sheron (USNRC)

VYAPF 6045.05
AP 6045 Original
Page 1 of 2
LPC #1

Technical Evaluation No. 2003-023

	dated April 25, 1997, "BWR Utility Commitments to the BWRVIP" (applicable pages attached).
4C	Letter Carl Terry (Niagara Mohawk Power Company – BWRVIP Executive Chairman) to Brian Sheron (USNRC) dated October 30, 1997. "BWR Utility Commitments to the BWRVIP"
4D	Letter Carl Terry (Niagara Mohawk Power Company – BWRVIP Executive Chairman) to Brian Sheron (USNRC). dated April 16, 2002. "Utility Implementation of BWRVIP Products"
5	ER-2002-0566 03, VY Root Cause Failure Assessment
6	Root Cause Team Collaborators consisted of: • GENE • GNF • EPRI • Aquarius Services • Reactor Engineering, Browns Ferry TVA • Design Engineering • Reactor Engineering • Independent consultants Note: Database entry is not necessary for this reference.
7	GE Proprietary, Report, GE-NE-0000-0006-0152-01 Rev 1 "Vermont Yankee Results of Cycle 22 Mid-Cycle Outage Fuel Deposit Sampling Campaign," November 2002, (Attached)
8	GNF Letter, Kingston to Metell, "GNF Recommendation for Hydrogen Injection for VY Cycle 22," March 28, 2002, (Attached)
9	GE Proprietary, Draft Letter DRF P34-00055-00, Sundberg to Metell, "Resumption of HWC at Vermont Yankee," October 21, 2002, (Attached)
10	Bill Russell, Engineering/Management Consultant (Former NRC Director)
11	Vermont Regulation - Part 5, Chapter 3 Radiological Health, Subchapter 1, Radiation Protection (Basis for 20 mr/year site boundary limit)
12	VYC-2194, "Vermont Yankee Site Boundary Direct Dose Determination Methodology"
13	TE-2002-049, "Hydrogen Water Chemistry (HWC) System Startup Considerations,"
14A	Installation and Test/ Special Test Procedure 2000-006.001, "Hydrogen Water Chemistry System Start-up, Tuning, and Benchmarking Procedure,"
14B	Mitigation Monitoring Panel - ECP Probe Data, September through November 2001 (Attached)
15	BWR Vessel and Internals Project, "Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection," BWRVIP-62, Figure 1-10 (Applicable portions attached).
16	Reactor Vessel Health Report (available January 23, 2003)
17	Reactor Vessel and FW Ion Trending – Chemistry Data from 2001 through 2002 (Attached)
18	Application/Control Number 09/844,163, Claim Rejection - Lindstrom et al, Patent 3, 067,121 (Applicable page attached)
19	Goldstein to Distribution, "Trip Report for BWRVIP Mitigation Committee Meeting 10/1-2/2002," October 23, 2002. (Attached)
20	GE Proprietary, "Vermont Yankee Durability Monitor Coupon Analysis," Report 5, January 2003,
21	BWRVIP Proprietary, Wilson/Pathania to Distribution, "Draft Addendum to BWRVIP Assessment of NMCA Fuel Issues," January 24, 2003, (Applicable pages attached).
22	Larsen to Girroir, System Engineering, "Strategy To Meet NRC's Desire To Have Two Mitigation Methods For Each Weld For Implementation of Code Case N-560, August 20, 2002
23	John Schardt, "BF2/VY Root Cause Investigation Report," March 17, 2003 (GNF Proprietary)
24	VYDC-2002-003, Cycle 22B Mid Cycle Reload, May 2002

Data Entered into Database Amy Graves 8/11/03 Entry Verified W. Metell 08/12/03

VYAPF 6045.05
AP 6045 Original
Page 2 of 2
LPC #1

TECHNICAL EVALUATION REVIEW

TE #: 2003-023 Required Date: 30 April 03 Reviewer Assigned: P. Pérez

Title: BWRVIP-79 Technical Justification for Delaying Hydrogen Injection Into the Reactor

Comments:	Resolution:
(1) Please consider implementing the editorial changes discussed.	1) Completed
(2) Please add VYDC-2002-003 as a reference for the Cycle22 core design discussed in TE Item (1) on Page 1	2) Completed
(3) TE Item (3) on Page 2 provides a reference for the -40 mV ECP. Item (6) introduces +55mV. Please provide a reference for this ECP.	3) I referenced the Mitigation Monitoring Panel and added the chart as an attached reference
(4) The Item (6)(a) on Page 2 discusses Cu and Zn deposits on fuel. Does the discussion consider the VY failed fuel lift-off scrape chemical analysis?	4) I edited to include Reference 7 which described the scraping analysis in detail.
(5) The Item 6(b) discusses potential for Zn injection. Should there be a commitment to monitor and make a recommendation. Please note that Commitment No.4 does not directly address Zn injection.	5) Resolved by discussion with independent reviewer.
(6) Item 6(b) discusses Fe_2O_3 impacting performance such as valve seats. Please provide an example if one exists (e.g. MSIVs).	6) Except during shutdown, MSIVs see primarily steam versus water. The oxide would primarily effect water valves as it should remain as a water slurry in the vessel. The major impact might be a RWCU valve; other areas would tend to be highly scoured. On that basis, I would prefer to leave as a generic description.

Pedro B. Perez, 4/28/03
Reviewer Signature Date
PEDRO B. PEREZ

A. M. Metell
A M Metell , 5/1/03
DCE Signature Date

Pedro B. Perez , 5/1/03
PEDRO B. PEREZ

Notes and Requirements:

1. Validate design input appropriateness relative to your area of expertise.
2. Verify Department Procedures, Program inputs, and output documents are appropriately addressed.
3. Meetings or discussions to resolve questions and comments are encouraged.
4. Resolution by telecon is acceptable and should be noted as such.
5. Make comments specific, and avoid generalizations and questions.
6. If no comments indicate "None"
7. Request Management assistance if resolution can not be achieved.
8. Return all comments to the CE by required date or request an extension.

VYAPF 6045.04 Pg 1 of 1

Technical Justification No. 2003-03

Title: Justification to Perform Less Than 5% of CRD Guide Tube Welds Within the First Six-Year Interval

Technical justification is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines.

BWRVIP Requirement *(Give BWRVIP document and Section reference with a restatement of the requirement.)*

BWRVIP-47-A, Table 3.2-1, requires 10% of the CRD guide tube population (specifically welds CRGT-2 and CRGT-3) to be inspected within a twelve year period and 5% within the first six years. The first six years is defined in the Reactor Internals Inspection Bases Document as the first six years following issuance of BWRVIP-47, which was published in December 1997. That first six years included RFO 20 (1998), RFO 21 (1999), RFO 22 (2001), and RFO 23 (2002). There are 89 CRD guide tube assemblies at Vermont Yankee. A sample of 10% would be nine guide tubes and a sample of 5% would be five guide tubes (rounded up to the next integer). CRGT-2 is the guide tube body-to-sleeve weld and CRGT-3 is the guide tube base-to-body weld.

Vermont Yankee Deviation *(Record how Vermont Yankee deviates or deviated from the BWRVIP requirement.)*

Only four CRD guide tube assemblies were inspected during the first six-year period. The inspections were not begun until RFO 22 (2001) when four guide tube assemblies were inspected. No guide tubes were accessible during RFO 23 (2002) because no control blades were changed during that outage.

BWRVIP-47-A also requires a VT-3 inspection of two other locations in the guide tube assembly. These are CRGT-1, the guide tube sleeve-to-alignment lug weld and FS/GT-ARPIN-1, the guide tube and fuel support alignment pin-to-core plate weld and the pin itself. The minimum sample of these locations was completed during the first six-year period [get history from Tom Stetson]. The VT-3 inspections were completed during the course of the orificed fuel support reinstallation/realignment procedure, OP 1111.

Justification ***(Provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the BWRVIP guidelines. The justification shall be supported by calculations when warranted. Clearly identify all available information and resources, which allow the deviation to be acceptable. Clearly identify the impact that the deviation will have on meeting the intent of the guideline.)***

The 5% sample inspection of CRD guide tubes would normally be completed during the course of control blade change-outs over a six-year period. Blade change-out requires orificed fuel support reinstallation and realignment, thus allowing access to the interior of the CRD guide tube. Typically, there are between three and ten blade change-outs each outage, so it is reasonable to expect that there would be at least nine blade change-outs during any twelve-year period and at least five blade change-outs during any six-year period. However, for the two outages within the first six-year period after inspections were commenced, only four guide tubes became available.

The only reason why exams of these components were not performed during either RFO 20 (1998) – when four blades were changed out – or during RFO 21 (1999) – when nine blades were changed out – was that it was fully expected that there would be at least five blades changed out during the following two outages. It should be noted that Vermont Yankee was one of the first plants to perform examination of these welds.

BWRVIP-47-A, Section 3.2.5 states, "The BWRVIP has determined that removing or dismantling of internal components for the purpose of performing inspections is not warranted to assure safe operation." The requirements of BWRVIP-47 were originally designed to allow inspections to be performed during the normal course of plant maintenance. In that way, the CRGT-2 and CRGT-3 welds could be inspected when control blades are changed out.

Inspecting one additional guide tube during RFO 23 (2002) to attain the 5% threshold would have required vacating an additional fuel cell (more fuel moves) and an added three hours for disassembly and reassembly (not counting the inspection time). This hardship is not justified in terms of safety in order to raise the inspection sample from 4.5% to 5%.

Inspections of the eight welds in the four guide tubes inspected during RFO 22 (2001) did not reveal any flaws.

The significance of a sample inspection reduction from 5.0% to 4.5% will be evaluated. If it is desired to find one flaw within a sample of welds, then a larger finite number of flaws must exist in the population to have a certain probability (akin to a level of confidence) of finding at least one flaw. This is implicit from the fact that the BWRVIP allows a sample inspection plan. For the sake of argument, assume that a level of confidence of 90% is required. It is assumed for simplicity that the probability of detecting (POD) any one flaw is 100%.

For a sample of 9 welds within 178 welds (5.0%), there must be 40 flaws in the total population in order to assure that there is about a 90% chance of detecting at least one of the flaws within the sample.

$$138/178 \times 137/177 \times 136/176 \times 135/175 \times 134/174 \times 133/173 \times 132/172 \times 131/171 \times 130/170 = 0.095$$

(90.5% confidence level)

The actual probability of detecting at least one of the 40 flaws in this example is 90.5%.

For a sample of 8 welds within 178 welds (4.5% – the actual sample examined), the probability of detecting at least one of the 40 flaws only drops to 87.5% (from 90.5%).

$$138/178 \times 137/177 \times 136/176 \times 135/175 \times 134/174 \times 133/173 \times 132/172 \times 131/171 = 0.125$$

(87.5% confidence level)

Clearly, this small incremental decrease in the confidence level is statistically insignificant and within acceptable limits given that the probability of detecting at least one flaw drops by only 3 percent.

The requirement for inspecting 10% of the CRD guide tubes over the first twelve years will be met.

<u>Duration of Technical Justification</u> *(State how long the deviation will be in effect.)*

This deviation will expire following RFO 24 (2004), because by that time the minimum 5% sample inspection will be completed.

Assumptions/Open Items *(List any assumptions used in the TE and provide a basis for each. List any open items requiring additional action prior to closure of the TE.)*

It is assumed that at least one blade will need to be changed out in RFO 24 (2004) and that at least four blades will be changed out between RFO 26 (2007) and RFO 27 (2008).

The industry implicitly accepts a certain number of flaws in a population of welds by endorsing the use of a sample inspection program.

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

Based on the above analysis, this Technical Justification recommends the following action:

1. Examine welds CRGT-2 and CRGT-3 in one guide tube during RFO 24 (2004) to complete the original 5% sample.
   Responsible Department – Code Programs
   Due Date – May 31, 2004

Approvals *(Print name and provide signature/date. A thorough review shall include and consider input from a wide variety of sources.)*

[signature] / 8/18/03 Reactor Internals Management Program Coordinator
N/A / N/A Mechanical/Structural Design (if applicable)
N/A / N/A Chemistry (if applicable)
[signature] / 8/13/03 Other Cross-Discipline or Independent Review (if applicable)
Dennis [signature] / 8/8/03 Code Programs Manager
(signature) (date)

Closeout *(All actions that were recommended by the Technical Justification and accepted by management have been initiated and any identified open items have been dispositioned.)*

_______________ / _______________ Reactor Internals Management Program Coordinator
(signature) (date)

Input Documents and other References – The following documents provide input to this Technical Justification.

#	Document Title (including Rev. No. and Date, if applicable)
1	BWRVIP-47, December 1997, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
2	BWRVIP-47-A, June 2002, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
3	PP 7027, Revision 1, "Reactor Vessel Internals Management Program"
4	Reactor Vessel Internals Components Basis For Inspection And Other Management Requirements, dated February 13, 2003

VYPPF 7027.01
PP 7027 Rev. 2
Page 3 of 4

Appendix C (Continued)

Design Output Documents – The following documents are impacted by this TE.

#	Document Title
N/A	None

Technical Justification No. 2003-04

Title: Continued Operation without a Feedwater Zinc Injection System

Technical justification is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines.

BWRVIP Requirement *(Give BWRVIP document and Section reference with a restatement of the requirement.)*

BWRVIP-79, Section 3.2.3.3 discusses the benefit of zinc in order to reduce cobalt 60 isotope buildup for dose considerations. It states, "While it is believed that a reasonably optimized value is 5-10 ppb zinc in the reactor water so that the benefits can be maximized while still remaining comfortably within the historical experience band, each utility must perform their own cost/benefit evaluation to discern what concentration is optimum for them."

BWRVIP-107, Section 5.1 also states, "Adjust feedwater zinc injection rate to result in a steady reactor water level of 5 to 10 ppb...and maintain this level during post NMCA operation. (Note: This recommendation may not be consistent with fuel vendor recommendations."

Vermont Yankee Deviation *(Record how Vermont Yankee deviates or deviated from the BWRVIP requirement.)*

Vermont Yankee – not possessing a zinc injection system – does not have a way to adjust zinc levels, although VY has maintained a reactor coolant zinc concentration around that general range as a result of having an admiralty condenser.

Justification *(Provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the BWRVIP guidelines. The justification shall be supported by calculations when warranted. Clearly identify all available information and resources, which allow the deviation to be acceptable. Clearly identify the impact that the deviation will have on meeting the intent of the guideline.)*

The presence of zinc in the reactor coolant will minimize the incorporation of Co-60 into oxide films on reactor vessel internals and associated piping. Zinc also stabilizes the existing fuel deposits, which reduces the release rate of corrosion products such as Co-60 into the reactor coolant. The original information came from comparing dose rates at plants that had admiralty condensers with filter demineralizers to those with stainless steel ones. Admiralty condensers have tubes that are approximately 21% zinc and 78% copper. Many utilities replaced their admiralty condensers or installed deep bed demineralizers based on copper concerns associated with Crud Induced Localized Corrosion. This factor prompted them to initiate zinc injection utilizing depleted zinc to reduce dose rates. Initially, this was done based on the substantial increase in reactor internal dose rates following initiation of Hydrogen Water Chemistry (HWC).

Three GE BWRs located in the US reported unexpectedly high release rates of activated corrosion products to the reactor coolant during cool-down prior to their refueling outages. Two of these plants received mid-cycle Noble Metal Coating (NMC) applications. One of these plants had no zinc injection while the others were maintaining zinc concentration in the range of 2-3 ppb, which is slightly lower than Vermont Yankee. Those plants that maintained zinc concentrations in the 5-10 ppb range saw a modest increase in one case and a reduction in dose rates in the others. In addition, these three plants did not experience inordinate releases of corrosion products during the refueling outage shutdowns. From this information, GE infers that the higher reactor water zinc concentrations more effectively stabilized the fuel deposits and minimized the release of activated corrosion products to the coolant.

In GE SIL No. 631, the following statement is made: "Maintain the reactor water zinc concentration in the 5-10 ppb range. This is applicable except where it has been clearly demonstrated that there have been no significant drywell dose rate issues for a complete cycle following the application while maintaining a lower zinc concentration in the reactor water." "At both plants that experienced higher than expected dose rates, the nominal zinc concentration in the reactor water was 2-3 ppb for the majority of the cycle following the NMCA."

BWRVIP-107 makes several recommendations relating to the injection of depleted zinc (DZO). It states, "Maintain reactor water zinc at 5-10 ppb and the 2x10-5 micro-Ci/ppb Co-60(s)/Zn(s) ratio. This recommendation is solely for the purpose of reducing out of core dose rates. It requires that a utility have a zinc injection system that utilizes depleted zinc.

The installation of a zinc injection system would cost approximately 1,000,000 dollars and require an expenditure of 150,000 dollars annually for depleted zinc. There are currently no plans to install such a system as long as we have an admiralty condenser. Engineering is currently evaluating the replacement of the condenser with a titanium condenser. They are now aware that such a change would require the implementation of zinc injection. The earliest probability of installing a new condenser is in 2007 based on current priorities.

Vermont Yankee has maintained a reactor coolant zinc concentration in the range of 2.5 –10 ppb over its entire operating history as a result of having an admiralty condenser. Since the replacement of our recirculation piping in 1986 with Hitachi 316 stainless, electro-polished pipe, we have maintained very low recirc pipe dose rates of approximately 75 –120 mR/hr. In the early 1980s we began a cobalt reduction program that included such activities as replacing the stellite in the feedwater regulation valves. For 2002, reactor coolant zinc concentration ranged from 3.7 to 11.7 ppb with a mean of 6.4 ppb and an average of 7.0 ppb. Thus, we generally met the requirement to keep RV zinc concentration in the range of 5-10 ppb.

However, we have minimal control over the concentration of zinc in the feedwater, as it is a function of condensate temperature and condensate demineralizer efficiency.

Natural zinc from the admiralty condenser ultimately becomes activated in the reactor coolant to Zn-65. VY has the highest reactor coolant Zn-65 concentration in the industry. However, this does not have a significant impact on recirc pipe dose rates. Those plants that inject depleted zinc to not have to worry about the zinc activation problem.

Vermont Yankee is unique among BWRs. This means that industry data relative to out-of-core dose rates and RV zinc concentration may not apply. The following set of conditions does not exist at any other BWR:

1. Filter Demineralizer plant without supporting deep beds
2. Admiralty Condenser
3. Low feedwater iron
4. NMCA without hydrogen addition
5. Recirc Pipe that is Hitachi 316 electro-polished stainless steel
6. 15 years of BRAC point dose rates in the 80-125 mR/hr range
7. No chemical decontaminations during the past 15 years

8. Very low ManRem outage exposure (last outage was a record for a BWR at ~76 ManRem
9. Highest feedwater copper levels in the industry
10. Highest reactor vessel Zn-65 levels in the industry.

Given our recent history, it appears that the current plant chemistry is adequate in keeping out of core dose rates within acceptable limits. The current plant chemistry does not support the expense required to initiate DZO. In addition, recent industry fuel problems have raised the question, "How much is too much zinc in the reactor coolant?" Some utilities are already reducing the amount of zinc that is injected to alleviate the crud buildup on the fuel.

Duration of Technical Justification *(State how long the deviation will be in effect.)*

This deviation will remain in effect until the main condenser tubing is replaced with an alternate material.

Assumptions/Open Items *(List any assumptions used in the TE and provide a basis for each. List any open items requiring additional action prior to closure of the TE.)*

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

Based on the above analysis, this Technical Justification recommends the following actions:

1. Continue plant operations as in the past relying on the natural zinc from the condenser to help control out-of-core dose rates. Plan on installing and initiating zinc injection in conjunction with a condenser replacement.

   Responsible Department – Systems Engineering
   Due Date – Not Applicable

Approvals *(Print name and provide signature/date. A thorough review shall include and consider input from a wide variety of sources.)*

(signature)	(date)	
[signature]	/ 8/18/03	Reactor Internals Management Program Coordinator
N/A	/ N/A	Mechanical/Structural Design (if applicable)
[signature]	/ 08/18/03	Chemistry (if applicable)
[signature]	/ 8/18/03	Other Cross-Discipline or Independent Review (if applicable)
[signature]	/ 8/18/03	Code Programs Manager

VYPPF 7027.01
PP 7027 Rev. 2
Page 3 of 4

Appendix C (Continued)

Closeout *(All actions that were recommended by the Technical Justification and accepted by management have been initiated and any identified open items have been dispositioned.)*

N/A / N/A Reactor Internals Management Program Coordinator
(signature) (date)

Input Documents and other References – The following documents provide input to this Technical Justification.

#	Document Title (including Rev. No. and Date, if applicable)
1	BWRVIP-79, dated February 2000, "EPRI BWR Water Chemistry Guidelines - 2000 Revision"
2	BWRVIP-107, dated November 2002, "Dose Reduction Through Optimizing Chemistry Using Depleted Zinc with Noble Metal Chemical Addition"
3	PP 7027, Reactor Vessel Internals Management Program
4	GE SIL 631, Revision 1, dated September 4, 2001, "Zinc Injection Following NobleChem Application"

Design Output Documents – The following documents are impacted by this TE.

#	Document Title
	None

VYPPF 7027.01
PP 7027 Rev. 2
Page 4 of 4

Technical Justification No. 2003-05 12/17/03

Title: Feedwater Copper Concentrations above Recommended Limits

Technical justification is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines.

BWRVIP Requirement *(Give BWRVIP document and Section reference with a restatement of the requirement.)*

BWRVIP-79, Table 4-6, Note c states, "An engineering evaluation should be performed before application of this value [>0.20 ppb feedwater total copper] at plants with copper alloy condenser tubes and powdered filter/demineralizers, since it may not be achievable without costly plant modifications. In these circumstances, a limit above 0.2 ppb may be justifiable based on previous performance and core design considerations."

Vermont Yankee Deviation *(Record how Vermont Yankee deviates or deviated from the BWRVIP requirement.)*

Feedwater Copper has been consistently above the current (2000 revision) EPRI guideline value of 0.2 ppb.

Justification *(Provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the BWRVIP guidelines. The justification shall be supported by calculations when warranted. Clearly identify all available information and resources, which allow the deviation to be acceptable. Clearly identify the impact that the deviation will have on meeting the intent of the guideline.)*

BACKGROUND:

Elevated feedwater copper levels for plants with admiralty condensers and filter demineralizers have been an industry issue for over a decade. Most BWRs have resolved the issue by replacing their condensers or adding deep bed demineralizers down stream of their condensate polishers. There are currently only two BWRs that have not taken the previously stated corrective actions. These are Columbia and Vermont Yankee. Columbia Station has developed a technical justification for maintaining a feedwater copper limit higher that recommended in the EPRI Guidelines but in accordance with their fuel warranty of 0.5 ppb.

Filter demineralizers are at approximately 90% efficient for removal of soluble species due to the very short residence time on the thin ion exchange resin layer on the precoat. The challenge is to maintain the current efficiency under all phases of plant operation. When tighter precoats are utilized they take out more of the insoluble species and seem to increase the ion exchange efficiency. The age of the elements is also a factor in the removal of insolubles. When low-cross-linked resins are used for crud control, they shed some sulfate. Their use is therefore limited based on the increase in reactor coolant sulfates. In the effort to maintain or increase the 90% efficiency we are also removing iron and zinc. Therefore, there is a balance between the achievable feedwater copper and the desired feedwater iron and reactor coolant zinc and sulfate concentrations.

The 2000 revision of the EPRI Guidelines indicates that the desired feedwater iron range is 1ppb +/- 0.5 ppb and that the desired reactor coolant range for zinc is 5-10 ppb. However, it further states that "experience has demonstrated that benefits are being achieved with RV zinc concentrations as low as 3 ppb. With regard to feedwater iron it states, "That between 0.3 and 0.5 ppb long term experience across multiple units is lacking"; "adverse consequences have been reported from long term operation near 0.1 ppb in the U.S., Sweden and Japan.

When the EPRI Water Chemistry Guidelines were issued in 1986, the recommended feedwater copper limit was 0.5 ppb. This limit was also in the 1996 revision to the Guidelines with a note that stated that this is a common value listed in fuel warranties. Vermont Yankee adopted the Guidelines and created a water chemistry policy, VYP-131. Plant management took an exception to the guideline's copper limit recommendation. Based on discussions with GE, the 1.0 ppb limit was maintained in accordance with the GE Fuel Contract. Vermont Yankee may not be able to meet the current Guideline value of 0.2 ppb copper under most operating conditions without making expensive plant modifications. However, given that copper has been implicated as a potential contributor to the Cycle 22 fuel failures, it is prudent to take actions to reduce feedwater copper to as low as reasonably achievable without compromising other parameters such as reactor coolant zinc, sulfate and feedwater iron.

Vermont Yankee has had recurrent feedwater copper excursions greater than 1.0 ppb for more than a decade. The most significant events occurred during 1988 and 1995 when the Fuel Contract Continuous limit of 1.0 ppb was exceeded for more than 14 days. During 1988 feedwater copper was >2.0 ppb for 6 weeks. There were some feedwater copper values >1.0 ppb in 1999 and in 2001, all of which occurred during the summer months. Feedwater copper is most difficult to control during summer months when condensate temperatures may reach 138 degrees F as a result of Close/Hybrid cycle operation.

Vermont Yankee has maintained feedwater copper at approximately 0.27 ppb (average) for the first 6 months of Cycle 23. There were approximately 30 days during the period when feedwater copper was at or below 0.2 ppb. During the summer months, feedwater copper is much more difficult to control. Achieving values below 0.2 ppb may not be possible during the warmest months between July and September without installing deep bed demineralizers or changing out the admiralty condenser. The current data indicates that the achievable range for feedwater copper is <0.2 ppb to 0.5 ppb with the yearly average being <0.3 ppb.

High feedwater copper ultimately results in high reactor coolant copper in the range of 5-10 ppb on average. Approximately 90% of the metals that are in the coolant plate out on core surfaces including the fuel. The increased crud loading on the fuel can create a problem known as CILC (Crud Induced Localized Corrosion) that may result in a fuel failure. Vermont Yankee did not have any clearly identified CILC related fuel failures during the 80s or 90s.

Admiralty condensers provide several benefits. Besides being resistant to corrosion, they provide a natural source of zinc. Zinc has been shown to be an important factor in reducing out of core dose rates. Plants without admiralty condensers have to inject zinc, whereas Vermont Yankee can maintain a reactor coolant zinc concentration of 3-6 ppb without needing to perform zinc injection. Having had natural zinc for its entire operating history has helped VY to maintain dose rates/personal exposure very low compared to the rest of the industry.

CORRECTIVE ACTIONS:

As a result of the feedwater copper excursions of 1988, 1995 and 1999, a series of corrective actions were put in place to minimize/prevent feedwater copper excursions. These corrective actions significantly reduced the number of excursions >1.0 ppb. In fact, there were none in 2000 and only one in 2001. The following are some of the corrective actions taken over the years:

1. Reduction in the source term. The turbine casing, a source of copper was replaced in 1994. Silicon brass nuts in the feedwater heaters were replaced by stainless steel ones. All of the condensate pumps' first three stages were changed from brass to stainless steel.
2. Placed the Body Feed system back into service in order to increase copper ion removal efficiency of the demineralizers by filling in gaps in the precoats.
3. Purchased high efficiency crud reduction resins for use during the warmer months
4. Feedwater metals analyzed more frequently during summer months.
5. A senior Graver Engineer (Charlie Mosser) was contracted to review VY's condensate polishing process and make appropriate recommendations for process improvement. Recommendations from his final report were implemented in 2001 and 2002 for improvements to the backwash and precoating process.
6. Following participation in an INPO assist visit to Quad Cities, an action plan was developed and implemented to change all of the elements in the condensate demineralizers from 2 inch diameter to 2.25 inch diameter. This increased the element surface area of each vessel by 115 sq ft. All vessels currently have these new elements.

CORRECTIVE ACTIONS IN THE COURSE OF DEVELOPMENT:

A revised action plan for feedwater copper control has been developed and will be implemented beginning in May of 2003. Chemistry will be optimizing filter demineralization processes and evaluating the data for process improvement. An assessment of the 2003 feedwater copper control strategy will be performed at the end of 2003 and the plan will be revised accordingly. Level 1 LOCA UND-2003-326_02 was created to review the action plan implementation status. Excerpts from the plan are as follows:

1. Starting April 2003, all condemins were precoated with a sandwich of P-202H followed by Meridian 2000. Once the vessel is placed in service it will be Body Fed with an additional three bags of Ecodex P-202H. Precoats of this type will continue until October 15th or such time that analytical data indicates that a change in precoat formulation is warranted. NOTE: Precoats of this type may increase RV sulfate to 2.0 ppb for a short duration and therefore more than one vessel should not be done within a three-day interval. Other copper removal resins may be used such as Purolite CG-125-H. Starting in June, feedwater copper control evaluation was begun with all 5 vessels having the same type of precoat.
2. Began Metals analysis for individual demineralizers in April 2003 and will continue on a weekly basis until November of 2003, at which time this process will be evaluated for continuation through the winter.
3. In May 2003, revised the existing technical justification for feedwater copper to indicate a formal approach to copper control that is ALARA providing administrative goals for summer and winter conditions.
4. Reviewed the approach to copper control at Columbia station and their technical justification for maintaining feedwater below 0.5 ppb, which is the current GE Fuel Warranty value.
5. Avoided condensate demineralizer system operation with 4 vessels in service as much as possible (other than for backwashes and precoats) during the period of May 15th until October 15th. This means that element replacement should not be done during this time. No condemin element change outs were scheduled during this interval.
6. Carefully review the copper data from the individual demineralizers and take appropriate actions to reduce effluent copper concentration, (Body Feeds or new Precoats). Establish an action level for demineralizer effluent copper based on plant operating conditions.

7. In order to determine the optimum runtime for a precoat and an achievable feedwater copper limit, the following data will need to be evaluated:
   a. Precoat formulation and number of Body Feeds
   b. Condensate temperature (see item #8)
   c. Reactor Vessel zinc, copper and sulfate concentrations
   d. Feedwater zinc, iron and copper concentrations
   e. Individual demineralizer effluent copper and precoat run time
   f. Condensate demineralizer influent copper

   Data Trending was started in April 2003.
8. Monitor the condensate demineralizer influent temperature, (ERFIS computer point F0-76 (steam packing exhauster effluent). This is to be done from May 15th until October 15th and more often if the plant is on closed cycle for longer than 2 hours (chlorination). This point should be trended along with the upstream river temperature. Temperatures above 130 degrees F negatively impact copper control and need to be brought to the attention of Chemistry/Plant Management. This is ongoing.
9. Project the increased costs for ion exchange resins and waste disposal.
10. Collected feedwater metals on a daily (24 hour) basis during the period from April 21st to October 15th. During the cooler months three 48-hour samples and one 24-hour sample are utilized.
11. Monitor the deposition of copper in the reactor coolant in micrograms/second based on the weekly reactor coolant metals analysis. This is ongoing.
12. Benchmark other utilities to see who uses online IC or XRF for metals analysis. How accurate is the process and the cost of equipment, installation and cost of maintenance. Consider long-term modification for copper monitoring. This is ongoing.
13. Performed individual condemin metals analysis twice per week during the period of 7-15 to 9-15
14. Incorporate the critical elements of copper control into a plant procedure or other technical document.

NOTE

Items 7,8,12 and 14 were recommendations from Mr. Bill Russell, a consultant for senior plant management who routinely makes quarterly assessment visits to the plant site.

TECHNICAL JUSTIFICATION BASIS:

Table 4-6 in the EPRI 2000 Guidelines provides the following note: "An engineering evaluation should be performed before application of this value with copper alloy condenser tubes and powdered filter/demineralizers, since it may not be achievable without costly plant modifications. In these circumstances, a limit above 0.2 ppb may be justified based on previous performance and core design considerations." The Guidelines further state in section 3-38; "This peculiar phenomenon called CILC resulted in several cases of fuel failure from late 1978 up to the late 1980s, but since has been mitigated by using higher nodular corrosion resistant cladding..."

General Electric expressed a concern in 1996 that high reactor coolant copper levels would interfere with Hydrogen Water Chemistry (HWC) but at the same time indicated that the corrosion resistance of their fuel had been improved to resist the CILC phenomenon. Since that time they issued a report entitled "The Cu Club, Laboratory Test Results", that supports HWC as well as Noble Metal Coating

Application (NMCA) under reactor coolant levels as high as 15 ppb. Copper levels in the reactor coolant at Vermont Yankee currently range from approximately 5-8 ppb.

The River Bend fuel failure incident of 1999 was thoroughly evaluated and discussed at several EPRI meetings attended by the VY Plant Chemist. River Bend experienced fuel failures in 7 fuel assemblies that appeared to be related to copper. Although there was an elevated amount of copper in the fuel crud, the failure mechanism was more a result of heavy deposition of iron oxide-based crud. Two cond tivity excursions during the October 1997 refueling outage and the subsequent startup are the sus ted causes for a large influx of corrosion products early in the operating cycle. Their feedwater ir evels were around 3.7 ppb. This did not account for all iron deposits on the fuel inside the core and it as not clear where this extra iron came from. . At Vermont Yankee, feedwater iron is always maintained be the EPRI Guideline value of 5.0 ppb and is infrequently above 2.0 ppb. As a result, the tw n at River Bend are not expected here, even with a feedwater copper ased on a review of the EPRI Guidelines (section 3-40), this incident was deline value for feedwater copper being reduced from 0.5ppb to 0.2 ppb. In dis ... Gen... Electric, they have indicated that the justification for this lower limit is not based on the data.

ad a feedwater copper limit of 0.5 ppb. Their feedwater copper runs in the range e performed a technical justification for this limit that basically states that the pact fuel performance. While Vermont Yankee can maintain feedwater copper <0.5 pp ly average, it may not be able to achieve this limit on a day to day basis in the hottest months of the year.

The General Electric BF2/VY Root Cause Investigation Report dated 03/17/2003 did not determine a root cause for the 5 failures identified during Cycle 22. The report indicated that the extraordinary high levels of copper likely contributed to accelerating the corrosion process along with some unknown initiating event. Fuel examinations indicated relatively high copper deposits on Cycle 19, 20 and 21 fuel. All 5 of the failures were from the same tubing lot and failed in VY reload number 20. The data indicate that other reloads residing in the core are not exhibiting the accelerated corrosion. It was noted that VY leads the fleet in feedwater copper, but that it does not represent a change – VY has always had high feedwater copper, and has not had related fuel failure problems since the late 1970s. The root cause evaluation did not provide any recommendations for copper control or indications that the current fuel warranty value of 0.5 ppb for feedwater copper would be changed.

Following the VY fuel failures, the Reactor Engineering department contacted Aquarius Services Corp. (Al Strasser) and requested that he evaluate all of the data associated with the fuel failures. This included GE evaluations and material, two cycles of Chemistry Data and plant operating history. Fuel manufacturing data was also reviewed. Some conclusions and notes from his report are as follows:

a. Nodular corrosion should not occur on an in-process heat-treated cladding. Of the two causes, corrosion by high copper chemistry water is unlikely, since GE work in the past showed that this does not occur either in or ex-reactor. High Cu chemistry with NMC might induce nodular corrosion by the change in redox conditions at the cladding surface. The previously proposed poor in-process heat-treating control could be a second cause.
b. The continued evaluation of the fuel examination tapes confirm previous conclusions that there is a correlation between the level of corrosion observed, some of the cladding lot numbers and some of the local peaking factor histories of the rods.
c. The author concurs with GNF's conclusion that three cladding lot numbers behaved poorly.

d. A cursory comparison of fuel rod local peaking factor histories of rods from the same cladding lot indicates a reasonable correlation of power with corrosion control.
e. Based on GE information, there does not appear to be a correlation between Cu content and liftoff measurements, and there does not appear to be a correlation between linear power generation and liftoff either. This indicates a lack of correlation between copper content and corrosion.
f. The maximum concentration of copper at a discreet axial location was 1885 ug/cm^2 that occurred at the 31" elevation of Rod D* Bundle YJF493. One should note that this was a rod without a fuel defect.

CONCLUSIONS:

Based on this review of industry documents and operating experience, the guidance in the GE Fuel Warranty and the justification prepared by Columbia Station, it is concluded that copper may play a role in the fuel corrosion process but that further evaluations are required, especially as they relate to fuel duty. Feedwater copper levels >0.2 ppb but <0.5 ppb will not interfere with NMCA, HWC or our IGSCC mitigation program. However, it is the plant goal to maintain feedwater copper levels as low as reasonably achievable and to comply with the EPRI 2000 Water Chemistry Guideline value of 0.2 ppb.

Duration of Technical Justification *(State how long the deviation will be in effect.)*

This deviation will remain in effect indefinitely.

Assumptions/Open Items *(List any assumptions used in the TE and provide a basis for each. List any open items requiring additional action prior to closure of the TE.)*

None.

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

Based on the above analysis, this Technical Justification recommends the following actions:

1. Continue implementation of the copper reduction plan.
   Responsible Department – Chemistry

Appendix C (Continued)

Approvals *(Print name and provide signature/date. A thorough review shall include and consider input from a wide variety of sources.)*

Carl Larson / 12/17/03 Reactor Internals Management Program Coordinator

N/A / N/A Mechanical/Structural Design (if applicable)

Richard Gibson / 12/17/03 Chemistry (if applicable)

N/A / N/A Other Cross-Discipline or Independent Review (if applicable)

Dennis Girroir / 12/17/03 Code Programs Manager Dennis Girroir

(signature) (date)

Closeout *(All actions that were recommended by the Technical Justification and accepted by management have been initiated and any identified open items have been dispositioned.)*

Carl Larson / 12/17/03 Reactor Internals Management Program Coordinator
(signature) (date)

Input Documents and other References – The following documents provide input to this Technical Justification.

#	Document Title (including Rev. No. and Date, if applicable)
1	BWRVIP-79, February 2000, "BWR Water Chemistry Guidelines" (EPRI TR-103515R2)
2	GE Fuel Warranty 23A4715, Revision 4, August 24, 2000
3	Vermont Yankee Current Fuel Warranty Contract, Section H, October 13, 1993
4	"Impact of feedwater copper level greater that 0.2 ppb on fuel performance at Columbia"; Document given to Plant Chemist on July 13, 2001 by Brian Shaw, Fuels Design, Energy Northwest
5	Commitment BWRVIP-079_01, completed in 2000, "Perform Plant Specific Evaluation of Higher Copper Action Level than Stated in the EPRI 2000 BWR Water Chemistry Guidelines"
6	GE NEDC-32885, DRF B13-01880, 12-98 "Cu Club" Laboratory Test Results
7	ER 99-0826, Feedwater Copper Concentration >1.0 ppb for more than 96 hours
8	INPO SEN #204, Water Chemistry Induced Fuel Leaks (River Bend Station)
9	INPO Plant Evaluation for Vermont Yankee, March, 2001
10	GE fuel scraping results for Vermont Yankee, 2002-2003
11	BF2/VY Root Cause Investigation report, GNF, March 17,2003
12	Columbia Station Technical Evaluation for FW Copper above the EPRI Limit
13	"Preliminary Evaluation of Vermont Yankee Fuel Performance" Al Strasser, Aquarius Services Corp., June 7, 2002
14	Bill Russell quarterly chemistry/fuel failure evaluations, 2002-2003
15	GE Nuclear Energy, NEDC-32885, December 1998 "Cu Club" Laboratory Test Results

Technical Justification No. 2004-01

Title: Justification for Alternative Inspection of Core Plate Rim Hold-down Bolts

Technical justification is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines.

BWRVIP Requirement *(Give BWRVIP document and Section reference with a restatement of the requirement.)*

BWRVIP-25 (Reference 1), Table 3-2 states that for rim hold-down bolts without repair wedges, "Perform enhanced VT-1 from below the core plate (or UT from above core plate once the technique is developed) of 50% of the hold-down bolts. If cracking is detected, inspect the remaining 50% of bolts: Reinspection strategy to be based on plant-specific analyses to assure that critical numbers of bolts are intact to prevent lateral displacement of core plate."

In the April 28, 1999 NRC Safety Evaluation (Reference 3), the NRC states, "The staff believes that an initial baseline inspection should be comprehensive, and include all components that are practicable to inspect, based on tooling availability." However in the Final Safety Evaluation of December 19, 1999 (Ref. 4), the NRC consents with the BWRVIP previous response (October 6, 1999) that the inspection should be limited to components required for plant safe shutdown. The BWRVIP response (Reference 2) had stated, "If not, no inspection is required. This strategy is adequate to ensure plant safety. Performing a baseline inspection of locations which, if failed, have no affect on plant safety, would require an unnecessary increase in outage time in addition to the cost associated with developing and qualifying additional inspection tooling. Consequently, the BWRVIP does not agree with the NRC suggestion that all locations on the core plate be inspected in a comprehensive baseline inspection."

BWRVIP-25 report states that, "...as long as the critical number of bolts remain intact, lateral support for the core plate assembly is assured...Therefore, there is no safety consequences of failure at Location 8". (Location 8 in BWRVIP-25 discusses failure location for Aligner Pin and Socket to Rim Welds).

BWRVIP-25 also discusses acceptable alternatives to inspection, specifically involving plant-specific analysis or repairs and/or modifications.

Vermont Yankee Deviation *(Record how Vermont Yankee deviates or deviated from the BWRVIP requirement.)*

In view of the fact that no vendors have yet developed a delivery system for top-of-bolt UT techniques, and that EVT-1 inspection from below the core plate has accessibility limitations, VY will perform VT-3 inspection of 50% (15) of the top of the bolted connections every other refueling outage. Should access to the lower plenum become available, VY plans to augment core bolt inspections by performing a VT-3 inspection of accessible rim hold-down bolt bottom locking engagement and accessible aligner pin assemblies.

Justification *(Provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the BWRVIP guidelines. The justification shall be supported by calculations when warranted. Clearly identify all available information and resources, which allow the deviation to be acceptable. Clearly identify the impact that the deviation will have on meeting the intent of the guideline.)*

The core plate assembly provides lateral support for the fuel bundles, control rod guide tubes, and in-core instrumentation during seismic events, and provides vertical support for the peripheral fuel assemblies. The core plate assembly consists of a perforated plate reinforced by stiffener beams and supported on the perimeter by a circular rim. There are stabilizer beams (or cross ties) between the stiffener beams, which also provide support for in-core housing monitors. The VY core plate rim is bolted to a ledge on the core shroud with 30 preloaded, 2.0" diameter, 304 stainless steel rim hold-down bolts, which prevent horizontal and vertical movement. The core plate is positioned on the shroud ledge by four 2.5" diameter vertical aligner pins. The pin assembly engages sockets, which are welded to both the core plate and the core shroud.

The core plate structure is prevented from horizontal translation during the design basis event by friction from the clamping force from the core plate rim hold-down bolts. VY has not yet calculated the minimum number of bolts required to resist sliding against seismic shear loads. However, the existence of the aligner pins in effect reduces the clamping preload required for the core plate bolts and would reduce the number of intact bolted connections required.

Alternate Inspection Acceptance Basis

VY verified the structural integrity of the top locking engagement of all bolts as installed per drawing requirements through VT-3 inspections. A baseline VT-3 examination of the tops of all 30 bolted connections was performed in RFO 19 (1996) (Reference 5). Then, during the last three refueling outages – RFO 21 (1999), RFO 22 (2001), and RFO 23 (2002) – a VT-3 examination of the tops of 50% (15) of the bolted connections was conducted (References 6, 7, and 8). The exams performed showed no signs of cracking or bolting disassembly.

VY plans to re-inspect by VT-3 a minimum 50% sampling of these bolted connections every other refueling outage (on a rotating basis) to assess the structural integrity of the bolts top locking engagement.

Should access to the lower plenum become available, VY plans to augment core bolt inspections by performing a VT-3 inspection of accessible rim hold-down bolt bottom locking engagement and accessible aligner pin assemblies.

VY considers this inspection plan adequate, with a high confidence level, for ensuring the structural integrity of its core plate configuration to resist sliding against shear loads.

The performed top-of-the-bolt inspections confirmed that all of the 30 bolts are in place; there is no sign of deformation nor cracking; and the upper nut, nut lock and fillet weld is in place in all of the examined locations.

The lower bolt connection (see Drawing 5920-1933, Reference 12) is similar to the top in that the nut is welded by a fillet weld to the bolt (side) to keep the nut in place. It is unlikely that where there are no failed connections in the sample that has been inspected (30) that a significant number of failed connections could exist in the remainder of the population (the uninspected lower end of the bolted connections).

Additionally, VY has very good water chemistry, which meets the requirements of BWRVIP-79 (Reference 14). All components below the top of the core shroud are protected by Noble Metal Chemistry Application (NMCA) with sufficient hydrogen injection to mitigate IGSCC of vessel internals.

This alternate inspection plan offers a practical solution to the inspection criteria required by BWRVIP-25, because:

(I) No vendors have yet developed a delivery system for top-of-bolt UT techniques.

(II) The EVT-1 inspection from below the core plate has accessibility limitations. The ASME Code Section XI defines "accessible surfaces" as those areas "made accessible for examination by removal of components during normal refueling outages," during a typical refueling outage. Neither the shuffling of fuel bundles nor the replacement of control blades allows access to the core plate. Therefore, this requirement would add unnecessary increase in outage time, with no compensating benefits because a representative inspection can be performed of the upper side of the bolted connection.

<u>Conclusion</u>

VY considers this alternate inspection plan to provide an acceptable level of quality for examination of its core plate against seismic shear loads.

<u>Duration of Technical Justification</u> *(State how long the deviation will be in effect.)*

This deviation will remain in effect until a delivery system for a top-of-bolt UT technique is developed. When this occurs, adequate time for site deployment will be also be factored, as allowed by PP 7027, Paragraph 4.2.1.

<u>Assumptions/Open Items</u> *(List any assumptions used in the TJ and provide a basis for each. List any open items requiring additional action prior to closure of the TJ.)*

None.

<u>Recommendations</u> *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

None.

Approvals *(Print name and provide signature/date. A thorough review shall include and consider input from a wide variety of sources.)*

(signature)	(date)	
Carl Larson	3/25/04	Reactor Internals Management Program Coordinator
N/A	N/A	Mechanical/Structural Design (if applicable)
N/A	N/A	Chemistry (if applicable)
[illegible]	3/26/04	Other Cross-Discipline or Independent Review (if applicable)
Dennis C. Girroir	3/26/04	Code Programs Manager

TJ 2004-01
Page 3 of 4

Closeout *(All actions that were recommended by the Technical Justification and accepted by management have been initiated and any identified open items have been dispositioned.)*

[illegible] / 3/26/04 Reactor Internals Management Program Coordinator
(signature) (date)

Input Documents and other References – The following documents provide input to this Technical Justification.

#	Document Title (including Rev. No. and Date, if applicable)
1	BWRVIP-25, December 1996, BWR Core Plate Inspection and Flaw Evaluation Guidelines (EPRI TR-107284)
2	BWRVIP Response to NRC RAI on BWRVIP-25 and BWRVIP-26, December 19, 1997
3	NRC Safety Evaluation of BWRVIP-25, April 28, 1999
4	NRC Final Safety Evaluation of BWRVIP-25, December 19, 1999
5	GENE Report dated October 6, 1996, "In-Vessel Visual Examination Report for the Vermont Yankee Nuclear Power Plant RFO 19 September/October 1996"
6	Framatome Technologies Report dated November 17, 1999, "1999 RFO 21 Outage Reactor, In-Vessel Services Report for Vermont Yankee Nuclear Power Corporation"
7	Framatome Technologies Report dated May 13, 2001, "2001 RFO 22 Outage Reactor, In-Vessel Services Report for Vermont Yankee Nuclear Power Station"
8	Framatome Technologies Report dated October 14, 2002, "2002 RFO 23 Outage Reactor, In-Vessel Services Report for Vermont Yankee Nuclear Power Station"
9	PP 7027, Reactor Vessel Internals Management Program
10	NE 8067, Reactor Vessel Internals Inspection Details
11	VY Drawing 5920-1101
12	VY Drawing 5920-1933
13	VY Drawing 5920-1097
	BWRVIP-79, February 2000, "BWR Water Chemistry Guidelines" (EPRI TR-103515R2)
15	BWRVIP-94, August 2001, BWRVIP Program Implementation Guide (1006288)

TJ 2004-01
Page 4 of 4

Technical Justification No. 2004-02

Title: Justification for Deferral of Inspection of Inaccessible Welds

Technical justification is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines.

BWRVIP Requirement *(Give BWRVIP document and Section reference with a restatement of the requirement.)*

The BWRVIP requires inspection of certain "hidden" or inaccessible welds. There are three hidden welds inside each of the two core spray nozzles and two hidden welds inside each of the ten jet pump recirc inlet nozzles. The BWRVIP also requires that the integrity of the P9 welds inside the core spray shroud collars be considered when the associated P8b weld integrity is diminished.

Core Spray

The core spray thermal sleeve provides the flow path for core cooling water from the vessel nozzle external piping to the core spray piping tee-box. The core spray hidden welds are described in VY procedure NE 8067 (Reference 8), Appendix A, Paragraph 5.2. "There are three welds on each thermal sleeve. CSTS-1 is the safe-end tuning fork-to-10-inch schedule-40 pup piece. CSTS-2 is the pup piece-to-10-inch by 8-inch concentric standard weight reducer. CSTS-3 is the reducer-to-8-inch schedule 40 pipe piece."

BWRVIP-18 (Reference 1), Paragraph 3.2.4, states, "There is currently no inspection technique to inspect the thermal sleeve welds. This development need is being addressed by the BWRVIP Inspection Committee as a high priority item. Inspection of thermal sleeve welds should be done when the capability exists, following Figure 3-1 as appropriate for creviced or non-creviced welds." Figure 3-1 also references the reinspection flowchart (Figure 3-3). Those flowcharts would require EVT-1 every refueling outage or UT inspection every other refueling outage of a "full target weld set". Since EVT-1 is impossible, that leaves UT. The full target weld set is defined in Table 3-5 as ¼ of the welds that are non-creviced. Therefore, if the thermal sleeve welds are non-creviced, they can be grouped into the target weld set where ¼ are required to be examined every other refueling outage.

Subsequent to publication of BWRVIP-18, the BWRVIP Inspection Committee produced a study (Reference 6) showing that inspection of the core spray and jet pump hidden welds could be possible, but it would be difficult and extremely costly. No vendor has undertaken the work to develop tooling in order to examine the hidden welds.

Further, indications have been recorded during ultrasonic examination of welds 1P8b and 3P8b (collar-to-shroud welds) at Vermont Yankee. A BWRVIP response, dated January 11, 1999, to the NRC Safety Evaluation of BWRVIP-18 contains guidance for the redundant core spray P9 welds inside the collar at the piping-to-shroud connection. This guidance is considered mandatory per BWRVIP-94, Section 1.3, because the BWRVIP Executives approved the response letter to the NRC. The guidance states in response to Issue 3.6(2) that, "Weld P9 is redundant to the P8a and P8b welds in BWR/3-5 plants. Therefore, consideration of the integrity of P9 only needs to be considered if the integrity of the P8a and P8b welds is insufficient."

Jet Pumps

The jet pump hidden welds are described in VY procedure NE 8067 (Reference 8), Appendix A, Paragraph 10.3. "The thermal sleeve attaches the N2 nozzle safe-end to the jet pump riser elbow. It provides a flow path and reduces temperature variations, and thus thermal loading, on the N2 nozzle. There are two full penetration circumferential welds in each of the ten jet pump thermal sleeves. TS-1 is the safe-end-to-thermal sleeve concentric reducer. TS-2 is the reducer-to-10-inch special pipe."

BWRVIP-41 (Reference 4), Table 3.3-1 requires for the baseline inspection of welds TS-1 and TS-2, "Modified VT-1 of 100% of weld HAZs over next two inspection cycles. 50% to be inspected in next inspection cycle." The required reinspection is, "25% per inspection cycle." A note states: "[These] welds may not be accessible for visual inspection. The BWRVIP Inspection Committee is currently addressing the need for developing an inspection technique for this weld. Inspection recommended when the technique becomes available."

Subsequent to publication of BWRVIP-41, the BWRVIP Inspection Committee produced a study (Reference 6) showing that inspection of the core spray and jet pump hidden welds could be possible, but it would be difficult and extremely costly. No vendor has undertaken the work to develop tooling in order to examine the hidden welds.

Inspection cycle is defined in BWRVIP-41, Section 3.2.1 as 6 years. Per PP 7027 (Reference 7), the first six-year inspection cycle is defined as starting as of the publication of BWRVIP-41, and thus covers the time frame of October 1997 through October 2003.

Vermont Yankee Deviation *(Record how Vermont Yankee deviates or deviated from the BWRVIP requirement.)*

In view of the fact that no vendors have yet developed a delivery system to examine any of the hidden welds in the thermal sleeves inside the either the core spray nozzles or the jet pump nozzles, no inspection of these welds has taken place. Further, even though examinations of the P9 welds at Vermont Yankee were attempted, the NDE technique qualifications for examination of the P9 weld were withdrawn by the BWRVIP. Therefore, no qualified examinations of the P9 welds redundant to the 1P8b and 3P8b welds have ever been performed.

Justification *(Provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the BWRVIP guidelines. The justification shall be supported by calculations when warranted. Clearly identify all available information and resources, which allow the deviation to be acceptable. Clearly identify the impact that the deviation will have on meeting the intent of the guideline.)*

Acceptance Basis

The hidden welds are not accessible for visual examination and would be a challenge for mechanized UT examination. There is currently no inspection technique developed to inspect the thermal sleeve welds either with some degree of component disassembly or through development of specialized techniques.

Core Spray Thermal Sleeve Welds

Until such time as an inspection technique is available, BWRVIP-18 (Reference 1), Section 3.2.4 "Hidden Welds", states, "...a qualitative assessment of thermal sleeve integrity can be based on a plant-specific evaluation of similar core spray piping welds. If a plant has uncreviced thermal sleeve welds, the evaluation welds should be the junction box-to-pipe welds and the upper elbow welds. If the thermal sleeve welds are creviced, the evaluation welds should be the junction box cover plate weld, where applicable, the P1 weld in BWR/3-5 plants where accessible for inspection, and the downcomer sleeve welds." Regardless of whether VY's thermal sleeve welds are creviced, none of the above "evaluation welds" at VY (28 welds in all) show any indications of cracking. Therefore, the qualitative assessment of the core spray thermal sleeve welds is satisfactory.

BWRVIP-18, Section 2.2.1, states that most thermal sleeve welds are full penetration welds, but that some are creviced fillet welds, and at least one is a creviced partial penetration weld. Then from the way that is worded, full penetration thermal sleeve welds would be considered to be non-creviced. The three core spray thermal sleeve welds in each of two nozzles are full penetration butt welds. So therefore, the likelihood that cracking could initiate in these welds is diminished.

BWRVIP-18, Section 3.2.4 further states that, "If a thermal sleeve weld were to crack to the point of separation, the thermal sleeve and attached core spray piping might undergo some displacement, but the brackets holding the piping and/or the tight clearance between the thermal sleeve and nozzle wall would prevent gross separation. In such an extreme scenario, core spray would still be provided, but with some leakage."

Core Spray P9 Welds

Because indications have been recorded during ultrasonic examination of welds 1P8b and 3P8b (collar-to-shroud welds) at Vermont Yankee, integrity of the P9 welds must be considered. The BWRVIP response, dated January 11, 1999, to the NRC Safety Evaluation of BWRVIP-18 states that, "Until such time that inspection of P9 is practical and demonstrated for all plant configurations, other technically founded approaches are needed...In the interim if the integrity of P8a or P8b is diminished, the condition of P9 would be considered in the overall integrity evaluation of the connection. The evaluation would consider the low likelihood of cracking to an extent that would jeopardize structural integrity considering susceptibility, operational loads, flaw tolerance, etc."

Vermont Yankee provided an evaluation to the NRC of these welds and the CS piping-to-shroud connection (Reference 10). That evaluation assumed in one of three cases that the collar-to-shroud weld failed completely, "in which case the core spray annulus piping is capable of displacing up to ¼ inch axially and up to 0.028 inches vertically and horizontally." This evaluation assumed an intact P9 weld, however.

The same logic that was used for the core spray thermal sleeve hidden welds can be applied to the P9 welds. A qualitative assessment of thermal sleeve integrity can be based on a plant-specific evaluation of similar core spray piping welds. The P9 welds are creviced. All other creviced core spray welds at Vermont Yankee – the junction box cover plate welds, the P1 welds, and the downcomer sleeve welds (16 welds in all) – show no indications of cracking. Therefore, the qualitative assessment of the core spray thermal sleeve welds is satisfactory.

Vermont Yankee has an internal commitment to perform examination of the P9 welds when an NDE technique becomes qualified.

Jet Pumps

BWRVIP-41 (Reference 4), Section 2.3.3.7 states, "The thermal sleeve welds are categorized as medium priority locations for plants that inject LPCI flow through the recirculation system." Also, "...the BWRVIP is pursuing analyses which may reduce or alleviate inspection of TS-1 through TS-4 welds. In the meantime, the same section further states, "If a thermal sleeve weld were to crack to the point of separation, the thermal sleeve and attached riser pipe may experience some displacement, but the displacement would be small as discussed in Section 2.3.3.5." Section 2.3.3.5 states, "Failure of welds TS-1 through TS-4 will not result in large vertical displacement of the jet pump assembly due to interference between the portion of the thermal sleeve which remains attached to the riser elbow and the interior surface of the nozzle. Therefore, jet pump disassembly is not predicted for this type of failure."

Further, "...horizontal displacement of the riser pipe is limited by interference with the shroud. Welds TS-1, TS-2, and TS-3 are far enough into the nozzle such that failure at these welds would not result in the thermal sleeve disengaging from the nozzle before the riser contacted the shroud." This has been confirmed to be true at VY, as follows: Weld TS-2 may be as close as 5¼" to the inside of the nozzle blend radius (Drawings 5920-656 and 5920-6625 – References 11 and 15). The extrados of the jet pump riser elbow is nominally 16½" from the vessel ID (Drawings 5920-656 and 5920-1127 – References 11 and 12). The OD shroud radius at the core elevation is 83 5/8" and the vessel ID radius is 102½" (Drawing 5920-3773, Sheet 2 – Reference 13). The shroud to vessel annulus dimension is therefore 18 7/8". Consequently, the jet pump could deflect approximately 2½" in the radial direction, which is much less than the 5¼" before weld TS-2 exited the confines of the nozzle.

If the thermal sleeve or riser piping severed it would be detected through jet pump M-ratio monitoring. OP 4110 (Reference 9) states, "M-ratio is a calculated value which is used to detect the severance and displacement of the jet pump riser pipe. ERFIS points C286 (recirc loop A M-ratio) and C287 (recirc loop B M-ratio) have a ±10% alarm setpoint while at or above a core flow of 42.0 M#/hr." Additionally, Technical Specification 4.6.F contains jet pump operability criteria.

VY has very good water chemistry, which meets the requirements of BWRVIP-79 (Reference 16). All components below the top of the core shroud are protected by Noble Metal Chemistry Application (NMCA) with sufficient hydrogen injection to mitigate IGSCC of vessel internals. This includes the jet pump thermal sleeve welds.

Conclusion

Vermont Yankee considers this technical justification to provide an acceptable level of quality to demonstrate the structural integrity of the core spray and jet pump thermal sleeves to perform their intended function.

Duration of Technical Justification *(State how long the deviation will be in effect.)*

This deviation will remain in effect until a delivery system for UT of the hidden welds is developed. When this occurs, adequate time for site deployment will be also be factored, as allowed by PP 7027 (Reference 7), Paragraph 4.2.1.

Assumptions/Open Items *(List any assumptions used in the TJ and provide a basis for each. List any open items requiring additional action prior to closure of the TJ.)*

None.

Appendix C (Continued)

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

None.

Approvals *(Print name and provide signature/date. A thorough review shall include and consider input from a wide variety of sources.)*

Carl Jansen / 3/25/04 Reactor Internals Management Program Coordinator

N/A / N/A Mechanical/Structural Design (if applicable)

N/A / N/A Chemistry (if applicable)

/ 3/26/04 Other Cross-Discipline or Independent Review (if applicable)

Dennis Gibbons / 3/26/04 Code Programs Manager

(signature) (date)

Closeout *(All actions that were recommended by the Technical Justification and accepted by management have been initiated and any identified open items have been dispositioned.)*

/ 3/26/04 Reactor Internals Management Program Coordinator

(signature) (date)

TJ 2004-02
Page 5 of 6

Appendix C (Continued)

Input Documents and other References – The following documents provide input to this Technical Justification.

#	Document Title (including Rev. No. and Date, if applicable)
1	BWRVIP-18, dated July 1996, BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines (EPRI TR-106740)
2	Letter USNRC to BWRVIP, dated December 2, 1999, "Final Safety Evaluation of Core Spray Internals Inspection and Flaw Evaluation Guidelines (BWRVIP-18)"
3	Letter BWRVIP to USNRC, dated January 12, 1999, "BWRVIP Response to NRC Safety Evaluation of BWRVIP-18"
4	BWRVIP-41, dated October 1997, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (EPRI TR-108728)
5	Letter USNRC to BWRVIP, dated February 4, 2001, "Final Safety Evaluation of the BWRVIP, BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines (BWRVIP-41)"
6	Framatome Technical Report for Contract WOB201-04, dated November 1999, "Probes and Hardware for Examination of Thermal Sleeve Welds in BWR Core Spray, Low Pressure Coolant Injection, and Recirculation Inlet Nozzles"
7	PP 7027, Reactor Vessel Internals Management Program
8	NE 8067, Reactor Vessel Internals Inspection Details
9	OP 4110, Reactor Recirc System Surveillance
10	Letter Vermont Yankee to USNRC, dated October 9, 1996, BVY 96-118, "Core Spray System Inspection at Vermont Yankee"
11	VY Drawing 5920-656
12	VY Drawing 5920-1127
13	VY Drawing 5920-3773
14	VY Drawing 5920-6624
15	VY Drawing 5920-6625
16	BWRVIP-79, February 2000, "BWR Water Chemistry Guidelines" (EPRI TR-103515R2)
17	BWRVIP-94, August 2001, BWRVIP Program Implementation Guide (1006288)

Technical Evaluation No. 2004-0018

Technical Evaluation No. TE-2004-0018

Title: Justification to Inspect Portions of Shroud Horizontal Welds H1, H2, and H3 on the OD In Lieu of the Top Guide Spacer Block Welds, the Shroud Flange Ring Segment Welds, and the Top Guide Ring Segment Welds

■ QA (Safety Class, OQA, or Vital Fire) ☐ Non QA (Non-Safety) (Check One)

Background *(Enter a concise summary of the condition or reason for the requested TE stating the existing condition and the desired results. State the scope of the requested TE.)*

In RFO 19 (1996), Vermont Yankee installed four tie-rods to repair the core shroud horizontal welds. Per UFSAR, Appendix K, the shroud welds repaired are considered to be H3, H4, H5, H6, and H7. An inspection by INPO identified a discrepant condition between what the repair designer (MPR) considers to be design-reliant welds and what was in fact inspected at VY as being design-reliant, this is documented in ER20012481. The designer of the shroud repair, MPR Associates, relied on the following welds as design-reliant:

- The twelve support blocks welded to the inside of the shroud at the top guide elevation
- Three ring segment welds at the shroud flange elevation
- Three ring segment welds at the top guide elevation
- Three ring segment welds at the core plate elevation
- All vertical welds, except those between H1 and H2

BWRVIP-76, Section 3.2, states, "At some plants, a shroud repair may not include all relevant horizontal welds. The inspection strategy for un-repaired horizontal welds in a repaired core shroud is identical to that for horizontal welds in un-repaired Category C shrouds (see Section 2)." Section 2.2.1 states, "For Category C shroud 100 percent of the accessible regions of welds H1 through H7 inclusive are to be inspected." Section 2.2.2 states, "...the preferred inspection techniques are volumetric inspection (UT) and/or a two-sided surface exam (i.e., EVT-1)..." Figure 2-3 of BWRVIP-76 provides a flow chart describing the inspection strategy.

Because of the difficulty in examining certain of the above welds, especially the support block welds, and the relative flaw-free condition of welds H1, H2, and H3. Vermont Yankee ISI Group requested the Mechanical/Structural Design Engineering Group (MSD) to evaluate and re-designate the welds that are design-reliant for the shroud repair.

Discussion *(Record the evaluation considerations and the results of the evaluation. Describe any features that required special attention during the TE process. Document and validate any assumptions made during the evaluation.)*

The MSD Group had the original shroud repair designer MPR Associates evaluate changing the design reliant welds. The calculations used are defined in References 1, 2, 3 and 4 in the Design Input Section of this TE. The calculations attempted to show that when the support blocks and

Technical Evaluation No. 2004-0018

associated welds were included in the model, (the blocks fit between the shroud wall between H1 and H2, and the horizontal section of the shroud between H3 and H2) the resultant weld stresses would be below 20% of allowable stress consistent with BWRVIP-26 (Ref. 5.) If these resultant stresses were below 20% of allowable stresses this would have made inspections the support block welds redundant. The stresses in the vertical shroud between H1 and H2 and in the horizontal shroud between H2 and H3 included both primary and secondary stresses from plate bending that resulted in stresses being greater than the 20% of allowable stresses. The model was then run with only the lateral support blocks included in the model and included all the loads from the Top Guide. The results showed that with the blocks removed the stresses in both the horizontal shroud (H3 to H2) and the vertical shroud (H2 to H1) all stresses were below allowable stresses. The required length of weld in H1, H2 and H3 is 3.13 inches in each of the four quadrants spaced approximately equally. The actual length to be inspected would be 18 inches in each quadrant to allow for crack growth over the next six years. The radial welds from H3 to H2 do not need to be inspected and similarly the vertical welds between H2 and H1 do not need to be inspected.

Through-wall cracks were assumed for the un-inspected length. Per BWRVIP-76, (Reference 13) Appendix D, if less than 50% of the length is inspected, then a statistical argument for the un-inspected region is not allowed. Statistical arguments were not used.

If cracks are found in the inspected regions of H1, H2 and H3 then an increase in the sample length of 18 inches should be done. The lengths of weld should be consistent with the requirement that the sample length be increased in that quadrant to ensure there is adequate length of good weld available.

Vermont Yankee is bound by certain commitments to follow the guidance of the BWRVIP (References 6 through 12). BWRVIP-94 (Reference 11), Appendix A states that a technical justification shall be required when utility methodology is inconsistent with the intent of the supporting BWRVIP guidelines. Additionally, at VY, the inside of the shroud is not accessible at H1, H2, and H3 to perform an EVT-1. The core spray spargers cover H1 and H2, and because of the grating that covers the periphery of the top guide, access to the shroud ID would be through vacated fuel cells, and this would result in the camera being too distant from the inspection surfaces to perform an adequate EVT-1 of H1, H2, or H3. Therefore, VY will not meet the BWRVIP requirement to inspect both the OD and ID of the welds and will not meet the BWRVIP requirement to inspect 100% of the length of the welds. This document justifies this variance from the BWRVIP requirements.

Although no BWRVIP guidance is given for one-sided visual examinations of horizontal welds, the six-year inspection frequency follows the guidance for a one-sided EVT-1 *of vertical welds* per BWRVIP-76, Figure 3-3. The excellent results obtained in the 1995 ultrasonic examination of the H1, H2 and H3 welds (very limited indications) and the 1996 ultrasonic examination of the vertical and ring segment welds (no indications found) provide additional assurance that a one sided EVT-1 is acceptable.

Technical Evaluation No. 2004-0018

Also, Appendix K of the FSAR will need to be revised. This section of the UFSAR states that H1 and H2 are design-reliant welds (but does not include H3) and it states that the ring segment welds between H2 and H3 are design-reliant welds and all the welds connecting the twelve support blocks to both the horizontal section of the shroud (H3 to H2) and the vertical section of the shroud (H2 to H1.) The ring segment welds and the support block welds are no longer design reliant.

Assumptions/Open Items *(List any assumptions used in the TE and provide a basis for each. List any open items requiring additional action prior to closure of the TE.)*

None

Material Requirements/Implementation Instructions *(List any identified specifications for equipment, materials, or services needed to implement the recommendations of the TE. Specify any special implementation instructions or cautions, such as field testing requirements or system interface requirements during implementation.)*

The required inspections are as follows:

1. For the shroud horizontal welds H1, H2 and H3, inspect 18 inches in length in each of the four quadrants from the outside diameter (OD) using EVT-1 methods in accordance with NE 8048. If cracks are found in a quadrant, expand the length inspected in that quadrant to detect 18 inches of unflawed weld. Due date 05/15/04.
2. Inspect 100% of the accessible length of the shroud vertical weldsS4V1, S4V2, S5V1, S5V2, S7V1 and S7V2 from the OD using EVT-1 methods in accordance with NE 8048. Inspect shroud ring segment welds S6R!, S6R2 and S6R3 (at the core plate elevation) from the OD using EVT-1 methods in accordance with NE 8048. Due date 11/15/05.

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

Based on the above analysis, this TE recommends the following actions:

1. Inspect the welds as follows:

1A. For the shroud horizontal welds H1, H2 and H3, inspect 18 inches in length in each of the four quadrants from the outside diameter (OD) using EVT-1 methods in accordance with NE 8048. If cracks are found in a quadrant, expand the length inspected in that quadrant to detect 18 inches of unflawed weld. Due date 05/15/04.

1B. Inspect 100% of the accessible length of the shroud vertical weldsS4V1, S4V2, S5V1, S5V2, S7V1 and S7V2 from the OD using EVT-1 methods in accordance with NE 8048. Inspect shroud ring segment welds S6R!, S6R2 and S6R3 (at the core plate elevation) from the OD using EVT-1 methods in accordance with NE 8048. Due date 11/15/05.

Responsible Department – System Engineering, Code Programs, dates as specified.

Technical Evaluation No. 2004-0018

Revise Appendix K of the UFSAR.
Responsible Department – Design Engineering
Due Date – Later

Appendix C (Continued)

Technical Evaluation No. 2004-0018

Approvals/Closeout ***(Print name and provide signature/date.)***

_____________________ / ________ Independent Reviewer (qq)

_____________________ / ________ Independent Reviewer (Code Programs Supervisor)

_____________________ / ________ Mechanical/Structural Design Supervisor

_____________________ / ________ Design Engineering Manager
(signature) (date)

Closeout ***(All actions that were recommended by the TE and accepted by management have been initiated and any identified open items have been dispositioned.)***

_____________________ / ____________ CE
(signature) (date)

Attachments ***(Provide a list of all forms, document markups, etc. provided as part of the TE package.)*** – **Please see the Technical Evaluation Data Base which lists all references used, and notes which references are attached.**

Technical Evaluation No. 2004-0018

TECHNICAL EVALUATION DATABASE INPUT

TE No.: 2004-0018

TE Title: Justification to Inspect Portions of Shroud Horizontal Welds H1, H2, and H3 on the OD In Place of the Top Guide Spacer Block Welds, the Shroud Flange Ring Segment Welds, and the Top Guide Ring Segment Welds

Keywords: Shroud, Reactor Internals, Examination, BWRVIP

Design Input Documents - The following documents provide design input to this TE.

#	Document Title (including Rev. No. and Date, if applicable)
1	**MPR Calculation 069-013-EBB-1, "Loads in the H2/H3 Support Ring Pads."**
2	**MPR Calculation 069-013-EBB-2, "Shroud Stresses."**
3	**MPR Calculation 069-013-JLH-1, "Support Pad (Blocks) and Aligner Pad Weld Evaluation."**
4	**MPR Calculation 069-013-CBS-1, "Required Intact Length for Shroud Welds H1, H2 and H3."**
5	**PP7027, Rev.1, "Reactor Vessel Internals Management Program."**
6	**BWRVIP-26, "BWR Top Guide Inspection and Evaluation Guidelines."**
7	**Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC) dated May 30, 1997, "BWR Utility Commitments to the BWRVIP"**
8	**BVY 97-123, dated September 30, 1997, "Vermont Yankee's Plans for the 1998 and 1999 Refueling Outages Regarding Reactor Vessel Internals"**
9	**Letter Brian Sheron (USNRC) to Carl Terry (BWRVIP Executive Chairman), dated July 29, 1997, "BWR Utility Commitments to the BWRVIP"**
10	**Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC) dated October 30, 1997, "BWR Utility Commitments to the BWRVIP"**
11	**BWRVIP-94, dated August 2001, "BWRIP Program Implementation Guide" EPRI TR 1006288**
12	**Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC), dated April 16, 2002, "Project No. 704 – Utility Implementation of BWRVIP Products"**
13	**BWRVIP-76, dated November 1999, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines" EPRI TR-114232**
14	**BWRVIP-03, Revision 5, dated December 2002, "Reactor Pressure Vessel and Internals Examination Guidelines"**
15	**Letter Carl Terry (BWRVIP Executive Chairman) to Brian Sheron (USNRC) dated May 30, 1997, "BWR Utility Commitments to the BWRVIP"**
16	**BWRVIP-14, dated March 1996, "Evaluation of Crack Growth in BWR Stainless Steel RPV Internals" EPRI TR-105873**
17	**Letter Jack Strosnider (USNRC) to Carl Terry (BWRVIP Executive Chairman), dated December 3, 1999, "Final Safety Evaluation of Proprietary Report TR 105873 'BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Stainless Steel RPV**

Technical Evaluation No. 2004-0018

	Internals (BWRVIP-14)' (TAC No. M94975)"
18	**GE Nuclear Energy Report No. GE-qq, Revision q, dated qq, "Shroud Welds Fl Evaluation Handbook for Vermont Yankee"**
19	**UFSAR Appendix K**

Design Output Documents - The following documents are impacted by this TE.

#	Document Title
UFSAR	**Revise Appendix K to define the new design reliant welds**

General References

#	Reference Title (including Rev. No. and Date, if applicable)
	None

Data Entered into Database ______________ / ________ Entry Verified ______________ / ________
Signature Date Signature
Date

Technical Evaluation No. 2004-0018

TECHNICAL EVALUATION REVIEW

TE #: 2004-0019 Required Date:________ Reviewer Assigned:________

Title: Justification to Inspect Portions of Shroud Horizontal Welds H1, H2, and H3 on the OD In Place of the Top Guide Spacer Block Welds, the Shroud Flange Ring Segment Welds, and the Top Guide Ring Segment Welds

Comments:

________ / ________
Reviewer Signature Date

Resolution:

________ / ________
DCE Signature Date

________ / ________
Reviewer Signature Date

Notes and Requirements:

1. Validate design input appropriateness relative to your area of expertise.
2. Verify Department Procedures, Program inputs, and output documents are appropriately addressed.
3. Meetings or discussions to resolve questions and comments are encouraged.
4. Resolution by telecon is acceptable and should be noted as such.
5. Make comments specific, and avoid generalizations and questions.
6. If no comments indicate "None"
7. Request Management assistance if resolution can not be achieved.
8. Return all comments to the CB by required date or request an extension.

VYAPF 6045.04
AP 6045 Original
Page 5 of 17

Appendix C (Continued)

Technical Justification No. TJ 2004-04

Title: Inspection Technique for Weld H9

Technical justification is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines.

<u>BWRVIP Requirement</u> *(Give BWRVIP document and Section reference with a restatement of the requirement.)*

BWRVIP-38 (Reference 1), Figure 3-5, requires an EVT-1 of the top surface of weld H9 – or as an alternative – a UT of weld H9. BWRVIP-104 (Reference 5), Section 9.2, states, "Perform an EVT-1 visual examination, or ultrasonic examination, of both the top and bottom surfaces of the shroud support plate-to-RPV weld (H9) in accordance with BWRVIP-38…The ultrasonic examination should be demonstrated in accordance with BWRVIP-03 for the detection of both axial and circumferential flaws in the weld material. The technique shall be capable of determining if any flaws have propagated into the RPV low alloy steel."

<u>Vermont Yankee Deviation</u> *(Record how Vermont Yankee deviates or deviated from the BWRVIP requirement.)*

EVT-1 of the underside of weld H9 would require access through fuel cells or jet pumps. In view of the fact that manipulation for cleaning and a close visual examination would not be possible with current visual inspection technology, Vermont Yankee will not visually inspect the underside of weld H9. Further, no vendors are now qualified to detect axial flaws using UT, so Vermont Yankee's UT inspection of weld H9 will not be capable of detecting axial flaws.

<u>Justification</u> ***(Provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the BWRVIP guidelines. The justification shall be supported by calculations when warranted. Clearly identify all available information and resources, which allow the deviation to be acceptable. Clearly identify the impact that the deviation will have on meeting the intent of the guideline.)***

<u>Acceptance Basis</u>

Vermont Yankee performed a UT inspection of 22% of H9 in RFO 19 (1996) and no flaws were found (Reference 8). This met the required extent of examination for BWRVIP-38 (Reference 1), as explained in PP 7027, Appendix B, Section 4.1. However, the UT technique was not capable of detecting transverse flaws.

Cracking has been identified in the Tsuruga 1 and Nine Mile Point 1 (both BWR/2 plants) shroud support-to-RPV welds, which indicates that stress corrosion cracking is present in the alloy 182 welds of those two shroud support structures. The identified cracking at both plants was primarily on the bottom surface of the H9 weld. See References 6 and 7. The Tsuruga operating experience was the instigator for the BWRVIP-104 (Reference 5) inspection recommendations. Vermont Yankee would be required to examine at least 10% of the weld by either EVT-1 from both the top and bottom sides or by UT. Either exam would be required to detect transverse cracking. For the H9 weld, transverse would be vertical in the radial direction.

Most industry inspections of the shroud support plate to RPV weld H9 have been performed using EVT-1 from the top surface of the weld, with some examinations by UT from the RPV OD surface. Visual examination of the bottom surface of the H9 weld typically has not been performed due to limited accessibility to that surface, which is only accessible through the JP diffuser or through a disassembled fuel cell. At Vermont Yankee, there is no access from the outside of the reactor vessel at the elevation of the H9 weld.

There are two reasons for inspection of H9 in accordance with BWRVIP-38 and BWRVIP-104. One is to assure that the integrity of the shroud support structure is maintained. The other is to assure that any flaws found in H9 do not propagate into the RPV pressure boundary governed by ASME Section XI.

The integrity of the shroud support structure is assured by: 1) a UT inspection in accordance with BWRVIP-38; 2) acceptable UT inspection results; and 3) good water chemistry.

BWRVIP-38 was written to assure that the integrity of BWR shroud supports is maintained. The UT inspection for circumferential flaws achieves this goal by meeting the guidance of BWRVIP-38. Per BWRVIP-38, Table 5-1, Vermont Yankee has the fourth lowest load multiplier in the fleet and therefore, one of the greatest flaw tolerances. Since Vermont Yankee has inspected 22% of the H9 weld, there is good assurance that the integrity of the shroud support has been maintained.

It is important to note that the core shroud support configuration of the BWR/2 plants, such as Tsuruga and Nine Mile Point 1 is different than the CBI BWR/3, 4 and 5 plants, in that the support at BWR-2 plants consists of a conical-shaped support ring, while the newer configuration has a horizontal supporting ring plate with legs. The new design appears to have better loading distribution. Vermont Yankee has the newer design with 14 legs. Other than some minor cracking in a leg weld at Monticello, there have been no adverse operating experience reports on the newer design.

Regarding the second issue concerning possible transverse flaw propagation into the RPV pressure boundary, the following arguments can be given. In both the Tsuruga and Nine Mile Point 1 shroud support H9 welds, the predominant flaws were transverse; however, there were also associated circumferential flaws in both cases. Because the Vermont Yankee H9 weld examination did not reveal any circumferential cracking, there is a lowered probability that associated transverse cracking would exist. Transverse cracking did not exist in the absence of circumferential cracking in the two known cases.

Also, Vermont Yankee does not know of any cases in any BWR where internal attachment weld flaws have propagated into low-alloy base material. BWRVIP-48 (Reference 2), Section 3.1.1 states, "No propagation of indications into the vessel base material has been found in the inspections [of attachment welds] performed to date." It is also important to note that of the many transverse cracks found in the H9 weld at Tsuruga, all were excavated and none of the flaws were found to have propagated into the RPV low alloy steel material (Reference 7). This is statistically a very large sample, and therefore, it can be concluded that a contrary result would occur with very low probability.

The shroud support examinations performed at Vermont Yankee have shown no signs of cracking. Vermont Yankee has very good water chemistry with HWC and NMCA, meeting the requirements of BWRVIP-79 (Reference 3).

Conclusion

Vermont Yankee considers this technical justification to provide an acceptable level of quality to demonstrate the structural integrity of shroud support weld H9 to perform its intended function.

Duration of Technical Justification *(State how long the deviation will be in effect.)*

This deviation will remain in effect until a UT technique is qualified for detection and sizing of transverse cracks in the H9 weld. When this occurs, Vermont Yankee will use such a technique at the subsequent examination of weld H9, per the scheduling requirements of BWRVIP-38.

Assumptions/Open Items *(List any assumptions used in the TJ and provide a basis for each. List any open items requiring additional action prior to closure of the TJ.)*

None.

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

None.

Approvals *(Print name and provide signature/date. A thorough review shall include and consider input from a wide variety of sources.)*

(signature)	(date)	
Carl Larsen	3/25/04	Reactor Internals Management Program Coordinator
N/A	N/A	Mechanical/Structural Design (if applicable)
N/A	N/A	Chemistry (if applicable)
[signature]	3/26/04	Other Cross-Discipline or Independent Review (if applicable)
[signature]	3/26/04	Code Programs Manager

Closeout *(All actions that were recommended by the Technical Justification and accepted by management have been initiated and any identified open items have been dispositioned.)*

(signature)	(date)	
[signature]	3/26/04	Reactor Internals Management Program Coordinator

TJ 2004-04
Page 3 of 4

Appendix C (Continued)

Input Documents and other References – The following documents provide input to this Technical Justification.

#	Document Title (including Rev. No. and Date, if applicable)
1	BWRVIP-38, September 1997, BWR Shroud Support Inspection and Flaw Evaluation Guidelines (EPRI TR-108823)
2	BWRVIP-48, February 1998, BWRVIP Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines (EPRI TR-108724)
3	BWRVIP-79, March 2000, BWR Water Chemistry Guidelines-2000 Revision
4	BWRVIP-94, August 2001, BWRVIP Program Implementation Guide (1006288)
5	BWRVIP-104, September 2002, BWR Evaluation and Recommendations to Address Shroud Support Cracking in BWRs (1003555)
6	GE SIL 624, March 24, 2000, Stress Corrosion Cracking in Alloy 182 Welds in Shroud Support Structure
7	BWRVIP Report, July 14, 2000, Summary of June 13, 2000 Meeting with JAPC on Tsuruga Unit 1 Shroud Support Cracking
8	Framatome Technologies Report dated December 18, 1996, "1996 Vermont Yankee Nuclear Power Corporation Project File Report for Core Shroud Examinations of the Vertical, Ring Segment, and H8/H9 Baffle Plate Welds"
9	NOP01A1, Reactor Vessel Internals Inspection Program
10	NE21.01, Reactor Vessel Internals Inspection Implementing Procedure
11	VY Drawing 5920-252

Technical Justification No. TJ-2004-05

Title: Inspection Deferral for UT of SLC Safe-end

Technical justification is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines.

__BWRVIP Requirement__ *(Give BWRVIP document and Section reference with a restatement of the requirement.)*

BWRVIP- 27-A (Reference 3), Sections 3.3.1 and 3.4.1 require a UT of the nozzle-to-safe-end weld and the safe-end extension when a reliable UT technique is identified. Until such time as a qualified volumetric examination is available, enhanced leakage inspections (EVT-2) or surface examinations (PT) may be performed. (When BWRVIP-27-A was published in August 2003, it replaced BWRVIP-27 [Reference 2].) BWRVIP-03 (Reference 1), Revision 6, Standard 2.6, Section 3.3, states, "Personnel performing final analysis and review of examinations of dissimilar metal welds in the standby liquid control system shall have current qualification for crack detection, length sizing, and/or depth sizing, as appropriate, in accordance with ASME Section XI, Appendix VIII, Supplement 10 [Reference 6]. The qualification's scope shall include the diameter and thickness of the applicable standby liquid control welds."

__Vermont Yankee Deviation__ *(Record how Vermont Yankee deviates or deviated from the BWRVIP requirement.)*

UT techniques and personnel have been qualified for detection in accordance with ASME Section XI, Appendix VIII, Supplement 10 (Reference 6), for the diameter and thickness of the VY nozzle-to-safe-end weld. Qualifications have been issued for ultrasonic techniques and personnel performing detection, length sizing, and depth sizing using automated equipment. Qualifications have also been issued for techniques and personnel performing manual ultrasonic detection and length sizing, but not for manual ultrasonic depth sizing.

There are two problems associated with the ultrasonic qualifications issued to date:

1) Automated UT equipment will probably not fit in this limited-access location.
2) It would be risky to perform manual UT for detection of cracking without having a through-wall sizing technique. If a flaw were detected, a repair (probably by weld overlay) would automatically be necessary. Automated weld overlay equipment would have the same access problem as automated UT equipment.

As an alternative to the ultrasonic examination, Vermont Yankee will continue to perform either EVT-2 every refueling outage or PT every other refueling outage.

__Justification__ *(Provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the BWRVIP guidelines. The justification shall be supported by calculations when warranted. Clearly identify all available information and resources, which allow the deviation to be acceptable. Clearly identify the impact that the deviation will have on meeting the intent of the guideline.)*

__Acceptance Basis__

BWRVIP-27-A (Reference 3), published in August 2003, governs inspection of the SLC and core plate ΔP system. BWRVIP-27-A asserts that the only safety critical welds in the SLC/Core Plate ΔP system within the scope of the BWRVIP are the welds outside the reactor vessel which connect the SLC system piping to the vessel. BWRVIP-27-A, Section 2.1.5 and Figure 2-5 describe the Vermont Yankee

TJ-2004-05
Page 1 of 3

configuration, which is a stainless steel safe-end welded to a carbon steel forged nozzle and fabricated by CB&I. VY Drawing 5920-358 (Reference 13) shows this configuration and Drawing 5920-5266 (Reference 14) shows the replacement safe-end of improved material installed shortly before initial start-up. The safe-end thickness on both drawings is 7/8". The OD of the safe-end extensions is 3.69". BWRVIP-27-A, Sections 3.3.1 and 3.4.1 state the requirements for the Vermont Yankee configuration; it requires that the nozzle-to-safe end weld and the safe-end extension be examined volumetrically. However, per those same Sections, until such time as a qualified volumetric examination is available, enhanced leakage inspections (EVT-2) or surface examinations (PT) may be performed.

VY performed EVT-2 inspections of this joint in RFO 20 (1998), RFO 21 (1999), and RFO 22 (2001) and PT of the joint in RFO 23 (2002) (see References 7 through 10).

Prior to the publication of BWRVIP-27-A in August 2003, BWRVIP-27 (Reference 2), which was published in October 1997, governed inspection of the SLC system. BWRVIP-27, Sections 3.3.1 and 3.4.1 also stated that, "until such time as a qualified volumetric examination is available, enhanced leakage inspection during each Category B-P pressure boundary leak test should be performed." An enhanced leakage test is defined as requiring a view of this joint specifically, rather than as would normally be required by ASME Section XI, which would be an examination for leakage in the general area. Per BWRVIP-27-A, insulation removal is required. This was not clarified until BWRVIP-27-A was issued as a draft in July 2002. Until that time the need for insulation removal was not explicitly stated (in BWRVIP-27) and VY did not do such in RFO 20 (1998), RFO 21 (1999), and RFO 22 (2001).

A stress corrosion crack through-wall crack would be detected before the safe-end would sever completely ("leak before break"). The alternative examinations – EVT-2 or PT – would detect a leak, especially with the insulation removed. Because Vermont Yankee has inspected this location recently (<1 cycle), and because of the short time planned between future inspections (one cycle for EVT-2 or two cycles for PT), growth over this short time would not result in a complete loss of structural integrity for this joint – especially given its large OD:ID ratio (1.9). It is highly unlikely that a crack would extend through-wall in one area while at the same time losing structural integrity over the entire circumference. If evidence of leakage is found a repair should be performed.

<u>Conclusion</u>

Vermont Yankee considers this technical justification to provide an acceptable level of quality to demonstrate the structural integrity of the SLC nozzle-to-safe-end weld and safe-end extension to perform its intended function.

<u>**Duration of Technical Justification**</u> *(State how long the deviation will be in effect.)*

This deviation will remain in effect either until a UT manual technique is qualified for through-wall sizing or until automated UT equipment is developed that could access the SLC safe-end. When this occurs, adequate time for site deployment will be also be factored, as allowed by PP 7027 (Reference 11), Paragraph 4.2.1.

<u>**Assumptions/Open Items**</u> *(List any assumptions used in the TJ and provide a basis for each. List any open items requiring additional action prior to closure of the TJ.)*

None.

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

None.

Approvals *(Print name and provide signature/date. A thorough review shall include and consider input from a wide variety of sources.)*

Carl Larson / 3/25/04 Reactor Internals Management Program Coordinator

N/A / N/A Mechanical/Structural Design (if applicable)

N/A / N/A Chemistry (if applicable)

WD / 3/26/04 Other Cross-Discipline or Independent Review (if applicable)

DeGrio / 3/26/04 Code Programs Manager

(signature) (date)

Closeout *(All actions that were recommended by the Technical Justification and accepted by management have been initiated and any identified open items have been dispositioned.)*

__________ / __________ Reactor Internals Management Program Coordinator

(signature) (date)

Input Documents and other References – The following documents provide input to this Technical Justification.

#	Document Title (including Rev. No. and Date, if applicable)
1	BWRVIP-03, Revision 6, dated December 2003, "BWRVIP, Reactor Pressure Vessel and Internals Examination Guidelines (EPRI TR-1008061)"
2	BWRVIP-27, dated April 1997, "BWRVIP BWR Standby Liquid Control System/Core Plate Delta P Inspection and Flaw Evaluation Guidelines (EPRI TR-107286)"
3	BWRVIP-27-A, dated August 2003, "BWRVIP BWR Standby Liquid Control System/Core Plate Delta P Inspection and Flaw Evaluation Guidelines (EPRI TR-1007279)"
4	BWRVIP-79, dated March 2000, BWR Water Chemistry Guidelines-2000 Revision
5	BWRVIP-94, dated August 2001, BWRVIP Program Implementation Guide (1006288)
6	ASME Section XI, 1989 Edition, "Rules for Inservice Inspection of Nuclear Power Plant Components," Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," Supplement 10, "Qualification Requirements for Dissimilar Metal Piping Welds"
7	Pressure Test Report qq, dated qq 1998, " qq RFO 20"
8	Pressure Test Report qq, dated qq 1999, " qq RFO 21"
9	Pressure Test Report qq, dated qq 2001, " qq RFO 22"
10	ISI PT Report qq, dated qq 2002, " qq RFO 23"
11	PP 7027, Reactor Vessel Internals Management Program
12	NE 8067, Reactor Vessel Internals Inspection Details
13	VY Drawing 5920-358
14	VY Drawing 5920-5266

Appendix C (Continued)

Technical Justification No.

Title: [Format Model]

Technical justification is required when utility procedures, inspections, methodology, or guidelines are inconsistent with the intent of the supporting BWRVIP guidelines.

BWRVIP Requirement *(Give BWRVIP document and Section reference with a restatement of the requirement.)*

Vermont Yankee Deviation *(Record how Vermont Yankee deviates or deviated from the BWRVIP requirement.)*

Justification *(Provide the basis for determining that the proposed deviation meets the same objective and intent, or level of conservatism exhibited by the BWRVIP guidelines. The justification shall be supported by calculations when warranted. Clearly identify all available information and resources, which allow the deviation to be acceptable. Clearly identify the impact that the deviation will have on meeting the intent of the guideline.)*

Duration of Technical Justification *(State how long the deviation will be in effect.)*

Assumptions/Open Items *(List any assumptions used in the TE and provide a basis for each. List any open items requiring additional action prior to closure of the TE.)*

Recommendations *(List detailed recommendations, as required, to resolve the evaluated condition. List all documents requiring changes and attach marked up pages. Clearly state recommendations for plant modifications or changes to operating practices, including recommended changes to plant procedures.)*

Based on the above analysis, this Technical Justification recommends the following actions:

1.

Responsible Department –

Due Date –

2.

Responsible Department –

Due Date –

Appendix C (Continued)

Approvals *(Print name and provide signature/date. A thorough review shall include and consider input from a wide variety of sources.)*

______________________ / ________ Reactor Internals Management Program Coordinator

______________________ / ________ Mechanical/Structural Design (if applicable)

______________________ / ________ Chemistry (if applicable)

______________________ / ________ Other Cross-Discipline or Independent Review (if applicable)

______________________ / ________ Code Programs Manager
(signature) (date)

Closeout *(All actions that were recommended by the Technical Justification and accepted by management have been initiated and any identified open items have been dispositioned.)*

______________________ / ________ Reactor Internals Management Program Coordinator
(signature) (date)

Input Documents and other References – The following documents provide input to this Technical Justification.

#	Document Title (including Rev. No. and Date, if applicable)
1	
2	
3	
4	