March 29, 2004

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: VERMONT YANKEE NUCLEAR POWER STATION - ISSUANCE OF AMENDMENT RE: REACTOR PRESSURE VESSEL FRACTURE TOUGHNESS AND MATERIAL SURVEILLANCE REQUIREMENTS (TAC NOS. MB8119 AND MB8379)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 218 to Facility Operating License DPR-28 for the Vermont Yankee Nuclear Power Station (VYNPS), in response to your application dated March 26, 2003, as supplemented on July 24, 2003.

The amendment revises the VYNPS Technical Specifications (TSs) regarding reactor pressure vessel (RPV) fracture toughness and material surveillance requirements (SRs). Specifically, the amendment revises the pressure-temperature limits for the RPV as specified in TS Figures 3.6.1, 3.6.2, and 3.6.3. In addition, the amendment deletes TS 4.6.A.5, which specifies plant-specific RPV material SRs. These plant-specific SRs are being replaced by implementing the Boiling Water Reactor Vessel and Internals Project (BWRVIP) RPV integrated surveillance program (ISP). The details of the BWRVIP ISP will be added to the VYNPS Updated Final Safety Analysis Report.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Richard B. Ennis, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures:
1. Amendment No. 218 to License No. DPR-28
2. Safety Evaluation

cc w/encls: See next page

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President
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cc:

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. David R. Lewis
Shaw, Pittman, Potts & Trowbridge
2300 N Street, N.W.
Washington, DC 20037-1128

Ms. Christine S. Salembier, Commissioner
Vermont Department of Public Service
112 State Street
Montpelier, VT 05620-2601

Mr. Michael H. Dworkin, Chairman
Public Service Board
State of Vermont
112 State Street
Montpelier, VT 05620-2701

Chairman, Board of Selectmen
Town of Vernon
P.O. Box 116
Vernon, VT 05354-0116

Mr. Michael Hamer
Operating Experience Coordinator
Vermont Yankee Nuclear Power Station
320 Governor Hunt Road
Vernon, VT 05354

G. Dana Bisbee, Esq.
Deputy Attorney General
33 Capitol Street
Concord, NH 03301-6937

Chief, Safety Unit
Office of the Attorney General
One Ashburton Place, 19th Floor
Boston, MA 02108

Ms. Deborah B. Katz
Box 83
Shelburne Falls, MA 01370

Mr. Raymond N. McCandless
Vermont Department of Health
Division of Occupational
and Radiological Health
108 Cherry Street
Burlington, VT 05402

Manager, Licensing
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500

Resident Inspector
Vermont Yankee Nuclear Power Station
U. S. Nuclear Regulatory Commission
P.O. Box 176
Vernon, VT 05354

Director, Massachusetts Emergency
Management Agency
ATTN: James Muckerheide
400 Worcester Rd.
Framingham, MA 01702-5399

Jonathan M. Block, Esq.
Main Street
P.O. Box 566
Putney, VT 05346-0566

Mr. John Kelly
Director, Nuclear Safety Assurance
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Gary Taylor
Chief Executive Officer
Entergy Operations
1340 Echelon Parkway
Jackson, MS 39213

cc:

Mr. John Herron
Sr. VP and Chief Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Dan Pace
Vice President, Engineering Entergy
Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Randall Edington
Vice President, Operations Support
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Director of Oversight
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. John M. Fulton
Assistant General Counsel
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Jay K. Thayer
Site Vice President
Entergy Nuclear Operations, Inc.
Vermont Yankee Nuclear Power Station
P.O. Box 0500
185 Old Ferry Road
Brattleboro, VT 05302-0500

Mr. Ken L. Graesser
BWR SRC Consultant
38832 N. Ashley Drive
Lake Villa, IL 60046

Jim Sniezek
BWR SRC Consultant
5486 Nithsdale Drive
Salisbury, MD 21801

Mr. Ron Toole
BWR SRC Consultant
605 West Horner Street
Ebensburg, PA 15931

Ms. Stacey Lousteau
Treasury Department
Entergy Services, Inc.
639 Loyola Avenue, Mail Stop L-ENT-15E
New Orleans, LA 70113

Mr. Michael Kansler
President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

Mr. Raymond Shadis
New England Coalition
Post Office Box 98
Edgecomb, ME 04556

Mr. James P. Matteau
Executive Director
Windham Regional Commission
139 Main Street, Suite 505
Brattleboro, VT 05301

Mr. William K. Sherman
Vermont Department of Public Service
112 State Street
Drawer 20
Montpelier, VT 05620-2601

ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 218
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment filed by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (the licensees) dated March 26, 2003, as supplemented by letter dated July 24, 2003, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-28 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 218, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days. The implementation of this amendment shall include the revision of the Updated Final Safety Analysis Report, Section 4.2.6 and Table 4.2.4, as described in the licensee's application dated March 26, 2003.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Darrell J. Roberts, Acting Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 29, 2004

ATTACHMENT TO LICENSE AMENDMENT NO. 218

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 218 TO FACILITY OPERATING LICENSE NO. DPR-28

ENTERGY NUCLEAR VERMONT YANKEE, LLC

AND ENTERGY NUCLEAR OPERATIONS, INC.

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated March 26, 2003, as supplemented on July 24, 2003 (References 1 and 2), Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy or the licensee) submitted a request to amend the Vermont Yankee Nuclear Power Station (VYNPS) Technical Specifications (TSs). The supplement dated July 24, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on April 29, 2003 (68 FR 22747).

The proposed amendment would revise the VYNPS TSs regarding reactor pressure vessel (RPV) fracture toughness and material surveillance requirements (SRs). Specifically, the amendment would revise the pressure-temperature (P-T) limits for the RPV as specified in TS Figures 3.6.1, 3.6.2, and 3.6.3. In addition, the amendment would delete TS 4.6.A.5, which specifies plant-specific RPV material SRs. These plant-specific SRs would be replaced by implementing the Boiling Water Reactor Vessel and Internals Project (BWRVIP) RPV integrated surveillance program (ISP). The details of the BWRVIP ISP would be added to the VYNPS Updated Final Safety Analysis Report (UFSAR).

2.0 REGULATORY EVALUATION

TS 3.6.A, "Pressure and Temperature Limitations," requires that the reactor coolant system (RCS) temperature and pressure be limited in accordance with TS Figures 3.6.1, 3.6.2, and 3.6.3. As described in the Bases for TS 3.6.A, all components in the RCS 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. During startup and shutdown, RCS pressure and temperature changes are limited so that the maximum heatup and cooldown rates are consistent with the design assumptions and the stress limits for cyclic operation. TS Figures 3.6.1, 3.6.2, and 3.6.3, also known as the P-T limit curves, define the acceptable pressure and temperature operating limits for hydrostatic pressure and leak tests (core not critical), normal operation (core not critical), and normal operation (core critical), respectively. These limits provide a margin to the brittle fracture of the RPV and piping of the reactor coolant pressure boundary (RCPB).

The purpose of the RPV material surveillance program, which is currently specified in TS 4.6.A.5, is to monitor changes in the fracture toughness properties of ferritic materials in the RPV beltline region which result from neutron irradiation and the thermal environment. Under the program, fracture toughness test data are obtained from material specimens exposed in surveillance capsules, which are withdrawn periodically from the RPV. The test results are used to update TS Figures 3.6.1, 3.6.2, and 3.6.3, as appropriate.

The construction permit for VYNPS was issued by the Atomic Energy Commission (AEC) on December 11, 1967. As discussed in Appendix F of the VYNPS UFSAR, the plant was designed and constructed based on the proposed General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereinafter referred to as "draft GDC"). The AEC published the final rule that added Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereinafter referred to as "final GDC").

Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the U.S. Nuclear Regulatory Commission (NRC or the Commission) Staff Requirements Memorandum for SECY-92-223, dated September 18, 1992 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003763736), the Commission decided not to apply the final GDC to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the final GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the final GDC were formally adopted was evaluated on a plant-specific basis, determined to be safe, and licensed by the Commission.

In order to determine the applicable regulatory acceptance criteria for the proposed amendment, the NRC staff reviewed the licensee's letters dated March 26, 2003, and July 24, 2003 (References 1 and 2). In addition, the staff reviewed NUREG-0800, "Standard Review Plan" (SRP), dated April 1996, Section 5.3.1, "Reactor Vessel Materials," and Section 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock."

Based on a review of SRP Sections 5.3.1 and 5.3.2, the NRC determined that final GDCs 14, 31, and 32 are applicable to the types of changes proposed by this amendment request. Attachment 2 to Entergy letter BVY 03-90, dated October 1, 2003 (ADAMS Accession No. ML032810447), provides a matrix of the draft GDCs versus the corresponding final GDCs. Based on Attachment 2 of letter BVY 03-90, final GDCs 14, 31, and 32 correspond to draft GDCs 9, 33, 34, 35, and 36. The draft GDC requirements are as follows:

- Draft GDC 9, "Reactor Coolant Pressure Boundary (Category A)," requires that the RCPB be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

- Draft GDC 33, "Reactor Coolant Pressure Boundary Capability (Category 4)," requires that the RCPB be accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant.

- Draft GDC 34, “Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A),” requires that the RCPB be designed to minimize the probability of rapidly propagating type failures.

- Draft GDC 35, “Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A)”, requires that service temperatures, for RCPB components constructed of ferritic material, ensure the structural integrity of such components when subjected to potential loadings.

- Draft GDC 36, “Reactor Coolant Pressure Boundary Surveillance (Category A),” requires a material surveillance program for the RPV.

Section 50.60 of 10 CFR Part 50, “Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation,” requires compliance with the fracture toughness and material surveillance program requirements set forth in Appendices G and H to 10 CFR Part 50. Compliance with the requirements of this rule, and the associated appendices, provides assurance regarding the structural integrity of the RCPB and, specifically, the RPV.

Appendix G, “Fracture Toughness Requirements,” to 10 CFR Part 50, requires that the pressure-retaining components of the RCPB that are made of ferritic materials must meet the requirements of the America Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), supplemented by the additional requirements set forth in Appendix G, for fracture toughness during system hydrostatic tests and any condition of normal operation including anticipated operational occurrences. Appendix G provides specific requirements for the RPV P-T limits. Compliance with the requirements of Appendix G is consistent with meeting the intent of draft GDCs 9, 33, 34, and 35 with regard to assuring that the RCPB acts in a non-brittle manner and that the probability of rapidly propagating failure and gross rupture of the RCPB is exceedingly low.

Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50, requires licensees to implement a RPV material surveillance program to monitor the changes in fracture toughness properties of ferritic materials in the RPV beltline region which result from exposure of these materials to neutron irradiation and the thermal environment. Compliance with the requirements of Appendix H provides assurance that the changes to the RPV materials resulting from the operational environment will be monitored, and that appropriate actions will be taken if significant changes occur in the material fracture toughness that may affect the integrity of the RPV. Compliance with the requirements of Appendix H is consistent with meeting the intent of draft GDC 36 with regard to establishing a RPV material surveillance program.

Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 2001, describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence.

RG 1.99, "Radiation Embrittlement of Reactor Vessel Material," Revision 2, dated May 1988, describes general procedures acceptable to the NRC staff for calculating RPV neutron radiation embrittlement.

In Generic Letter (GL) 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from the Technical Specifications," dated January 4, 1991, the NRC staff advised licensees that they may request a license amendment to remove the withdrawal schedule for RPV material specimens from the TSs. As discussed in this GL, the removal of this information from the TSs will not result in any loss of regulatory control because changes to the withdrawal schedule are controlled by the requirements of Appendix H to 10 CFR Part 50.

In 10 CFR 50.36, the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TSs.

In general, there are two classes of changes to TSs: (1) changes needed to reflect modifications to the design basis (TSs are derived from the design basis), and (2) changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TSs over time. In determining the acceptability of such changes, the staff interprets the requirements of 10 CFR 50.36, using as a model the accumulation of generically approved guidance in the improved Standard Technical Specifications (STS). For this review, the staff used NUREG-1433, Revision 2, "Standard Technical Specifications, General Electric Plants BWR/4."

On July 22, 1993 (58 FR 39132), the Commission published a "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (Final Policy Statement) which discussed the criteria to determine which items are required to be included in the TSs as LCOs. The criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36 (60 FR 36953).

Within this general framework, licensees may remove material from their TSs if the material is not required to be in the TSs based on the staff's interpretation of 10 CFR 50.36, including judgements about the level of detail required in the TSs. As discussed in the Final Policy Statement, the NRC staff reviews, on a case-by-case basis, whether enforceable regulatory controls are needed for the relocated material (e.g., 10 CFR 50.59). Licensees may revise the remaining TSs to adopt current improved STS format and content provided that plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards.

3.0 TECHNICAL EVALUATION

3.1 P-T Limits

Licensee's Proposed Change

As described in the licensee's letter dated March 26, 2003 (Reference 1), TS Figures 3.6.1, 3.6.2, and 3.6.3, currently contain a restriction on their use, such that the figures are only valid through the end of the current operating cycle (Cycle 23). As discussed in the NRC's Safety Evaluation (SE) for VYNPS Amendment No. 203, dated May 4, 2001 (ADAMS Accession No. ML011150218), this restriction was put in place because the RPV neutron fluence estimate used to support generation of the current P-T limit curves was not based on a methodology consistent with the guidance in RG 1.190. The proposed amendment would revise the figures such that they would be valid through the end of the current 40-year operating license. In Reference 1, the licensee states that the proposed figures were developed using an estimate of RPV neutron fluence based on a methodology that follows the guidance of RG 1.190.

The specific proposed TS changes are as follows:

- Figures 3.6.1, 3.6.2, and 3.6.3 each currently contain a statement that the figure is valid through the end of Cycle 23. That duration would be changed such that each figure would be valid through 4.46×10^8 megawatt-hours thermal (MWH(t)). In Reference 1, the licensee states that this value corresponds to an integrated plant operation of 32 effective full power years (EFPY), and is based on the expected neutron fluence over 40 years of operation at the current licensed power level, accounting for periods of downtime.

- To improve the legibility of the curves, the grid line divisions would be changed and more data would be used to plot the curves to improve resolution.

- A note would be added to Figure 3.6.2 to specify a minimum acceptable flange temperature when using local test instrumentation during flange tensioning and detensioning operations.

- The tabulation of pressure and temperature data in Figure 3.6.3 would be revised to more accurately reflect the plot of the curves.

- Conforming changes would be made to the TS Bases for Sections 3.6 and 4.6.

NRC Staff Evaluation of Neutron Fluence Calculations

As described in Reference 1, the VYNPS neutron fluence calculations were updated by General Electric (GE) using the NRC staff-approved methodology described in GE's proprietary Licensing Topical Report NEDC-32983P-A, "General Electric Methodology for Reactor Vessel Fast Neutron Flux Evaluations," Revision 1, dated December 2001. The licensee's letter dated July 24, 2003 (Reference 2), provided a GE evaluation which describes the application of the methodology for the neutron fluence calculations.

The core power level used in the GE evaluation was 1593 MWt which is the currently licensed rated thermal power level. The NRC staff notes that, in a separate license amendment request dated September 10, 2003 (ADAMS Accession No. ML032580089), the licensee has requested to increase the VYNPS maximum authorized power level to 1912 MWt. The effect of the proposed power uprate on the P-T limit curves will be evaluated in that license amendment request.

The GE evaluation consisted of two tasks. Task 1 evaluated the capsule flux and neutron flux distributions at the RPV and at the shroud for the period during which the last removed capsule (30-degree azimuthal capsule) was resident in the plant. Since the capsule was removed after shutdown in March 1983 (end of Cycle 9), the core data used was for Cycles 8 and 9. The measured and calculated results for this capsule were documented in Battelle Report BCL-585-84-3, “Examination, Testing and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the Vermont Yankee Nuclear Power Station,” dated May 15, 1984 (ADAMS Accession No. 8412130439). The measured-to-calculated ratio for the capsule flux indicated excellent agreement. The projected peak fluence value for the RPV inside surface was estimated to be in the range of 2.0×10^{17} neutrons per centimeter squared (n/cm^2) to 2.9×10^{17} n/cm^2.

Task 2 evaluated the capsule flux and neutron flux distributions at the RPV and at the shroud for a representative modern core to determine the impact of core and fuel design changes that occurred since the capsule removal. The flux evaluation was based on Cycle 21 core data. Sensitivity studies of contemplated core loadings, including the current Cycle 23, were performed and the results indicated that Cycle 23 fluxes would be bounded by Cycle 21. The peak inside surface fluence value was estimated at 2.99×10^{17} n/cm^2; virtually the same as the upper limit of the 30-degree azimuthal capsule evaluated by the South West Research Institute (SwRI) as discussed in the Battelle report. The Battelle estimate was 2.2×10^{17} n/cm^2 which is within the range of the SwRI evaluation. GE presented only the Cycle 8-9 and Cycle 21 analyses without arguments regarding cycle-to-cycle similarity or the ability to extrapolate for the entire licensing period. However, the approximate agreement of the measured flux value compared to that derived using the approved GE methodology provides sufficient confidence to find it acceptable.

The approximations, radial and axial meshes, and the cross sections that were used in the GE evaluation, meet or exceed the guidance in RG 1.190 and, therefore, are acceptable. In addition, the uncertainty used by GE for the flux calculation was within the guidance of RG 1.190.

Based on a review of the licensee's submittals, the NRC staff finds that the licensee's proposed fluence value was estimated using a methodology consistent with the guidance in RG 1.190. Therefore, the staff concludes that the peak vessel fluence value of 2.99×10^{17} n/cm^2 is acceptable for the calculation of PT-limit curves through the current license period (32 EFPY).

<u>NRC Staff Evaluation of P-T Limit Curve Changes</u>

Appendix G to 10 CFR Part 50 requires the P-T limits for an operating plant to be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the ASME Code were applied. The methodology of Appendix G to Section XI of the ASME Code postulates the existence of a sharp surface flaw in the RPV. For materials in the beltline and upper and lower head regions of the RPV, the maximum flaw size is postulated to have a depth that is equal to 1/4 of the thickness and a length equal to 1.5 times the thickness. For the case of evaluating RPV nozzles, the surface flaw is postulated to propagate parallel to the axis of the nozzle's corner radius.

The basic parameter in Appendix G to the ASME Code for calculating P-T limit curves is the stress intensity factor (K_I) which is a function of the stress state and flaw configuration. The methodology requires that the licensee determine the material fracture toughness (K_{IA} or K_{IC}) values, which vary as a function of temperature, from the RCS operating temperature, and from the adjusted reference temperature (ART) values for the limiting materials in the RPV. Thus, the critical locations in the RPV beltline and head regions are the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations, which correspond to the points of the crack tips if the flaws are initiated and grown from the inside and outside surfaces of the vessel, respectively. RG 1.99, Revision 2, includes methods for adjusting material ART values in the beltline region of the RPV, where the effects of neutron irradiation may induce an increased level of embrittlement in the materials.

The methodology of Appendix G to the ASME Code requires that P-T curves must 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 RCS leak rate or hydrostatic pressure tests are performed. Table 1 to 10 CFR Part 50, Appendix G provides the staff's criteria for meeting the P-T limit requirements of Appendix G to Section XI of the ASME Code and the minimum temperature requirements of the rule for bolting up the vessel during normal and pressure testing operations.

Appendix G to 10 CFR Part 50 requires that the licensee determine material ART values. A material's ART value is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term per RG 1.99, Revision 2.

As discussed in Section 3.2.5 of Attachment 1 of Reference 1, the licensee states that the VYNPS P-T limit curves evaluation (documented in Attachment 2 of Reference 1) demonstrates that the limiting beltline component (RPV plate 1-14) remained the same, and the ART values (at 32 EFPY or 4.46×10^8 MW-hours) calculated in accordance with RG 1.99, Revision 2, remain bounded by the values used to develop the current P-T curves. In Section 3.2.5 of Attachment 1 of Reference 1, the licensee also states that the equivalent fluence, when compared to the updated fast fluence of 2.99×10^{17} n/cm^2, remains very conservative, as demonstrated in Attachment 2 of Reference 1.

Tables 2-1 and 2-2 of Attachment 2 to Reference 1, provided the ART values for the limiting material as 57 °F at 1/4T fluence (2.20×10^{17} n/cm^2) and 48 °F at 3/4T fluence (1.20×10^{17} n/cm^2). Section 2.0 of Attachment 2 to Reference 1, states that for purposes of determining the P-T curves for the vessel core region material, VYNPS has elected to maintain the more conservative ART values previously used by VYNPS (89 °F at the 1/4T point and 73 °F at the 3/4T point). The licensee's submittal states that, based on RG 1.99, Revision 2, lower values of ART could have been used.

The NRC staff's assessment included an independent calculation of the ART values for both the 1/4T and 3/4T locations of the VYNPS RPV beltline regions based on the revised 32 EFPY neutron fluence specified in the submittal for VYNPS. The staff confirmed, using the peak vessel fluence value of 2.99×10^{17} n/cm^2 and the methodology of RG 1.99, Revision 2, that the limiting beltline material was the RPV plate 1-14 with an ART of 57 °F at the 1/4T location and 48 °F at the 3/4T location. Previously, the P-T limit curves were based on a peak vessel fluence value of 1.24×10^{18} n/cm^2 resulting in the limiting material (RPV plate 1-14) having an ART of 89 °F at the 1/4T location and 73 °F at the 3/4T location. Since the staff has confirmed that the previous ART values bound the revised ART values, calculated using the fluence values determined with the guidance of RG 1.190, the staff agrees that the previous ART values and P-T limit curves can be used for 32 EFPY of operation for VYNPS.

In Section 3.2.3 of Attachment 1 to Reference 1, the licensee states that the updated P-T limit curves are based, in part, on the application of ASME Code Case N-640. Pursuant to 10 CFR 50.12 and by letter dated April 16, 2001 (ADAMS Accession No. ML010650232), the NRC granted an exemption to allow VYNPS to deviate from the requirements of Appendix G to 10 CFR Part 50 in the use of this alternative method. The NRC agrees with the licensee that the application of ASME Code Case N-640 is acceptable, as approved by NRC letter dated April 16, 2001. As mentioned above, the licensee will retain the current and more conservative P-T limit curves, approved in VYNPS Amendment No. 203, that used the application of ASME Code Case N-640, through the current license period.

TS Figures 3.6.1, 3.6.2, and 3.6.3 would extend their applicability to 4.46×10^8 MWH(t), which corresponds to 32 EFPY of plant operation at the current licensed power level. The NRC staff finds this acceptable because the TS figures are based on the 32 EFPY fluence calculated using the guidance of RG 1.190 at the current licensed power level, and accounting for periods of downtime as stated in Reference 1. However, these P-T limit curves do not account for the proposed extended power uprate (EPU) amendment submitted by the licensee on September 10, 2003, which is currently being reviewed by the NRC. Any changes necessary to the P-T limit curves based on the EPU conditions will be assessed during the evaluation of the EPU submittal.

As shown in Attachment 5 to Reference 1, a note would be added to TS Figure 3.6.2 to specify requirements for a minimum temperature when using local test instrumentation during flange tensioning and detensioning operations. The proposed new note would state:

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

The use of this instrumentation is limited to the condition when the RPV is vented and the RPV fluid level is below the flange region to ensure that the RPV cannot be pressurized while using test instrumentation. Section 3.2.2 of Attachment 1 to Reference 1 states that since acceptable test instrumentation will be more accurate than permanent temperature instrumentation (+/-10 °F), a limit of ≥72 °F may be established when using test instrumentation. Therefore, a 72 °F limit for test instrumentation corresponds to an 80 °F limit for permanent temperature instrumentation taking instrumentation uncertainties into account. The analytical limit for head bolt-up is 70 °F without instrument uncertainty, as stated in current TS 3.6.A. The NRC staff finds the use of more accurate test instrumentation, when the RPV is vented and the fluid level is below the flange region, is acceptable to ensure temperature limits are adhered to.

Other changes were made to TS Figures 3.6.1, 3.6.2, and 3.6.3 which included identifying 100 pounds per square inch (psi) increments rather than 200 psi increments for improved resolution of the graphs, and minor changes to the tabulation of pressure and temperature data on Figure 3.6.3 to more accurately reflect the plot of the curves, which have remained unchanged. The NRC staff finds that these changes will enhance the readability and accuracy of the figures. Therefore, the proposed changes are acceptable.

The licensee has also proposed conforming changes to the TS Bases for Sections 3.6 and 4.6. The NRC staff has no objections to these changes.

NRC Staff Conclusion

The NRC staff concludes that the proposed P-T limit curves for VYNPS are consistent with the requirements of Appendix G to Section XI of the ASME Code, as modified by Code Case N-640, and Appendix G to 10 CFR Part 50. The proposed P-T limit curves also meet the requirements of RG 1.99, Revision 2 for the calculation of the ART values. Hence, the proposed P-T limit curves may be incorporated into the VYNPS TSs and shall be valid through 32 EFPY (4.46×10^8 MWH(t)) of operation at the current licensed power level without EPU conditions.

3.2 RPV Material Surveillance Program

Licensee’s Proposed Change

The details regarding the VYNPS plant-specific RPV material SRs are currently described in TS 4.6.A.5. TS 4.6.A.5 states that:

> 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.

As described in the licensee's letter dated March 26, 2003 (Reference 1), the proposed amendment would delete TS 4.6.A.5, and details regarding the BWRVIP ISP (which would be adopted in place of the current plant-specific requirements) would be added to the UFSAR. In addition, conforming changes would be made to the TS Bases for Sections 3.6 and 4.6.

<u>NRC Staff Evaluation of Adoption of the BWRVIP</u>

The BWRVIP ISP was submitted for NRC staff review and approval in proprietary topical reports BWRVIP-78, “BWR Vessel and Internals Project, BWR Integrated Surveillance Program Plan,” dated December 22, 1999, and BWRVIP-86, “BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan” dated December 22, 2000. Additional information necessary to establish the technical basis for, and proposed implementation of, the BWRVIP ISP was provided in letters from the BWRVIP to the NRC dated December 15, 2000, and May 30, 2001. The NRC staff approved the proposed BWRVIP ISP in a SE dated February 1, 2002 (ADAMS Accession No. ML020380691). However, the NRC staff’s SE required that plant-specific information be provided by BWR licensees who wish to implement the BWRVIP ISP for their facilities. In Reference 1, the licensee addressed the VYNPS plant-specific information required in the NRC staff’s February 1, 2002, BWRVIP ISP SE.

As discussed in Section 2.0 of this SE, nuclear power plant licensees are required by Appendix H to 10 CFR Part 50 to implement RPV surveillance programs to monitor changes in the fracture toughness properties of ferritic materials in the RPV beltline region which result from exposure of these materials to neutron irradiation and the thermal environment. Two specific alternatives are provided with regard to the design of a facility’s RPV surveillance program which may be used to address the requirements of Appendix H to 10 CFR Part 50.

The first alternative is the implementation of a plant-specific RPV surveillance program consistent with the requirements of American Society for Testing and Materials (ASTM) Standard Practice E 185, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.” In the design of a plant-specific RPV surveillance program, a licensee may use the edition of ASTM Standard Practice E 185, which was current on the issue date of the ASME Code to which the RPV was purchased, or later editions through the 1982 edition.

The second alternative provided in Appendix H to 10 CFR Part 50 is the implementation of an ISP. An ISP is defined in Appendix H to 10 CFR Part 50 as occurring when, “the representative materials chosen for surveillance for a reactor are irradiated in one or more other reactors that have similar design and operating features.” Five specific criteria are stated in Section III.C.1 of Appendix H to 10 CFR Part 50 which must be met to support approval of an ISP:

a. The reactor in which the materials will be irradiated and the reactor for which the materials are being irradiated must have sufficiently similar design and operating features to permit accurate comparisons of the predicted amount of radiation damage.

b. Each reactor must have an adequate dosimetry program.

c. There must be adequate arrangement for data sharing between plants.

d. There must be a contingency plan to assure that the surveillance program for each reactor will not be jeopardized by operation at reduced power level or by an extended outage of another reactor from which data are expected.

e. There must be substantial advantages to be gained, such as reduced power outages or reduced personnel exposure to radiation, as a direct result of not requiring surveillance capsules in all reactors in the set.

As noted above, the NRC staff approved the proposed BWRVIP ISP in a SE dated February 1, 2002. All of the criteria cited above for approval of the ISP were evaluated by the NRC staff in Section 4.2 of the staff's SE. The SE concluded that 10 CFR Part 50, Appendix H, Criteria III.C.1.a, IIII.C.1.d, and IIII.C.1.e were adequately addressed by the BWRVIP. However, the SE concluded Appendix H, Criteria III.C.1.b and III.C.1.c need to be addressed on a plant-specific basis as discussed below.

Based on review of the criterion related to the need for an adequate dosimetry program (i.e., Appendix H, Criterion III.C.1.b), Section 4.2 of the NRC staff's SE dated February 1, 2002, concluded that:

> [O]ne condition of the NRC's approval of the ISP is that an individual BWR licensee who wishes to participate in the BWR ISP shall provide, for NRC staff review and approval, information which defines how they will determine RPV and/or surveillance capsule fluences based on the dosimetry data which will become available for its facility. The staff will require that this information be submitted concurrently with each licensee's submittal to replace their existing plant-specific surveillance program with BWR ISP as part of their facility's licensing basis. The information submitted must be sufficient for the staff to determine that:
>
> (1) RPV and surveillance capsule fluences will be established based on the use of an NRC-approved fluence methodology that will provide acceptable results based on the available dosimetry data, and
>
> (2) if one "best estimate" methodology is used to determine the neutron fluence values for a licensee's RPV and one or more different methodologies are used to establish the neutron fluence values for the ISP surveillance capsules which "represent" that RPV in the ISP, the results of these differing methodologies are compatible (i.e., within acceptable levels of uncertainty for each calculation).

Based on review of the criterion related to data sharing between plants (i.e., Appendix H, Criterion III.C.1.c), Section 4.2 of the NRC staff's SE dated February 1, 2002, concluded that:

> [T]he NRC recognizes that BWRVIP processes have been demonstrated in other programs to be sufficient for establishing methods to share data between BWR facilities. The staff accepts the commitment by the BWRVIP in the BWRVIP-78 and BWRVIP-86 reports to develop a "program plan to manage data sharing...in the implementation phase of the ISP." The NRC staff, however, would also note that by the incorporation of the ISP into the licensing basis for each participating BWR facility, each licensee is further responsible for ensuring that they acquire and evaluate in a timely manner all relevant ISP data which may affect RPV integrity evaluations for their facility.

In Attachment 4 to Reference 1, the licensee submitted proposed changes to VYNPS UFSAR Section 4.2.6 which would be implemented as part of the proposed relocation of the RPV material SRs from TS 4.6.A.5 to the UFSAR. The proposed UFSAR revision states, in part, that:

> 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."

The NRC staff concludes that the inclusion of this statement in the VYNPS UFSAR is sufficient to address the plant-specific information pertaining to 10 CFR Part 50, Appendix H, Criterion III.C.1.b, as discussed above. Specifically, regarding item (1), the licensee's use of a methodology for determining the VYNPS RPV neutron fluence values, in accordance with RG 1.190, will provide acceptable results based upon the available dosimetry data. Regarding item (2), RPV surveillance capsules tested under the BWRVIP ISP will have their fluences determined by the use of a methodology which is consistent with the attributes of RG 1.190. The NRC staff concludes that any two (or more) different fluence methodologies will provide "compatible" (as defined in the NRC staff's SE dated February 1, 2002) results provided that the best estimate fluence values are within each other's uncertainty bounds.

The proposed revision to VYNPS UFSAR Section 4.2.6 (in Attachment 4 to Reference 1) also states, in part, that:

> Vermont Yankee [VY] is a participant in the Boiling Water Reactor Vessel and Internal 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 10 CFR 50, "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 10 CFR 50, 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 NRC staff concludes that the inclusion of this statement in the VYNPS UFSAR is sufficient to address the plant-specific information pertaining to 10 CFR Part 50, Appendix H, Criterion III.C.1.c, as discussed above. Specifically, the proposed UFSAR revision adequately addresses the data sharing aspect of the ISP in the VYNPS licensing basis and the need for the licensee to evaluate relevant ISP data which may affect RPV integrity evaluations for VYNPS.

Based on the above, the NRC staff concludes that the information provided in the proposed revision of the VYNPS UFSAR is adequate to document the licensee's intent to appropriately implement the BWRVIP ISP as the method for demonstrating the compliance of VYNPS with the requirements of Appendix H to 10 CFR Part 50.

NRC Evaluation of Proposed Deletion of TS 4.6.A.5

As discussed above, the proposed amendment would delete TS 4.6.A.5, which provides the RPV material specimen withdrawal schedule and associated SRs. In addition, conforming changes would be made to the TS Bases for Sections 3.6 and 4.6.

Section III.B.3 of Appendix H to 10 CFR Part 50 requires that for RPV material specimens, a proposed withdrawal schedule, along with a technical justification, be submitted to the NRC for approval prior to implementation. As discussed in GL 91-01, the placement of the withdrawal schedule in the TSs duplicates the regulatory control on changes to this schedule that have been established by Appendix H. Based on this consideration, GL 91-01 provides guidance for removal of the schedule for the withdrawal of the RPV material specimens from the TSs. The main provisions of GL 91-01 are summarized as follows:

1. If the SRs in the TSs on P-T limits specify that the results of the specimen examinations be used to update the P-T limits, this TS requirement shall be retained.

2. The TS Bases section related to the TS for the withdrawal schedule should be updated, as needed, to reflect removal of the withdrawal schedule from the TSs.

3. Licensees should commit to include the NRC-approved withdrawal schedule in the next revision of the UFSAR.

Regarding the first provision of GL 91-01, TS 4.6.A.5 currently contains a statement that "[t]he results shall be used to reassess material properties and update Figures 3.6.1, 3.6.2, and 3.6.3, as appropriate." However, the licensee has proposed to delete TS 4.6.A.5 in its entirety as part of deleting the withdrawal schedule. As discussed in Section 3.1 of Attachment 1 to Reference 1, as part of the proposed implementation of the BWRVIP, no further surveillance capsules will be removed and tested from the VYNPS RPV since VYNPS is not a host ISP plant for providing surveillance capsules. As indicated in the test matrix of BWRVIP-86-A, RPV weld and plate surveillance materials from Susquehanna Unit 1 have been selected from among the existing plant surveillance programs to represent the corresponding limiting plate and weld material in the VYNPS RPV. The two remaining capsules will continue to reside in the VYNPS RPV in case they are needed in the future as a contingency. As discussed in GL 91-01, the intent of the first provision was to ensure that the surveillance specimens are withdrawn at the proper time. Since VYNPS is not required to withdraw any further capsules as part of the implementation of the BWRVIP, the staff concludes that the guidance in the first provision of GL 91-01 is not applicable to VYNPS. In addition, the NRC staff notes that the SRs for the P-T limits in the STS (NUREG-1433, Revision 2, TS 3.4.10) also do not contain similar requirements related to updating the P-T limits based on specimen examination results.

Regarding the second provision of GL 91-01, the license has proposed to revise the TS Bases for Sections 3.6 and 4.6 to delete the following statement:

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

The NRC staff finds that the proposed deletion is consistent with the second provision of GL 91-01 and, therefore, is acceptable.

Regarding the third provision of GL 91-01, the VYNPS UFSAR Table 4.2.4 currently includes the withdrawal schedule. As shown in Attachment 4 to Reference 1, the licensee has proposed to revise this table consistent with implementation of the BWRVIP ISP to change the withdrawal schedule for the second capsule from “30 years” to “Standby.” The third capsule is currently also shown as a “Standby” capsule. The NRC staff finds that the proposed revisions to UFSAR Table 4.2.4 are consistent with the implementation of the BWRIP ISP, and are acceptable. Since the licensee has committed to incorporate the UFSAR Table 4.2.4 revisions as part of implementation of the proposed amendment, the staff finds that the third provision of GL 91-01 has been met.

Based on the above, the NRC staff finds that the proposed deletion of TS 4.6.A.5 is consistent with the applicable guidance of GL 91-01. Since the removal of the information from the TSs will not result in any loss of regulatory control because changes to the withdrawal schedule are controlled by the requirements of Appendix H to 10 CFR Part 50, the staff concludes that the proposed deletion is acceptable. In addition, the NRC staff notes that the SRs for the P-T limits in the STS (NUREG-1433, Revision 2, TS 3.4.10) do not include the withdrawal schedule.

NRC Staff Conclusion

The NRC staff concludes that the BWRVIP ISP can be implemented for VYNPS as the basis for demonstrating the facility’s continued compliance with the requirements of Appendix H to 10 CFR Part 50. As part of the implementation and documentation of the licensee’s intent to utilize the BWRVIP ISP for this purpose, the licensee shall modify the VYNPS UFSAR Section 4.2.6 and Table 4.2.4, as described in Reference 1. In addition, the NRC staff concludes that the proposed deletion of TS 4.6.A.5 is acceptable.

3.3 Technical Evaluation Conclusion

Based on the considerations in SE Sections 3.1 and 3.2, the NRC staff concludes that the proposed amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Vermont State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendment involves no significant increase in amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 22747). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1) Letter from M. A. Balduzzi (Entergy) to NRC, dated March 26, 2003, "Technical Specification Proposed Change No. 258, RPV Fracture Toughness and Material Surveillance Requirements" (ADAMS Accession No. ML030870156).

2) Letter from J. K. Thayer (Entergy) to NRC, dated July 24, 2003, "Technical Specification Proposed Change No. 258, RPV Fracture Toughness and Material Surveillance Requirements, Additional Information - Neutron Flux Evaluation" (ADAMS Accession Nos. ML032170915 and ML032170921).

Principal Contributors: L. Lois
J. Honcharik
C. Long
R. Ennis

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