VERMONT YANKEE
NUCLEAR POWER CORPORATION

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February 13, 2001
BVY 01-13

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)
Supplement to Technical Specification Proposed Change No. 244
Withdrawal of Exemption Request to use Code Case N-588

On December 19, 2000, Vermont Yankee (VY) submitted to the Staff a combined Proposed Change to the Technical Specifications and Exemption Request[1]. In that submittal, VY requested an exemption from the requirements of 10CFR50, Appendix G, to allow the use of ASME Code Cases N-588 and N-640 as the basis for the revised P/T curves. As a result of additional reviews, it has been determined that the use of ASME Code Case N-588 is not warranted. Accordingly, VY hereby withdrawals the Exemption Request to use Code Case N-588.

As a result of this change, corresponding minor changes are necessary to the Technical Specification Bases pages submitted in our Proposed Change. Attachment 1 provides a mark-up of the proposed Technical Specification Bases pages Insert from our original submittal. Attachment 2 is the retyped Technical Specification Bases pages.

It is noted that the determination of no significant hazards consideration that accompanied our Proposed Technical Specification Change contained a reference to Code Case N-588. However, since the requirements of 10CFR50, Appendix G are satisfied by utilizing the conservatively bounding 1995 ASME Code, Section XI, Appendix G (including the Summer 1996 Addenda) in lieu of Code Case N-588, the conclusions of the determination of no significant hazards remain unchanged.

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1 Reference Vermont Yankee Nuclear Power Corporation letter to the USNRC, BVY 00-113, "Technical Specification Proposed Change No. 244, Revised P/T Curves and Exemption Request to use Code Cases N-588 and N-640," dated December 19, 2000.

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

Sincerely,

VERMONT YANKEE NUCLEAR POWER CORPORATION

Don M. Leach
Vice President, Engineering

STATE OF VERMONT)
)ss
WINDHAM COUNTY)

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

T.B. Silko

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

Attachments

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

Attachment 1

Vermont Yankee Nuclear Power Station

Supplement to Proposed Technical Specification Change No. 244

Revised P/T Curves

Marked-up Version of the Current Technical Specifications Bases Pages

Insert 1

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

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

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

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

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

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

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

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

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

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

Attachment 2

Vermont Yankee Nuclear Power Station

Supplement to Proposed Technical Specification Change No. 244

Revised P/T Curves

Retyped Technical Specification Bases Pages

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.

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

BASES: 3.6 and 4.6 (Cont'd)

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