Entergy

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September 25, 2003
BVY 03-83

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 Fourth-Interval Inservice Inspection (ISI) Program Plan - Submittal of Relief Request ISI-06

On April 1, 2003, Vermont Yankee Nuclear Power Station (VY) submitted to the NRC a revised Inservice Testing (ISI) Program[1] as required by 10CFR50.55a(a)(3)(i). The subject submittal contained a number of relief requests for NRC review and approval, however, Relief Request ISI-06 was identified within that submittal as being on hold. Accordingly, attached for your review and approval is Relief Request ISI-06 "ASME Section XI, Class 1, Examination Category B-A, Code Item No. B1.11, Circumferential Shell Welds."

Attachment 1 identifies the commitments contained within this letter. Attachment 2 contains Relief Request ISI-06.

VY requests NRC review and approval in time to support our upcoming outage scheduled for April 2004.

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

Sincerely,

James M. DeVincintis
Manager, Licensing

Attachments

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

[1] Reference VY Letter to USNRC, dated April 1, 2003, BVY 03-28, "Fourth-Interval Inservice Inspection Program Plan and Fourth-Interval Inservice Inspection Pressure Test Program and Request for Approval of ISI Relief Requests."

A047

Attachment 1

Vermont Yankee Nuclear Power Station

Supplement to Fourth-Interval Inservice Inspection (ISI) Program Plan – Submittal of Relief Request ISI-06

List of Commitments

SUMMARY OF VERMONT YANKEE COMMITMENTS

BVY NO.: 03-83

The following table identifies commitments made in this document by Vermont Yankee. Any other actions discussed in the submittal represent intended or planned actions by Vermont Yankee. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Licensing Manager of any questions regarding this document or any associated commitments.

COMMITMENT	COMMITTED DATE OR "OUTAGE"
NONE	N/A

Attachment 2

Vermont Yankee Nuclear Power Station

Supplement to Fourth-Interval Inservice Inspection (ISI) Program Plan – Submittal of Relief Request ISI-06

Relief Request ISI-06

LICENSEE/UTILITY NAME – Entergy Nuclear Northeast
PLANT NAME, UNIT – Vermont Yankee
10-YEAR INTERVAL – Fourth Interval
REQUEST FOR RELIEF No. ISI-06

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. **ASME Code Component(s) Affected**

   ASME Section XI, Class 1, Examination Category B-A, Code Item No. B1.11, Circumferential Shell Welds

2. **Applicable Code Edition and Addenda**

   1998 Edition with Addenda through 2000

3. **Applicable Code Requirements**

   Perform a volumetric examination of RPV shell circumferential welds in accordance with the requirements of Item B1.11 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of the 1998 Edition with addenda through 2000 of ASME, Section XI.

4. **Reason for Request**

   To avoid unnecessary inspections and to conserve radiological dose, while still maintaining an acceptable level of quality and safety for examination of the affected welds.

5. **Proposed Alternative**

   In accordance with the Final Safety Evaluation of BWRVIP-05 (Reference 6) and the guidance of GL 98-05 (Reference 7), perform a volumetric examination of essentially 100% of the accessible RPV axial welds to the extent possible and the incidental portions of circumferential welds where they intersect the axial welds during the First Period of the Fourth ISI interval. Additionally, in accordance with BWRVIP-05 (Reference 1), Section 9.1 the relevant flaw size of concern is one greater than 0.200" in depth emanating from the clad-to-base metal interface.

   Basis for Use

   Vermont Yankee complied with 10CFR50.55a(g)(6)(ii)(A)(2) required augmented examination of RPV circumferential and axial shell welds during RFO 19 (1996) in the First Period of the Third ISI interval. Vermont Yankee provided the results of these examinations to the NRC (Reference 2). The NRC issued a safety evaluation of the Vermont Yankee augmented examination results (Reference 3) and a letter authorizing use of the alternative inspection coverage (Reference 9). The RFO 19 scope of examinations included essentially all of the accessible circumferential and axial shell welds.

   The examinations detected seven indications in the vessel, one of which exceeded the ASME Section XI, IWB-3511 acceptance criteria. The flaw was accepted in an NRC Safety Evaluation (Reference 3). Subsequently, Vermont Yankee sought relief from successive examinations of this

flaw (Reference 8) under the provisions of BWRVIP-05, Section 9.2.1.2, with the stipulations that the flaw be acceptable for continued service in accordance with IWB-3600 and the flaw be demonstrated acceptable for the intended service life of the vessel. The Staff accepted BWRVIP-05's recommendations for successive additional examinations in the Final Safety Evaluation of BWRVIP-05 (Reference 6) and the NRC granted Vermont Yankee's alternative (Reference 10).

The general basis for the use of the alternative examination criteria is found in BWRVIP-05, Sections 3 through 8 (Reference 1), and in the BWRVIP's response to the NRC's Request for Additional Information on BWRVIP-05 (Reference 5), and for the most part is not repeated here. It should be noted that the NRC did not take any exceptions to the acceptance criteria of BWRVIP-05, Section 9.1 in the associated Request for Additional Information, the Safety Evaluation, or in Generic Letter 98-05 (References 4, 6, and 7).

Vermont Yankee complies with GL 98-05 (Reference 7) provisions for seeking relief from the requirements of examination of BWR RPV circumferential shell welds as recommended in BWRVIP-05 (Reference 1). GL 98-05 states that licensees may request relief from the examination of the RPV circumferential shell welds by demonstrating the following:

1. At the expiration of the license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the staff's July 28, 1998, safety evaluation related to BWRVIP-05 (Reference 6).

2. Licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 28, 1998, safety evaluation related to BWRVIP-05.

Vermont Yankee has performed an assessment of the above GL 98-05 provisions, as discussed below.

<u>VERMONT YANKEE ASSESSMENT OF GL 98-05 PROVISIONS</u>

1. The following demonstrates that at the expiration of the Vermont Yankee Operating License in 2012, Vermont Yankee RPV circumferential shell welds will continue to satisfy limiting conditional failure probability for the circumferential welds stated in the Staff's July 28, 1998 safety evaluation (Reference 6).

   a. <u>Neutron Fluence/Embrittlement</u>

      BWRVIP-05 stated, "*Embrittlement issues are addressed in 10CFR50 Appendix G through requirements associated with upper shelf energy (USE) and the reference temperature of nil-ductility transition (RT_{NDT}). In order to account for the effects of embrittlement, adjusted reference temperatures (ARTs), defined as the initial RT_{NDT} plus the irradiation shift for fluence, are determined. It is possible that ARTs may result in pressure-temperature testing criteria that are difficult to meet due to increased temperature requirements. However, due to low BWR fluence, an unacceptable ART will not be reached, even when extended life is planned.*" Also, the report states that, "*In addition to increasing RT_{NDT} the USE of low alloy steel materials decreases with neutron exposure. However, for the relatively low fluence BWR, maintaining a USE above 50 ft-lbs is not a concern. Also, Code margins required by Appendix G are satisfied at USE values as low as 35 ft-lbs and thus are not a safety concern. Based on the above, it can be seen that although irradiation embrittlement of materials can*

be a significant concern, its effect is minimal for the relatively low fluence environment of a BWR."

As documented in Reference 11, Vermont Yankee projects that the decrease in USE data for the end of the current operating license is well within the limits provided in NEDO-32205, the equivalent margins topical report applicable to Vermont Yankee. This topical report follows the methods provided in Code Case N-512 and was accepted by the NRC. The projected decrease in limiting plate/weld projected USE decrease will be less than 13.5% / 7.4% and well below the 21% / 34% allowable decrease from NEDO-32205. Therefore, Vermont Yankee 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.

b. Probabilistic Fracture Mechanics (PFM) Analysis

Although BWRVIP-05 provides a technical basis for this relief, an independent NRC risk informed assessment of the analysis contained in the BWRVIP-05 report was conducted (Reference 4). The independent NRC assessment used the FAVOR code to perform probabilistic fracture mechanics (PFM) analysis to estimate RPV failure probabilities. Three key assumptions in the PFM analysis are: the neutron fluence was estimated to be end-of-license mean fluence, the chemistry values are mean values based on vessel types, and the potential for beyond design basis events were considered.

The following is a statement contained in the "Executive Summary" of the "NRC Staff Final Safety Evaluation of BWRVIP-05 Report (Reference 6). *"It should be noted that the failure frequency for axial welds cited above are relatively high, but that there are known conservatisms in these estimates. For example, these analyses were based on the assumption that the flaws in axial weld with the limiting material properties and chemistry are all located at the inside surface of the BWR RPV and at the location of peak end-of-license (EOL) azimuth fluence. Since flaws are distributed throughout the weld and EOL neutron fluence will not occur for many years, the staff has concluded that the present RPV failure frequency is substantially below that reported by the BWRVIP, and independently calculated by the staff, and is not a near-term safety concern."*

The following information is provided to show the conservatism of the NRC analysis with respect to the Vermont Yankee plant. Changes in RT_{NDT} may be used as one of the means for monitoring radiation embrittlement of reactor vessel materials. For plants with RPVs fabricated by Chicago Bridge and Iron (CBI), the mean end-of-license neutron fluence for circumferential welds used in the NRC staff and BWRVIP Limiting Plant-Specific Analysis (32 EFPY), Table 2.6-4 of the Safety Evaluation for BWRVIP-05, was 5.1×10^{18} n/cm^2. However the peak end of license (EOL) fast fluence of 2.99×10^{17} n/cm^2 (E>1.0 MeV) for Vermont Yankee's entire belt-line is far less than that used in the NRC analysis. Therefore there is significant conservatism with regard to the effect of fluence on embrittlement in the already low circumferential weld failure probabilities as related to the Vermont Yankee RPV.

The Table below shows a comparison between the NRC Final Evaluation of the BWRVIP-05 Limiting Plant Specific Analysis data and Vermont Yankee specific data for weld chemistry factor (CF) and initial RT_{NDT}.

Vermont Yankee RPV Shell Weld Information Bounding Circumferential Weld (Reference 11)		
Parameter Description	**Vermont Yankee Shell Bounding Beltline 32 EFPY Bounding Comparative Parameters (Bounding Circ. Weld)**	**USNRC Limiting 32 EFPY Bounding CBI Vessel Parameters SER Table 2.6-4**
Neutron fluence at the end of the requested relief period (Peak Surface Fluence Entire Beltline)	2.99×10^{17} n/cm^2	5.1×10^{18} n/cm^2
Initial (unirradiated) reference temperature (RT_{NDT}), °F	-70	-65
Weld Chemistry factor (CF), °F	54	134.9
Weld Copper content %	.04	0.10
Weld Nickel content %	1	0.99
Increase in reference temperature (ΔRT_{NDT}), °F = CF*(Fluence/10^19)^(0.28-0.1*LOG(Fluence/10^19))	11.8 F	109.5
Mean adjusted reference temperature (ART), °F = $RT_{NDT(u)} + \Delta RT_{NDT}$	-58.2	44.5

As shown above, the impact of irradiation results in lower plant-specific mean RT_{NDT} for the Vermont Yankee circumferential weld material, as compared to that for any of the Staff's plant-specific analyses that were performed for the CBI fabricated RPVs with the highest adjusted reference temperatures. Therefore, based on plant specific data, there is a lower conditional probability of failure for circumferential welds at Vermont Yankee than that stated in the NRC's Final Safety Evaluation of the BWRVIP-05.

2. The following demonstrates that Vermont Yankee has implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 28, 1998 safety evaluation.

   At an industry meeting on August 8, 1997, the NRC indicated that the potential for, and consequences of, non-design basis events not addressed in the BWRVIP-05 report should be considered. Later, in a Request for Additional Information (RAI) to the BWRVIP, the NRC requested that the BWRVIP evaluate the potential for non-design basis cold over-pressure transients (Reference 4) and responded to in BWRVIP letter to NRC dated December 18, 1997 (Reference 5). The NRC also considered beyond design basis events, such as low temperature over-pressure (LTOP) events in their PFM analysis. In the BWRVIP responses to the RAI the total probability of an occurrence of cold overpressure for other than BWR-4s was reported as 9E-4. It was concluded that it is highly unlikely that a BWR would experience a cold over-pressure transient. In fact, for a BWR to experience such an event would generally require several operator errors. The NRC described several types of events that could be precursors to BWR RPV cold over-pressure transients. These were identified as precursors because no cold over-pressure event has occurred at a U.S. BWR. Also, the NRC identified one actual cold over-pressure event that occurred during shutdown at a non-U.S. BWR. This event apparently included several operational errors that resulted in a maximum RPV pressure of 1150 psi with a temperature range of 79° to 88° F.

The following addresses the high-pressure injection sources, administrative controls, and operator training regarding a cold overpressure event for the Vermont Yankee plant.

a. Review of Potential High Pressure Injection Sources

1. High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) Systems

   The HPCI and RCIC systems use steam driven turbines to pump cold water into the Vermont Yankee vessel. During reactor cold shutdown conditions, there is no steam available to operate these systems, making a cold over pressurization event impossible as the result of operation of these systems.

2. Feedwater / Condensate Systems

   The feedwater / condensate systems are potential high-pressure injection water sources into the reactor vessel. The condensate pumps provide water source to the reactor feed pumps. The feed pumps provide water to the vessel. The normal reactor feed pump discharge pressure is approximately 1300 psig (with condensate pumps running) and the shutoff head pressure is approximately 1680 psig (assuming condensate pumps at shutoff head pressure). The normal condensate pump discharge pressure is approximately 420 psig and the shutoff head is approximately 550 psig. A system design feature of the reactor feed pumps is an automatic trip of all feed pumps on high vessel water level.

   The startup procedure requires monitoring of reactor vessel temperatures and pressures. The condensate and feed water pumps are used to control vessel level during startup. The reactor head vents are not closed until the coolant temperature is greater than 200°F. This administrative action for head vent closure serves as a mechanism to reduce the likelihood of pressurization above 250 psig. When shutting down, Vermont Yankee procedures require securing RFPs in sequence depending on the reactor power levels. Monitoring of reactor temperature, pressure, and cool down rates, are prescribed in procedures and Technical Specifications. During refueling outages the feedwater lines are isolated by closing block valves. At low power (approximately 5 to 10%), the lines are secured by removing both main feed regulating valves from service and manually controlling feed water with the auxiliary feed regulating valve.

   Reactor over pressurization by the feedwater/condensate systems is very unlikely since strict controls on temperature and pressure are imposed below 450°F and the capacity of the systems to inject water is limited by using the auxiliary feed regulating valve. Any unexpected change in reactor water level would allow for operator action.

   Therefore, these systems do not present a significant potential for over pressurization.

3. Standby Liquid Control System

   The Standby Liquid Control System (SLC) is a potential source of high-pressure water into the RPV during cold shutdown conditions. A key lock switch in the control room is required to operate the system since it does not have any automatic start capabilities. As a result, operation of SLC is a deliberate act strictly controlled

by plant procedures and training. Even if the system was activated, the maximum SLC flow is 40 gpm, a rate that would allow time to control RPV pressure. Therefore, this system does not present a significant potential for over pressurization.

4. The Low-Pressure Coolant Injection, Core Spray, and Residual Heat Removal Systems

The Low-Pressure Coolant Injection, Core Spray, and Residual Heat Removal systems' inadvertent operation do not present a significant potential for over pressurization. The pressure-temperature curves for the Vermont Yankee RPV permit cooling water pressures up to 250 psig over the temperature range of 80 to 110°F and rapidly increase to 810 psig at 110°F. Shutoff head pressures for Core Spray and RHR pumps are approximately 300 psig. Since these shut-off head pressures could pressurize the reactor vessel to greater than 250 psig, a review of LPCI, RHR and Core Spray operating modes is provided.

The reactor vessel is not likely to pressurize under LOCA conditions. The LPCI and Core Spray would be actuated with the reactor in a depressurized but metal hot condition during a LOCA from full power, which rapidly depressurizes the reactor vessel to a depressurized but metal hot condition. The metal temperature lags pressure substantially and would be greater than 100°F.

In an emergency, following a loss of shutdown cooling, an alternate shutdown-cooling mode is permitted. This mode of shutdown cooling uses available pump(s) to circulate water from the torus, using a flow path through the reactor vessel and SRV discharge lines. The SRV control switches are placed in the "OPEN" position. With the SRV control solenoids energized, when pressure at the SRV main discharge reaches approximately 50 psig, the SRV will open allowing coolant to exit the reactor vessel and flow to the torus via the SRV discharge lines. In this situation, the open SRVs prevent reactor pressure from exceeding 250 psig. This mode of shutdown cooling would only be utilized in emergency situations.

The normal testing of LPCI and Core Spray at power and shutdown uses a flow path of suction from the torus with return to the torus. The testing flow path will not enter the reactor vessel because there are two valves in-series that are interlocked to prevent simultaneous opening with reactor pressure above 350 psig. In a shutdown condition with either system under test lineup, an inadvertent injection would be detected by operations and the injection would be terminated before exceeding 250 psig based on observation and alarm of reactor vessel level. A Core Spray pump may be used for reactor vessel and cavity-fill during the outage. The full flow capability of the Core Spray injection check valves are checked during this evolution. Under these conditions the reactor vessel head is removed which will prevent over pressurization.

The shutdown-cooling mode of the RHR system has design prevention from over pressurization since the RHR suction valves from the reactor will close at pressure greater than 150 psig. Closure of the suction valves will trip the running RHR pump preventing vessel pressures from approaching 250 psig.

The reactor head vents are not closed until the coolant temperature is greater than 200°F. This administrative action for head vent closure serves as a mechanism to

reduce the likelihood of pressurization above 250 psig.

5. Control Rod Drive (CRD) and Reactor Water Clean-up (RWCU) Systems

   The CRD and RWCU systems are used to control RPV water level and pressure during cold shutdown conditions using a feed and bleed process. Reactor pressure is controlled by venting the reactor to the Main Condenser via the Main Steam lines, Main Steam line vents, or HPCI or RCIC steam line drains when the reactor coolant temperature is less than 200°F. The low flow rate of these pumps (CRD 55 gpm, RWCU 130 gpm) allows sufficient time for operator action to react to unanticipated level changes and thus pressure changes. Therefore, these systems do not present a significant potential for over pressurization.

6. Class 1 Pressure Test

   The procedure used at Vermont Yankee to perform hydrostatic testing incorporates controls, limitations, and precautions that are reviewed by all personnel involved with maintaining RPV pressure/temperature controls. Rigorous abort criteria provides for immediate actions when limits are approached. Senior Operations Management provides management oversight during the evolution and a Senior License Operator is assigned as a Test Director to assure the procedure is coordinated from start to finish. The procedure is only initiated after a detailed pre-job briefing of all affected people at which time all procedural precautions, limitations, and abort criteria are presented. RPV temperature and pressure are monitored throughout the test to ensure compliance with required pressure-temperature limits. Pressurization rates are controlled throughout the test and direction is provided for control using the RWCU or the CRD pumps. Therefore, this system test does not present a significant potential for over pressurization.

In conclusion, the review of potential high-pressure injection sources confirms that a cold overpressure event at Vermont Yankee is extremely unlikely.

b. Review of Operator Training and Work Control Process

   1. Reactor Operator Training

      Licensed Operator Training provides another method to control reactor water level, temperature, and pressure, in addition to the design and procedural barriers discussed above. Simulator training for start-up and shut down scenarios provides an opportunity to perform the operator actions to maintain reactor pressure-temperature limits.

      Procedural controls for reactor temperature, water level, and pressure are an integral part of Operator training. Specifically, operators are trained in methods of controlling RPV water level within specified limits, as well as responding to abnormal RPV water level conditions outside the established limits. Additionally, control room operators receive training on compliance with the Technical Specification pressure-temperature limits curves and the basis for these limitations and curves. Plant-specific procedures have been developed to provide guidance to the operators regarding compliance with the Technical Specification requirements on pressure-temperature limits.

2. Work Control Process

During plant outages, work control procedures require that the outage schedule and changes to the schedule receive a risk assessment review commensurate with their safety significance. Senior Operations personnel provide input to the outage schedule to avoid conditions that could adversely impact reactor water level, pressure, or temperature. Schedules are issued listing the work activities to be performed.

During refueling outages, work is coordinated through the Outage Control Center. In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor water level or decay heat removal. The Control Room Operator is required to provide positive control of reactor water level and pressure within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken. Cognizant individuals involved in the work activity attend pre-job briefings. Expected plant responses and contingency actions to address unexpected conditions, or responses that may be encountered, are included in the briefing discussion.

Based upon the above, the probability of a low temperature RPV over-pressure event at Vermont Yankee is considered to be less than or equal to that used in the USNRC safety evaluation.

3. Conclusion

Deferral of the RPV circumferential shell weld examinations for Vermont Yankee's Fourth Interval does not impact quality and safety as discussed above. Vermont Yankee believes deferral in performing the augmented inspections of the RPV circumferential shell welds for the Fourth Interval provides an acceptable level of quality and safety.

6. Duration of Proposed Alternative

It is proposed to use the alternative for the duration of the Vermont Yankee Fourth Ten-Year Interval (September 1, 2003 through August 31, 2013).

7. Precedents

1. Entergy (Pilgrim Station) Letter to USNRC, "Pilgrim Relief Request (PRR)-28, Revision 1, Relief form ASME Code, Section XI, Examinations of Reactor Vessel Circumferential Shell Welds, Pursuant to Generic Letter 98-05," dated February 3, 2003.
2. USNRC letter to Entergy (Pilgrim Station), "Pilgrim Nuclear Power Station- Pilgrim Relief Request No. 28, Relief form ASME Code, Section XI, Examinations of Reactor Pressure Vessel Circumferential Shell Welds, (TAC NO. MB6074)," dated March 24, 2000.

8. References

1. BWRVIP-05, dated September 1995, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations."
2. Letter VYNPC to USNRC, BVY 96-105, dated September 10, 1996, "Augmented Examination of the Vermont Yankee Reactor Pressure Vessel Shell Welds."

3. Letter USNRC to VYNPC, NVY 96-159, dated October 11, 1996, "Evaluation of Flaw Indication Found During Reactor Pressure Vessel Inspections at Vermont Yankee Nuclear Power Station."
4. Letter USNRC to BWRVIP, dated August 14, 1997, "Transmittal of NRC Staff's Independent Assessment of the Boiling Water Reactor Vessel and Internals Project BWRVIP-05 Report and Proprietary Request for Additional Information."
5. Letter BWRVIP to USNRC, dated December 18, 1997, "BWRVIP Response to NRC Request for Additional Information on BWRVIP-05."
6. Letter USNRC to BWRVIP, dated July 28, 1998, "Final Safety Evaluation of the BWRVIP-05 Report."
7. USNRC Generic Letter 98-05, dated November 10, 1998, "BWR Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds."
8. Letter VYNPC to USNRC, BVY 99-09, dated January 28, 1999, "Circumferential Weld Examination Frequency."
9. Letter USNRC to VYNPC, NVY 99-16, dated February 18, 1999, "Augmented Examination of the Reactor Pressure Vessel Shell Welds at Vermont Yankee Nuclear Power Station."
10. Letter USNRC to VYNPC, NVY 99-40, dated April 15, 1999, "Reactor Pressure Vessel Circumferential Weld Examination Frequency at Vermont Yankee Nuclear Power Station."
11. Letter VYNPC to USNRC, BVY 03-29, dated March 26, 2003, "Technical Specifications Proposed Change No. 258, RPV Fracture Toughness and Material Surveillance Requirements."