Docket No. 50-271

APR 17 1990

Mr. L. A. Tremblay
Licensing Engineer
Vermont Yankee Nuclear Power Corporation
580 Main Street
Bolton, Massachusetts 01740-1398

Dear Mr. Tremblay:

SUBJECT: ISSUANCE OF AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. DPR-28 - VERMONT YANKEE NUCLEAR POWER STATION (TAC NO.75499)

The Commission has issued the enclosed Amendment No. 120 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment is in response to your application dated November 10, 1989.

This amendment revises the Pressure-Temperature limit curves in Technical Specification (TS) Figure 3.6.1.

The enclosed Safety Evaluation also closes out our review (TAC NO. 71563) of your letter of November 10, 1988 which responded to our Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials" dated July 12, 1988. The Generic Letter included NRC guidance for the calculation of the nil-ductility reference temperature of reactor vessel beltline materials which related to this pressure/temperature TS change.

Notice of Issuance of this amendment will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

/S/

Morton B. Fairtile, Project Manager
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 120 to License No. DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

c/P-1 enl

OFC	:PDI-3/LA	:PDI-3/PM	:OGC	:PDI-3/D	:	:	:
NAME	:MRushbrook	:MBFairtile	: E Holler	:RHWessman	:	:	:
DATE	:03/ /90	:03/6 /90	:03/11 /90	:03/17 /90	:	:	:

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DF01 1/1

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

Richard H. Wessman, Director
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: APR 17 1990

OFC	:PDI-3/PM	:PDI-3/LA	:OGC	:PDI-3/D	:	:	:
NAME	:MBFairtile	:MRushbrook	: E HOLLER	:RHWessman	:	:	:
DATE	: / /90	: 4/ /90	: 4/11/90	:4/17/90	:	:	:

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AMENDMENT NO. 120 TO DPR-28 VERMONT YANKEE NUCLEAR POWER STATION DATED APR 17 1990

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NUCLEAR REGULATORY COMMISSION
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APR 17 1990

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Notice of Issuance of this amendment will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

Morton B. Fairtile

Morton B. Fairtile, Project Manager
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 120 to License No. DPR-28
2. Safety Evaluation

cc w/enclosures:
See next page

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Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. L. A. Tremblay

APR 1 7 1990

cc:

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Adjudicatory File (2)
Atomic Safety and Licensing Board
Panel Docket
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. DPR-28

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

   A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated November 10, 1989 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 set forth in 10 CFR Chapter I;

   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:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 120 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

RH Wessman

Richard H. Wessman, Director
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: APR 17 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

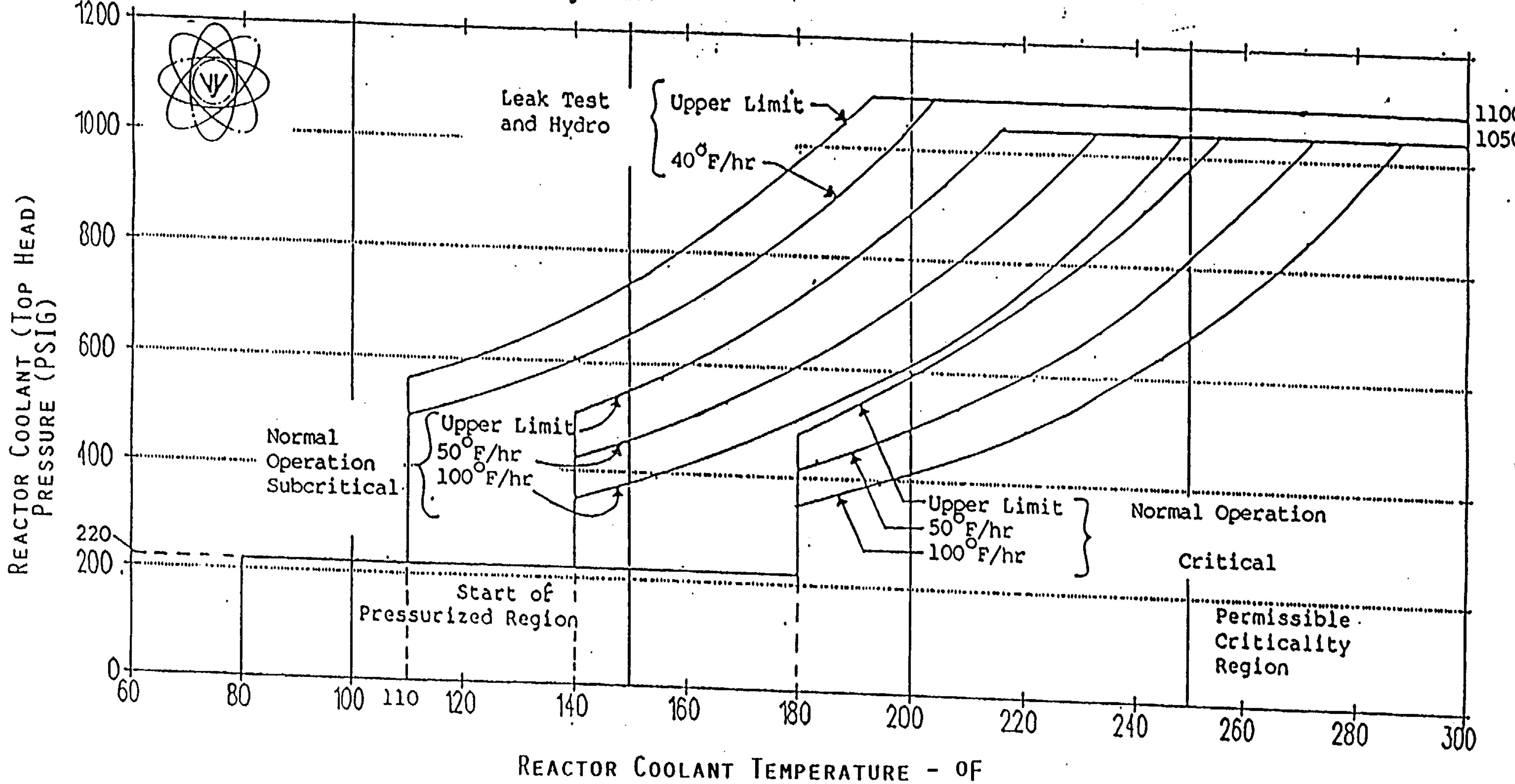
Remove	Insert
111	111
117	117

Figure 3.6.1
Reactor Vessel
Pressure-Temperature Limitations

Valid Thru 4.46E8 MWH(t) [32 EFPY]

Adjusted Reference Temperature at 1/4T = 89°F
Adjusted Reference Temperature at 3/4T = 73°F
Adjusted Reference Temperature at Flange = 20°F

Curves are in accordance with 10CFR50 App G and RG 1.99 Rev 2



Amendments No. ~~62~~, ~~61~~, ~~63~~, 120

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.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by their internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing locations.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures should be within 50°F of each other prior to startup of an idle loop.

The reactor vessel materials have been tested to determine their initial reference temperature nil-ductility transition temperature (RT_{NDT}) of 40°F maximum. Reactor operation and resultant fast neutron (E >1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature can be predicted using current industry practices and Vermont Yankee Surveillance Program data. (Regulatory Guide 1.99, Revision 2, and Battelle Columbus Laboratory Report BCL 585-84-3, dated May 15, 1984. The pressure/temperature limit curve, Figure 3.6.1, includes predicted adjustments for this shift in RT_{NDT} for operation through 4.46×10^8 MWH(t), as well as adjustments for possible errors in the pressure and temperature sensing instruments.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

INTRODUCTION

By letter dated November 10, 1989, the Vermont Yankee Nuclear Power Corporation (the licensee) requested an amendment to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The proposed amendment would revise the Presure-Temperature limit curves in Technical Specification (TS) Figure 3.6.1.

This proposed TS change reflects the shift in transition temperature for the reactor vessel materials for operation through a cumulative energy output of 4.46×10^8 MWh(t). The change is necessary because the existing curves are limited to a cumulative energy output of 1.79×10^8 MWh(t) which is expected to be reached by May 1990.

Previous to their request of November 10, 1989 the licensee responded to our Generic Letter 88-11 "NRC Position on Radiation Embrittlement of Reactor Vessel Materials" by a letter dated November 10, 1988. This Generic Letter provided guidance for the calculation of the nil-ductility reference temperature of reactor vessel beltline materials which relates to this Pressure-Temperature limit curve change request as discussed below.

DISCUSSION

The licensee has requested permission to revise the pressure/temperature (P/T) limits in the Vermont Yankee Nuclear Power Station Technical Specifications, Section 3.6. The request was documented in a letter from the licensee dated November 10, 1989. This revision also changes the effectiveness of the P/T limits to 32 effective full power years (EFPY). The proposed P/T limits were developed based on Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program and to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the rector beltline.

EVALUATION

The staff evaluated the effect of reactor neutron irradiation embrittlement on each beltline material in the Vermont Yankee reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 32 EFPY was plate I-14 with 0.11% copper (Cu), 0.63% nickel (Ni), and an initial RTNDT of 40°F.

The licensee has removed one surveillance capsule from the Vermont Yankee reactor vessel. The results from this capsule were published in the Battelle-Columbus Laboratories report BCL-585-84-3. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal and HAZ.

For the limiting beltline material, plate I-14, the staff calculated the ART to be 63.1° F at 1/4T (T = reactor vessel beltline thickness) and 55.5° F for 3/4T at 32 EFPY. The staff used a neutron fluence of 1.6E17 n/cm^2 at 1/4T and 9E16 n/cm^2 at 3/4T. The ART was determined using Section 1 of RG 1.99, Rev. 2, as only one surveillance capsule has been removed from Vermont Yankee.

The licensee used a more conservative safety factor then the one in RG 1.99, Rev. 2, to calculate an ART of 89° F EFPY at 1/4T and 73° F at 3/4T at 32 EFPY for a plate material in the beltline. The staff believes that an ART of 63.1° F is sufficient to protect the reactor vessel from embrittlement. Substituting the ART of 63.1° F into the equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the belt material requirement in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Paragraph IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120° F for normal operation and by 90° F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the pre-service system hydrostatic test pressure. In this case the minimum permissible temperature is 60° F (33° C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 20° F, the staff has determined that the proposed P/T limits satisfy paragraph IV.A.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. No USE data were available for plates I-15, I-16, and I-17. However, Charpy impact data at 40° F were available. The lowest individual reading was for plate I-17--ie., 65 ft-lb. Using this for a USE value and Figure 2 of RG 1.99, Rev. 2, it was predicted that the EOL USE would be 59.5 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

SUMMARY

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 32 EFPY as the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 guidance as the licensee used the methods of RG 1.99, Rev. 2 to calculate the ART. Therefore, the proposed P/T limits are acceptable for incorporation into the Vermont Yankee Technical Specification.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the 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 published a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this 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 this amendment.

CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 2449) on January 24, 1990 and consulted with the State of Vermont. No public comments were received and the State of Vermont did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2, Pressure-Temperature Limits

3. L. M. Lowery et al., "Final Report on Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the Vermont Yankee Nuclear Power Station to Yankee Atomic Electric Company," BCL-585-84-3, Battelle-Columbus Laboratories, May 15, 1984

4. R. W. Capstick (VYNPC) to USNRC Document Control Desk, Subject: Vermont Yankee Response to NRC Generic Letter 88-11, November 10, 1988.

5. W. P. Murphy (VYNPC) to USNRC Document Control Desk, Subject: Proposed Change to Revise the Reactor Vessel Pressure-Temperature Curves in the Vermont Yankee Technical Specification (Generic Letter 88-11), November 11, 1988.

Principal Contributor: John Tsao

Dated: APR 1 7 1990