DCQ 016

June 24, 1986

Docket No.: 50-271

Mr. R. W. Capstick
Licensing Engineer
Vermont Yankee Nuclear Power
Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Capstick:

The Commission has issued the enclosed Amendment No. 93 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment consists of changes to the Technical Specifications in response to your application dated May 10, 1985, with supplemental information provided by letter dated November 2, 1985.

The amendment revises the Technical Specifications to accommodate shifts in transition temperature for the reactor pressure vessel materials that were induced by radiation damage. These shifts are accounted for by revision of the plant pressure-temperature limits for heating up and cooling down the reactor vessel. Periodic review and adjustment, if necessary, of the curves to account for the effects of irradiation are required by 10 CFR 50, Appendices G and H.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Vernon L. Rooney, Project Manager
BWR Project Directorate #2
Division of BWR Licensing

Enclosures:
1. Amendment No. 93 to License No. DPR-28
2. Safety Evaluation

cc w/enclosure:
See next page

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Mr. R. W. Capstick
Vermont Yankee Nuclear Power Corporation
Vermont Yankee Nuclear Power Station

cc:

Mr. J. G. Weigand
President & Chief Executive Officer
Vermont Yankee Nuclear Power Corp.
R. D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Mr. Donald Hunter, Vice President
Vermont Yankee Nuclear Power Corp.
1671 Worcester Road
Framingham, Massachusetts 01701

New England Coalition on
Nuclear Pollution
Hill and Dale Farm
R. D. 2, Box 223
Putney, Vermont 05346

Mr. Walter Zaluzny
Chairman, Board of Selectman
Post Office Box 116
Vernon, Vermont 05345

J. P. Pelletier, Plant Manager
Vermont Yankee Nuclear Power Corp.
Post Office Box 157
Vernon, Vermont 05354

Raymond N. McCandless
Vermont Division of Occupational
& Radiological Health
Administration Building
10 Baldwin Street
Montpelier, Vermont 05602

Honorable John J. Easton
Attorney General
State of Vermont
109 State Street
Montpelier, Vermont 05602

John A. Ritscher, Esquire
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

W. P. Murphy, Vice President &
Manager of Operations
Vermont Yankee Nuclear Power Corp.
R. D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Mr. Gerald Tarrant, Commissioner
Vermont Department of Public Service
120 State Street
Montpelier, Vermont 05602

Public Service Board
State of Vermont
120 State Street
Montpelier, Vermont 05602

Vermont Yankee Decommissioning
Alliance
Box 53
Montpelier, Vermont 05602-0053

Resident Inspector
U. S. Nuclear Regulatory Commission
Post Office Box 176
Vernon, Vermont 05354

Vermont Public Interest
Research Group, Inc.
43 State Street
Montpelier, Vermont 05602

Thomas A. Murley
Regional Administrator
Region I Office
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. DPR-28

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

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated May 10, 1985, as supplemented November 21, 1985, 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 2.C(2) of Facility Operating License No. DPR-28 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications, contained in Appendix A, as revised through Amendment No. 93, 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 the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

for

Daniel R. Muller, Project Director
BWR Project Directorate #2
Division of BWR Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 24, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

111
111a
111b
116
117
118

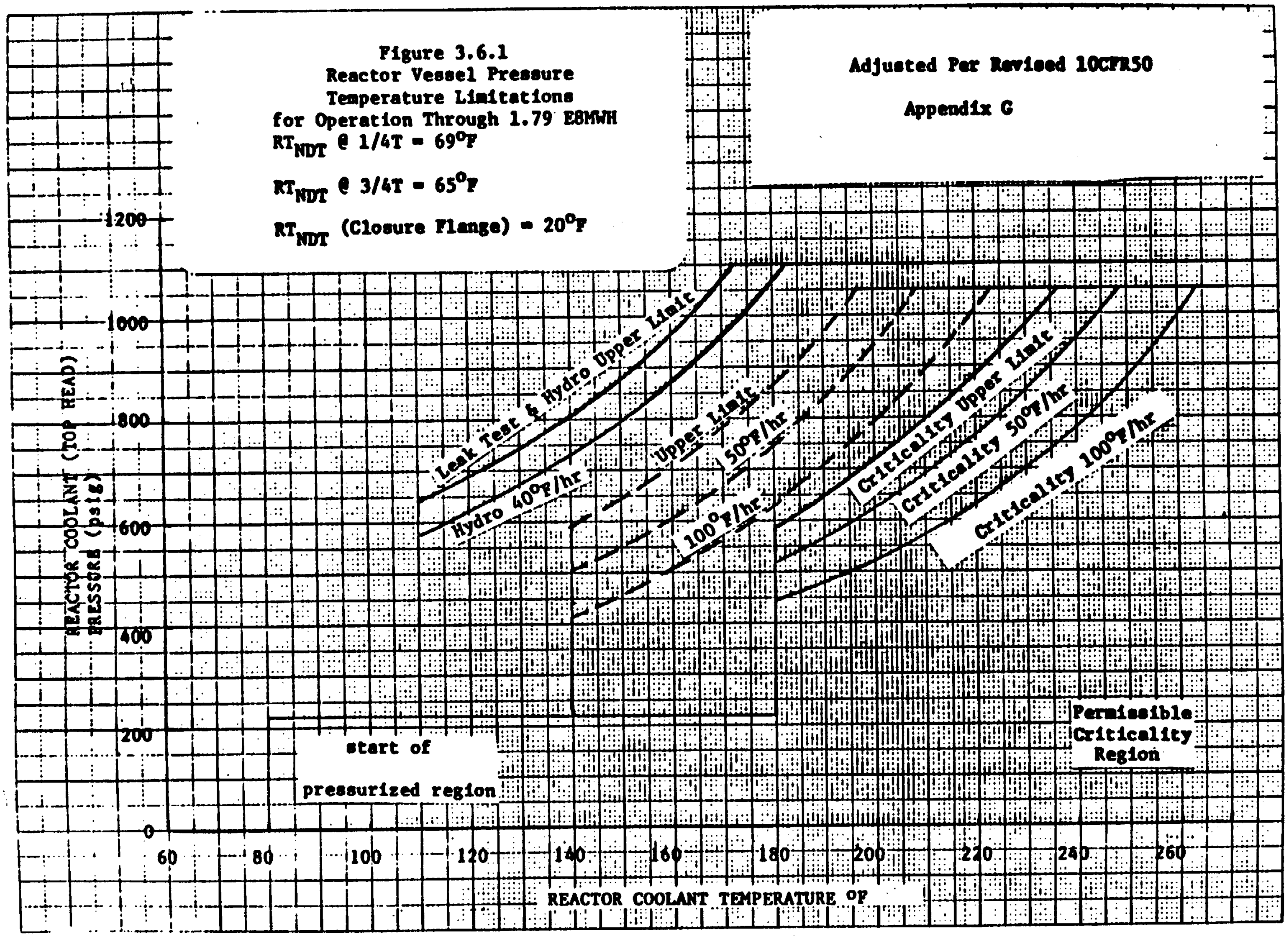
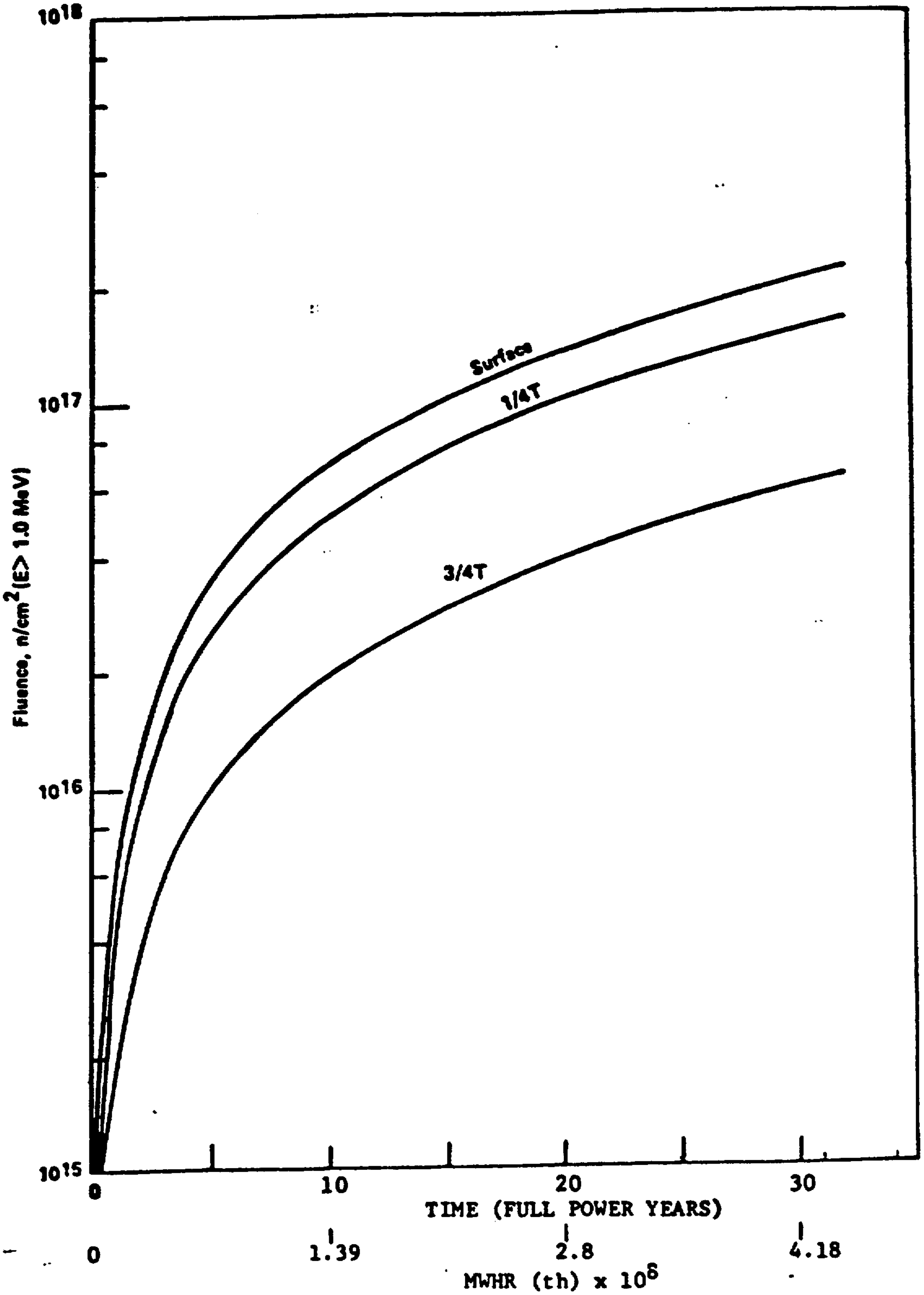


Figure 3.6.1
Reactor Vessel Pressure
Temperature Limitations
for Operation Through 1.79 E8MWH
RT_{NDT} @ 1/4T = 69°F

RT_{NDT} @ 3/4T = 65°F

RT_{NDT} (Closure Flange) = 20°F

FIGURE 3.6.2
FAST NEUTRON FLUENCE (E > 1 MEV) AS A FUNCTION OF THERMAL ENERGY
AND FULL POWER YEARS



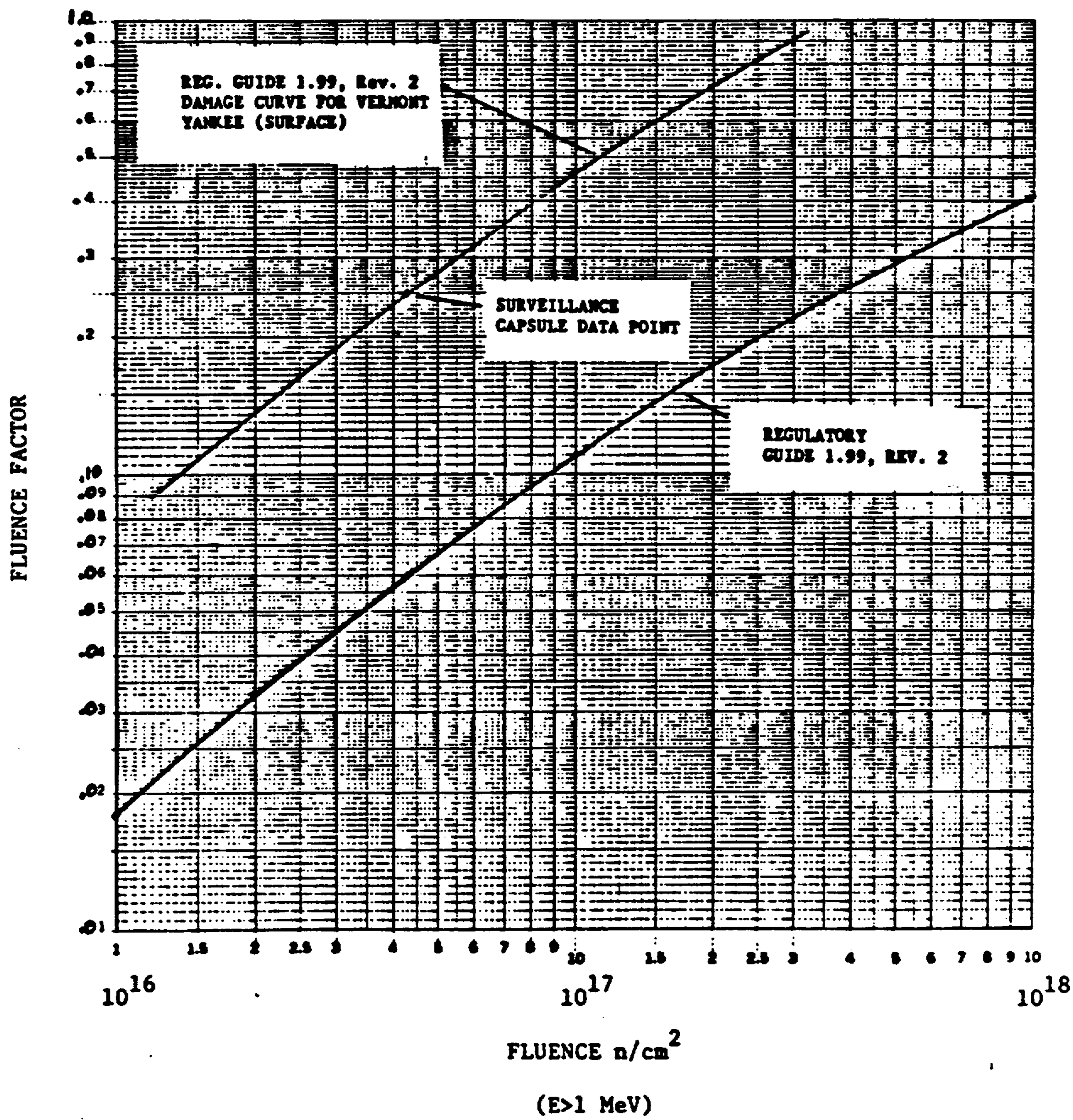
REFERENCE: L. M. Lowry et al. "Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from Vermont Yankee Nuclear Power Station.

Batelle Columbus Laboratories Report #BCL-585-84-3, May 15, 1984

FIGURE 3.6.3

FLUENCE FACTOR FOR USE IN REGULATORY GUIDE 1.99

Rev. 2



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 1.79×10^8 MWH(t), as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The reference temperature of the closure flange material was determined by material testing and Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements for Older Plants". The closure flange is located in a low neutron fluence area and therefore no measurable RT_{NDT} shift is expected over the life of the plant.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185 reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. Battelle Columbus Laboratory Report BCL-585-84-3, dated May 15, 1984, provides this information for the ten-year surveillance capsule. In order to estimate the material properties at the 1/4 and 3/4 T positions in the vessel plate, the shift in RT_{NDT} is determined in accordance with Regulatory Guide 1.99, Revision 2. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines, shown on Figure 3.6.1, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice leak and hydrostatic testing.

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

<u>Coolant Chemistry</u>

A steady-state radioiodine concentration limit of 1.1 μCi of I-131 dose equivalent per gram of water in the Reactor Coolant System can be reached if the gross radioactivity in the gaseous effluents is near the limit, as set forth in Specification 3.8.C.1a, or there is a failure or prolonged shutdown of the cleanup demineralizer. In the event of a steam line rupture outside the drywell, the NRC staff calculations show the resultant radiological dose at the site boundary to be less than 30 Rem to the thyroid. This dose was

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION AND BACKGROUND

By letter dated May 10, 1985, Vermont Yankee Nuclear Power Corporation proposed that the Vermont Yankee Nuclear Power Station Technical Specification pressure and temperature limit curves be changed. The proposed change would revise the Technical Specifications to accommodate the change in toughness properties for the reactor vessel materials that were induced by radiation effects. Periodic review and, if necessary, adjustment of the pressure and temperature limit curves to account for the effects of increased neutron exposure are required by 10 CFR Part 50, Appendices G and H.

This change adjusts the reactor vessel pressure and temperature limitations to compensate for the effects of increased neutron exposure to permit operation to a cumulative energy output of 1.790×10^8 MWh(t). This adjustment is necessary because the existing curves are limited to an energy output of 1.33×10^8 MWh(t), a value which is expected to be reached during 1986. This change also adjusts the fluence factor and fluence vs. thermal energy curves to incorporate revised fast neutron fluence calculations. The licensee submitted clarifying information by letter dated November 21, 1985.

The new curves incorporate results from the surveillance capsule removed in March 1983 and new tests performed on unirradiated specimens for archival base, weld, and heat affected zone materials.

2.0 EVALUATION

The purpose of the reactor vessel surveillance program is to monitor the effect that neutron irradiation and the thermal environment will have on the beltline materials' reference nil-ductility temperature (RT_{NDT}). The method recommended by the staff for predicting the effect of neutron irradiation and the thermal environment on beltline materials' RT_{NDT} is documented in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." In revision 1 to this regulatory guide, the estimate of the increase in RT_{NDT} is based

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upon the amount of copper, phosphorus and the neutron fluence. In proposed revision 2, dated February 1986, the increase in RT_{NDT} is based upon the amount of copper, nickel and the neutron fluence. Proposed revision 2 was prepared from the analysis of commercial reactor vessel material surveillance data generated during the staff's review of the issue of "Pressurized Thermal Shock" and has been issued for public comment.

In order to develop revised curves, several material parameters needed to be re-established or revised for the Vermont Yankee reactor vessel limiting material. Changes were needed to reflect the results of impact tests performed on surveillance capsule material which was removed from the reactor vessel in March 1983. In addition, new tests were performed on unirradiated archival base, weld, and heat affected zone specimens to more clearly establish initial nil-ductility transition temperatures.

The base metal for the Vermont Yankee reactor pressure vessel is A533 Grade B, Class 1 steel. Charpy V-notch and tensile specimens were prepared from an actual beltline plate (No. 2 shell and piece marked 1-14). The specimens were prepared from A533 steel plate (Heat No. C3017-2) provided by Lukens Steel Corporation in 1969.

Only two plates lie in the vessel belt line, pieces 1-14 and 1-15. The limiting plate has been established as piece 1-14 which is the surveillance plate. An initial $RT_{NDT} = 40°F$ was established for this plate. From the Battelle tests, the shift in RT_{NDT} was 19°F at a fluence of 4.3×10^{16} n/cm^2. Utilizing the calculational procedure of the proposed Regulatory Guide 1.99, Revision 2, a shift of only 4.7°F results at this fluence. The Chemistry Factor (CF) for piece 1-14 is 76, representing a copper content of 0.11 weight percent and a nickel content of 0.68 weight percent. The measured shift is within one standard deviation of that calculated (Regulatory Guide 1.99, Revision 2 assumes $1\sigma = 17°$ for base metal). However, because the calculational procedures of the Regulatory Guide results in a less conservative prediction of shift, a modified Regulatory Guide fluence factor curve was developed. The modified curve utilizes the same curve shape and damage prediction as Regulatory Guide 1.99, Revision 2, but passes through the Vermont Yankee surveillance capsule data point. In effect, the fluence factor parameter in the Regulatory Guide 1.99 reference temperature shift equation is multiplied by a factor of 4.17 to duplicate the measured RT_{NDT} shift at Vermont Yankee. Future shift values can then be determined from this curve until the next surveillance specimen is removed.

Regulatory Guide 1.99, Revision 2 proposes that surveillance test results can be used after two capsules have been tested with reliable results. However, we consider the described method of using one data point from one capsule to be very conservative in this case and therefore acceptable. The results of this procedure are also conservative with respect to Revision 1 of the Guide.

In the future, a revised shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Because the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel.

We have reviewed the calculations which form the basis for the proposed change and find them acceptable. The proposed changes to the Technical Specifications relating to the pressure and temperature limits for hydrostatic and leak tests, subcritical/critical heat up and cool down, and operation meet the requirements of 10 CFR 50, Appendix G, Regulatory Guide 1.99, Revisions 1 and 2 and Appendix G, Section III of the ASME Code. The proposed changes are acceptable for incorporation into the Technical Specifications.

3.0 ENVIRONMENTAL CONSIDERATIONS

This 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. 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 the cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this 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.

4.0 CONCLUSION

We have 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 the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor:
H. Conrad

Dated: June 24, 1986