September 30, 1996

Mr. Donald A. Reid
Vice President, Operations
Vermont Yankee Nuclear Power Corporation
Ferry Road
Brattleboro, VT 05301

SUBJECT: ISSUANCE OF AMENDMENT FOR VERMONT YANKEE NUCLEAR POWER STATION (TAC NO. M95150)

Dear Mr. Reid:

The Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station in response to your application dated April 4, 1996.

The amendment revises the Technical Specifications regarding the surveillance requirement for control rod over-travel by modifying surveillance requirements following rod de-coupling and moving the current surveillance methodology to licensee administratively controlled documents. Specifically, the amendment removes the requirement in Specification 4.3.B.1(b) to verify prior to coupling that the over-travel indicating light is working properly by withdrawing an uncoupled control rod drive to the over-travel position.

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,

/S/

C. Craig Harbuck, Acting Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-271

Enclosures: 1. Amendment No. 149 to DPR-28
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\VERMONT\VY95150.AMD *See Previous Concurrence

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DATE: September 30, 1996

ISSUANCE OF AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-28

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
License No. DPR-28

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

   A. The application for amendment filed by the Vermont Yankee Nuclear Power Corporation (the licensee) dated April 4, 1996, 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:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.149 , 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 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

S. Singh Bajwa, Acting Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 30, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 149

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
83	83
89	89
90	90

3.3 LIMITING CONDITIONS FOR OPERATION

2. The Control Rod Drive Housing Support System shall be in place when the Reactor Coolant System is pressurized above atmospheric pressure with fuel in the reactor vessel unless all operable control rods are fully inserted.

3. While the reactor is below 20% power, the Rod Worth Minimizer (RWM) shall be operating while moving control rods except that:

   (a) If after withdrawal of at least 12 control rods during a startup, the RWM fails, the startup may continue provided a second licensed operator verifies that the operator at the reactor console is following the control rod program; or

   (b) If all rods, except those that cannot be moved with control rod drive

4.3 SURVEILLANCE REQUIREMENTS

positive coupling and the results of each test shall be recorded. The drive and blade shall be coupled and fully withdrawn. The position and over-travel lights shall be observed.

2. The Control Rod Drive Housing Support System shall be inspected after reassembly and the results of the inspection recorded.

3. Prior to control rod withdrawal for startup the Rod Worth Minimizer (RWM) shall be verified as operable by performing the following:

   (a) The Reactor Engineer shall verify that the control rod withdrawal sequence for the Rod Worth Minimizer computer is correct.

   (b) The Rod Worth Minimizer diagnostic test shall be performed.

BASES:

3.3 & 4.3 CONTROL ROD SYSTEM

A. Reactivity Limitations

1. Reactivity Margin - Core Loading

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. At each refueling the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk with the highest worth control rod fully withdrawn and all others inserted. The value of R in % Δk is the amount by which the calculated core reactivity, at any time in the operating cycle, exceeds the reactivity at the time of the demonstration. R must be a positive quantity or zero. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all inverted poison tubes present in the core. The 0.25% Δk is provided as a finite, demonstrable, sub-criticality margin.

2. Reactivity Margin - Inoperable Control Rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved with drive pressure. If a rod is disarmed electrically, its position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shutdown at all times with the remaining control rods, assuming the highest worth, operable control rod does rod insert. An allowable pattern for control rods valved out of service will be available to the reactor operator. The number of rods permitted to be inoperable could be many more than the six allowed by the Specification, particularly late in the operation cycle; however, the occurrence of more than six could be indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housing, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The surveillance requires verifying a control rod does not go to the withdrawn over-travel position. The over-travel position feature provides a positive check on the

BASES: 3.3 & 4.3 (Cont'd)

coupling integrity since only an uncoupled CRD can reach the over-travel position. The verification is required to be performed when a control rod is fully withdrawn after each refueling outage (since work on the control rod or CRD System may have affected coupling), and after each uncoupling.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage of the primary coolant system. The design basis is given in Subsection 3.5.2 of the FSAR, and the design evaluation is given in Subsection 3.5.4. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing.

3. In the course of performing normal startup and shutdown procedures, a pre-specified sequence for the withdrawal or insertion of control rods is followed. Control rod dropout accidents which might lead to significant core damage, cannot occur if this sequence of rod withdrawals or insertions is followed. The Rod Worth Minimizer restricts withdrawals and insertions to those listed in the pre-specified sequence and provides an additional check that the reactor operator is following prescribed sequence. Although beginning a reactor startup without having the RWM operable would entail unnecessary risk, continuing to withdraw rods if the RWM fails subsequently is acceptable if a second licensed operator verifies the withdrawal sequence. Continuing the startup increases core power, reduces the rod worth and reduces the consequences of dropping any rod. Withdrawal of rods for testing is permitted with the RWM inoperable, if the reactor is subcritical and all other rods are fully inserted. Above 20% power, the RWM is not needed since even with a single error an operator cannot withdraw a rod with sufficient worth, which if dropped, would result in anything but minor consequences.

4. Refer to the Vermont Yankee Core Performance Analysis report.

5. The Source Range Monitor (SRM) system has no scram functions. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are a function of the initial neutron flux. The requirement of at least three counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel is adequate to monitor the approach to criticality, therefore, two operable SRM's are specified for added conservatism.

6. The Rod Block Monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power level operation. During reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR less than the fuel cladding integrity safety limit. During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods will provide added assurance that improper withdrawal does not occur. It is the responsibility of the Nuclear Engineer to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable control rods.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated April 4, 1996, the Vermont Yankee Nuclear Power Corporation (the licensee) submitted a request for changes to the Vermont Yankee Nuclear Power Station (VYNPS) Technical Specifications (TSs). The proposed changes would remove the requirement in Specification 4.3.B.1(b) to check the operability of the control rod over-travel indicator circuit prior to coupling each control rod. The licensee will continue to monitor the operability of the overtravel circuit by observing the position and overtravel lights during a coupling check following de-coupling. The licensee proposed to maintain testing of the over-travel indication function using the current methodology in administratively controlled documents. The actual requirement in Specification 4.3.B.1(b) to verify control rod drive coupling would not be changed.

The safety objective of control rod drive coupling verification is to ensure that the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. As discussed in the TS Bases for the control rods, control rod dropout accidents can lead to significant core damage. Consequently, in order to eliminate the possibility of a rod dropout accident, the TS require maintaining control rod drive coupling integrity.

The purpose of the control rod drive over-travel indication instrumentation is to provide a positive check on the integrity of the control rod drive coupling. This check is based on the design of the control rod drive mechanism and control rod which prevents a coupled drive from reaching the over-travel position. That is, an uncoupled drive may be withdrawn until the over-travel annunciator indicates an over-travel position has been reached; but a coupled drive cannot actuate the over-travel annunciator when fully withdrawn.

2.0 EVALUATION

TS 4.3.B.1(b) requires verifying control rod drive coupling integrity (1) when a rod is fully withdrawn by observing that the rod does not go to the over-travel position, and (2) prior to startup following a refueling outage by continuously withdrawing each rod to observe that the rate of withdrawal is proper and that the rod does not go to the over-travel position. These provisions are not proposed to be changed. However, this TS also states, "Following uncoupling, each control rod drive and blade shall be tested to verify positive coupling and the results of each test shall be recorded." It then prescribes how to perform this test:

> "*This test shall consist of checking the operability of the over-travel circuit prior to coupling by withdrawing the drive and observing the over-travel light.* The drive and blade shall *then* be *immediately* coupled and fully withdrawn. The position and over-travel lights shall be observed."

The licensee proposed to remove the operability verification of the over-travel circuit from this specification by deleting the portions of the preceding text indicated by italics. The specification would then require the licensee, following uncoupling and subsequent re-coupling, to check the operability of the overtravel circuit by observing both the position and overtravel lights.

An attempt to withdraw a fully-withdrawn control rod past the backseat position will result in one of the following displays:

- For a coupled drive, the control rod will not withdraw past the backseated position as indicated by the control rod position indicator probe (PIP) displays of "48" and "full-out" and the over-travel indicator not displayed.

- For an uncoupled drive, the control rod will withdraw past the backseated position as indicated by display of the over-travel indicator, but no PIP displays.

- If neither the PIP displays nor the over-travel display appear, then the drive may be either (a) coupled with both PIP displays inoperable, or (b) uncoupled with the over-travel display inoperable.

  If the "48" and "full-out" indicators are no longer displayed, but the over-travel indicator is also not displayed, this is indicative of a probable over-travel indication problem and should be investigated prior to continuing or declaring the control rod and drive mechanism "coupled." This is why the current Specification requires observing both "the position and over-travel lights." This type of redundant indication makes the required additional specific surveillance of the over-travel indication unnecessary and redundant. On this basis, the changes to surveillance testing of the overtravel circuit and corresponding TS and TS bases are acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC 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 issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 20860). 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.

5.0 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.

Principal Contributor: C. Harbuck

Date: September 30, 1996