March 28, 1983

Docket No. 50-271

Mr. J. B. Sinclair
Licensing Engineer
Vermont Yankee Nuclear Power
Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Sinclair:

The Commission has issued the enclosed Amendment No. 76 to Facility Operating License No. DPR-28 for Vermont Yankee Nuclear Power Station. This amendment consists of changes to the Technical Specifications in response to your application dated January 10, 1983, as clarified through subsequent discussions between the NRC staff and members of your staff.

These changes to the Technical Specifications pertain to limiting conditions of operation and surveillance requirements related to the scram discharge system modifications and the analog trip system.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

Vernon L. Rooney, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 76 to DPR-28
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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Mr. J. B. Sinclair

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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. 76
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 January 10, 1983, 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 the 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:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 76 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 28, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise the Technical Specifications as follows:

REMOVE	INSERT
19	19
22	22
25	25
28	28
47	47
48	48
50	50
51	51
52	52
53	53
54	54
57	57
80	80

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Trip Settings	Modes in Which Functions Must Be Operating: Refuel(1)	Startup	Run	Minimum Number Operating Instrument Channels Per Trip System(2)	Required Conditions When Minimum Conditions For Operation Are Not Satisifed(3)
1. Mode Switch In Shutdown		X	X	X	1	A
2. Manual Scram		X	X	X	1	A
3. IRM						
High Flux	≤120/125	X	X	X(11)	2	A
INOP		X	X	X(11)	2	A
4. APRM						
High Flux (Flow Bias)	≤0.66W+54%(4)			X	2	A or B
High Flux (Reduced)	≤15%	X	X		2	A
INOP				X	2(5)	A or B
Downscale	≥2/125			X	2	A or B
5. High Reactor Pressure	≤1055 psig	X	X	X	2	A
6. High Drywell Pressure	≤2.5 psig	X	X	X	2	A
7. Reactor Low Water Level	≥127.0 inches(6)	X	X	X	2	A
8. Scram Discharge Volume High Level	≤ 21 gallons	X	X	X	2 (per volume)	A

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VYNPS

TABLE 4.1.1

SCRAM INSTRUMENTATION AND LOGIC SYSTEMS FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENTATION, LOGIC SYSTEMS AND CONTROL CIRCUITS

Instrument Channel	Group(3)	Functional Test(7)	Minimum Frequency(4)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	Each Refueling Outage
Manual Scram	A	Trip Channel and Alarm	Every 3 Months
IRM			
High Flux	C	Trip Channel and Alarm(5)	Before Each Startup & Weekly During Refueling(6)
Inoperative	C	Trip Channel and Alarm	Before Each Startup & Weekly During Refueling(6)
APRM			
High Flux	B	Trip Output Relays(5)	Once Each Week
High Flux (Reduced)	B	Trip Output Relays(5)	Before Each Startup & Weekly During Refueling(6)
Inoperative	B	Trip Output Relays	Once Each Week
Downscale	B	Trip Output Relays(5)	Once Each Week
Flow Bias	B	Trip Output Relays(5)	(1)
High Reactor Pressure	B	Trip Channel and Alarm(5)	(1)
High Drywell Pressure	B	Trip Channel and Alarm(5)	(1)
Low Reactor Water Level(2)(8)	B	Trip Channel and Alarm(5)	(1)
High Water Level in Scram Discharge Volume	B	Trip Channel and Alarm(5)	(1)
High Main Steam Line Radiation(2)	B	Trip Channel and Alarm(5)	Once Each Week
Main Steam Line Iso. Valve Closure	A	Trip Channel and Alarm	(1)
Turbine Con. Valve Fast Closure	A	Trip Channel and Alarm	(1)
Turbine Stop Valve Closure	A	Trip Channel and Alarm	(1)

TABLE 4.1.2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group(1)	Calibration Standard(4)	Minimum Frequency(2)
High Flux APRM			
Output Signal	B	Heat Balance	Once Every 7 Days
Output Signal (Reduced)	B	Heat Balance	Once Every 7 Days
Flow Bias	B	Standard Pressure and Voltage Source	Refueling Outage
LPRM	B(5)	Using TIP System	Every 1000 Equivalent Full Power Hours
High Reactor Pressure	B	Standard Pressure Source	Once/Operating Cycle
Turbine Control Valve Fast Closure	A	Standard Pressure Source	Every 3 Months
High Drywell Pressure	B	Standard Pressure Source	Once/Operating Cycle
High Water Level in Scram Discharge Volume	B	Water Level	Once/Operating Cycle
Low Reactor Water Level	B	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	(6)	Refueling Outage
High Main Steam Line Radiation	B	Appropriate Radiation Source(3)	Refueling Outage
First Stage Turbine Pressure Permissive	A	Pressure Source	Every 6 Months and After Refueling
Main Steam Line Isolation Valve Closure	A	(6)	Refueling Outage

3.1 (continued)

The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specification 2.1.

Instrumentation (pressure switches) is provided to detect a loss-of-coolant accident and initiate the core standby cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

The Control Rod Drive Scram System is designed so that all of the water that is discharged from the reactor by the scram can be accommodated in the discharge piping. This discharge piping is divided into two sections. One section services the control rod drives on the north side of the reactor, the other serves the control rod drives of the south side. A part of the piping in each section is an instrument volume which accommodates in excess of 21 gallons of water and is at the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram. During normal operation, the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level instrumentation has been provided for the instrument volume which scram the reactor when the volume of water reaches 21 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water, and precludes the situation in which a scram would be required but not be able to perform its function adequately. The present design of the Scram Discharge System is in concert with the BWR Owner's Group criteria, which have previously been endorsed by the NRC in their generic "Safety Evaluation Report (SER) for Scram Discharge Systems", dated December 1, 1980.

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the clad safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the clad safety limit from being exceeded in the event of a turbine trip transient without bypass.

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

	Minimum Number of Operable Instrument Channels Per Trip System	Trip Function	Modes in Which Function Must be Operable			Trip Setting
			Refuel	Startup	Run	
		Startup Range Monitor				
(Note 1)	2	a. Upscale (Note 2)	X	X		$\leq 5 \times 10^5$ cps (Note 3)
(Note 1)	2	b. Detector Not Fully Inserted	X	X		
		Intermediate Range Monitor				
(Note 1)	2	a. Upscale	X	X		<108/125 Full Scale
(Note 1)	2	b. Downscale (Note 4)	X	X		≥5/125 Full Scale
(Note 1)	2	c. Detector Not Fully Inserted	X	X		
		Average Power Range Monitor				
(Note 1)	2	a. Upscale (Flow Bias)			X	<0.66W + 42% (Note 5)(Note 8)
(Note 1)	2	b. Downscale			X	≥2/125 Full Scale
		Rod Block Monitor (Note 6)				
(Note 10)	1	a. Upscale (Flow Bias) (Note 7)			X	≤0.66W + N (Note 5)
(Note 10)	1	b. Downscale (Note 7)			X	≥2/125 Full Scale
(Note 9)	1 (per volume)	Scram Discharge Volume	X	X	X	≤12 Gallons
	1	Trip System Logic	X	X	X	

TABLE 3.2.5 NOTES

1. There shall be two operable or tripped trip systems for each function in the required operating mode. If the minimum number of operable instruments are not available for one of the two trip systems, this condition may exist for up to seven days provided that during the time the operable system is functionally tested immediately and daily thereafter; if the condition lasts longer than seven days, the system shall be tripped. If the minimum number of instrument channels are not available for both trip systems, the systems shall be tripped.

2. One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

3. This function may be bypassed when count rate is $\geq$100 cps or when all IRM range switches are above Position 2.

4. IRM downscale may be bypassed when it is on its lowest scale.

5. "W" is percent rated drive flow where 100% rated drive flow is that flow equivalent to 48 x 10^6 lbs/hr core flow. Refer to LCO 3.11.C for acceptable values for N.

6. The minimum number of operable instrument channels may be reduced by one for maintenance and/or testing for periods not in excess of 24 hours in any 30-day period.

7. The trip may be bypassed when the reactor power is < 30% of rated. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

8. For special stability tests, the APRM rod block shall be $\leq$ 0.66W + 75% for the duration of testing.

9. With the number of operable channels less than required by the minimum operable channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

10. With one RBM channel inoperable:

    a. Verify that the reactor is not operating on a limiting control rod pattern, and

    b. Restore the inoperable RBM channel to operable status within 24 hours.

    Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

TABLE 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCIES

EMERGENCY CORE COOLING ACTUATION INSTRUMENTATION

Core Spray System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
High Drywell Pressure	(Note 1)	Once/Operating Cycle	Once Each Day
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
Low Reactor Pressure	(Note 1)	Once/Operating Cycle	-----
Pump 14-1A, Discharge Press	(Note 1)	Every 3 Months	-----
Auxiliary Power Monitor	(Note 1)	Every Refueling	Once Each Day
Pump Bus Power Monitor	(Note 1)	None	Once Each Day
High Sparger Pressure	(Note 1)	Every 3 Months	-----
Trip System Logic Except Relays 14A-K11A 14A-K11B 14A-K19A 14A-K19B	Every 6 Months (Note 2)	Every 6 Months (Note 3)	-----

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TABLE 4.2.1 (continued)

Low Pressure Coolant Injection System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low Reactor Pressure #1	(Note 1)	Once/Operating Cycle	----
High Drywell Pressure #1	(Note 1)	Once/Operating Cycle	Once Each Day
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
Reactor Vessel Shroud Level	(Note 1)	Every 3 Months	----
Low Reactor Pressure #2	(Note 1)	Once/Operating Cycle	----
RHR Pump Discharge Pressure	(Note 1)	Every 3 Months	----
High Drywell Pressure #2	(Note 1)	Every 3 Months	----
Low Reactor Pressure #3	(Note 1)	Every 3 Months	----
Auxiliary Power Monitor	(Note 1)	Every Refueling Outage	Once Each Day
Pump Bus Power Monitor	(Note 1)	None	Once Each Day
LPCI Crosstie Monitor	None	None	Once Each Day
Trip System Logic	Every 6 Months (Note 2)	Every 6 Months (Note 3)	----

TABLE 4.2.1 (continued)

High Pressure Coolant Injection System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
Low Condensate Storage Tank Water Level	(Note 1)	Every 3 Months	----
High Drywell Pressure	(Note 1)	Once/Operating Cycle	Once Each Day
High Suppression Chamber Water Level	(Note 1)	Every 3 Months	----
Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic	Every 6 Months (Note 2)	Every 6 Months (Note 3)	----

TABLE 4.2.1 (continued)

Automatic Depressurization System			
Trip Function	Functional Test(8)	Calibration(8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
High Drywell Pressure	(Note 1)	Once/Operating Cycle	Once Each Day
Bus Power Monitor	(Note 1)	None	Once Each Day
Trip System Logic (Except Solenoids of Valves)	Every 6 Months (Note 2)	Every 6 Months (Note 3)	-----

TABLE 4.2.2

MINIMUM TEST AND CALIBRATION FREQUENCIES

PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

Trip Function	Functional Test (8)	Calibration (8)	Instrument Check
Low-Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	Once Each Day
High Steam Line Area Temperature	(Note 1)	Each Refueling Outage	----
High Steam Line Flow	(Note 1)	Every 3 Months	Once Each Day
Low Main Steam Line Pressure	(Note 1)	Every 3 Months	----
Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	----
High Main Steam Line Radiation	(Notes 1 & 7)	Each Refueling Outage	Once Each Day
High Drywell Pressure	(Note 1)	Once/Operating Cycle	Once Each Day
Condenser Low Vacuum	(Note 1)	Every 3 Months	----
Trip System Logic Except Relays 16A-K13 16A-K14 16A-K15 16A-K16 16A-K26 16A-K27	Every 6 Months (Note 2)	Every 6 Months (Note 3)	----

TABLE 4.2.3

MINIMUM TEST AND CALIBRATION FREQUENCIES

REACTOR BUILDING VENTILATION AND STANDBY GAS TREATMENT SYSTEM ISOLATION

Trip Function	Functional Test[8]	Calibration[8]	Instrument Check
Low Reactor Vessel Water Level	(Note 1)	Once/Operating Cycle	----
High Drywell Pressure	(Note 1)	Once/Operating Cycle	----
Reactor Building Vent Exhaust Radiation	Monthly	Every 3 Months	Once Each Day
Refueling Floor Zone Radiation	Monthly	Every 3 Months	Once Each Day During Refueling
Reactor Building Vent Trip System Logic	Every 6 Months (Note 2)	Every 6 Months (Note 3)	----
Standby Gas Treatment Trip System Logic	Every 6 Months (Note 2)	Every 6 Months (Note 3)	----
Logic Bus Power Monitor	(Note 1)	None	Once Each Day

3.4 LIMITING CONDITIONS FOR OPERATION

B. **Operation with Inoperable Components**

From and after the date that a redundant component is made or found to be inoperable, reactor operation is permissible during the succeeding seven days unless such component is sooner made operable.

C. **Liquid Poison Tank - Boron Concentration**

The liquid poison tank shall contain a boron bearing solution that satisfied the volume concentration requirements of Figure 3.4.1 and the solution temperature, including that in the pump suction piping, shall be not less than the temperature presented in Figure 3.4.2.

D. If Specification 3.4 A or B are not met an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

E. If Specification 3.4 .C is not met action shall be immediately initiated to correct the deficiency. If at the end of 12 hours the system has not been restored to full operability, then a shutdown shall be initiated with the reactor in cold shutdown within 24 hours of initial discovery.

4.4 SURVEILLANCE REQUIREMENTS

Disassemble and inspect one explosion valve so that it can be established that the valve is not clogged. Both valves shall be inspected in the course of two operating cycles.

Test that the setting of the system pressure relief valves is between 1400 and 1490 psig.

B. **Operation with Inoperable Components**

When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

C. **Liquid Poison Tank - Boron Concentration**

The solution volume and temperature in the tank shall be checked at least daily.

Boron concentration shall be determined at least once a month and at any time water or boron are added or if the solution temperature drops below the limits specified by Figure 3.4.2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 76 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 January 10, 1983, Vermont Yankee Nuclear Power Corporation (the licensee) proposed changes to the Technical Specifications pertaining to limiting conditions of operation and surveillance requirements related to modifications of the scram discharge system and the analog trip system.

2.0 Evaluation

2.1 Scram Discharge System Modifications

As a result of events involving common cause failures of Scram Discharge Volume (SDV) limit switches and SDV drain valve operability, the NRC staff issued IE Bulletin 80-14 on June 12, 1980. In addition, the staff sent a letter dated July 7, 1980 to all operating BWR licensees requesting that they propose Technical Specification changes to provide surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements on SDV limit switches. The licensee's submittal dated October 5, 1981, contained the proposed Technical Specifications requested by the staff, which were made part of the Vermont Yankee license by License Amendment No. 73 dated November 29, 1982. As described in the Safety Evaluation accompanying License Amendment No. 73, the licensee planned to install a second instrument volume and provide four reactor protection system (RPS) level instruments for each of the two instrument volumes, for a total of eight instruments for the RPS. The second instrument volume significantly improves the design and reliability of the SDV.

During the March 1983 refueling outage, the licensee is performing the planned modifications to provide two independent instrument volumes. Each volume will be monitored by four level transmitters. Signals from these transmitters feed into analog to digital trip units which provide reactor scram signals, when the appropriate level is reached in either instrument volume. These analog instrument channels replace the float switches utilized in the previous design. In addition to the scram signals described above, one of the transmitters from each instrument volume feeds a separate analog to digital trip unit which provides a signal to the rod block actuation system.

The numerical trip point for the high scram discharge volume water level scram is lower than previous, due to the configuration of the new piping. However, since only half as many control rod drive units discharge into each volume, the reduced capacity should not create spurious signals. The numerical trip point for the control rod block will remain the same as it has been.

The surveillance and calibration requirements have been modified to agree with requirements which have previously been approved for similar analog instrumentation utilized at Vermont Yankee, as provided in License Amendment No. 58, dated November 3, 1980.

Based upon our review of the licensee's January 10, 1983 submittal, we conclude that the licensee's proposed Technical Specifications satisfy staff requirements for limiting conditions of operation and surveillance requirements for SDV level instrumentation. Consequently, we find the licensee's proposed Technical Specifications for SDV level instrumentation acceptable.

2.2 Analog Trip System

By License Amendment No. 58, dated November 3, 1980, the staff approved installation at Vermont Yankee of improved safety system instrumentation referred to as the analog trip system. This analog trip system is similar to that developed by General Electric (GE) and described in GE's Topical Report NEDO-21617 of April 1977 and NEDO-21617-1 of January 1978 entitled, "Analog Transmitter/Trip Unit System" (ATTUS). GE submitted this topical report to the NRC staff for review and it was found acceptable by the staff as stated in the letter to GE dated June 27, 1978.

At the time of the original installation of the analog trip system, the licensee converted certain reactor pressure and water level loops to analog loops. The licensee is now converting from pressure switch to analog loop design for drywell pressure instrumentation inputs to the reactor protection and emergency core cooling systems and reactor pressure instrumentation for the emergency core cooling low pressure permissive.

The proposed Technical Specifications change the calibration frequency for channels converted to the analog trip system from once per three months to once per operating cycle. The justification given for this change is that operating experience indicates that calibration drift and mechanical problems associated with mechanical float switches are not experienced with the analog equipment. Also, the overall accuracy and repeatability of the analog instrumentation is significantly better than the equipment it will replace. We agree with this justification. We consider that the daily instrument check and the monthly functional test of the analog system provides further assurance of the operability of this system when compared to the surveillance required for the system which it replaced. We approved identical changes in calibration frequency for channels converted to the analog system in License Amendment No. 58, and we also find the revised surveillance requirements to be acceptable for this proposed Technical Specification change.

3.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability of consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Dated: March 28, 1983

Principal Contributor: V. Rooney

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 76 to Facility Operating License No. DPR-28, issued to Vermont Yankee Nuclear Power Corporation which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station (the facility) located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications with respect to limiting conditions of operation and surveillance requirements for the scram discharge system and the analog trip system.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to shia ction, see (1) the application for amendment dated January 10, 1983, (2) Amendment No. 76 to License No. DPR-28, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 28th day of March 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

D Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing