Docket File, D: 3-016

November 27, 1981

Docket No. 50-271

RECEIVED
DEC 3 1981
U.S. NUCLEAR REGULATORY COMMISSION

Mr. Robert L. Smith
Licensing Engineer
Vermont Yankee Nuclear Power
Corporation
1671 Worcester Road
Framingham, Massachusetts 01701

Dear Mr. Smith:

The Commission has issued the enclosed Amendment No. 70 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment is in response to your application of September 2, 1981 and supplemental information dated October 28 and 30 and November 6, 13, 23 and 23, 1981.

The amendment changes the Technical Specifications to incorporate the limiting conditions for operation and surveillance requirements associated with Cycle 9 operation. The Cycle 9 analysis for Vermont Yankee is based on both General Electric and Yankee Atomic Electric methods. Although our review of Yankee Atomic Electric methods is not yet complete for all of cycle 9 and for future reloads, the review is sufficiently complete to approve the use of these methods to support operation of the Vermont Yankee plant to a cycle exposure of EOC-2 GWD/T. In order to permit our timely review of operation out to the end of cycle 9, you are required to submit core wide transient analyses using methods acceptable for the end of cycle 9, by March 31, 1982.

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. 70 to DPR-28
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

Distribution:	Docket File	NRC PDR	Local PDR	ORB#2 Rdg	D. Eisenhut
S. Norris	V. Rooney	OELD	I&E(4)	T. Barnhart(4)	L. Schneider
D. Brinkman	ACRS(10)	OPA(C. Miles)		R. Diggs	NSIC
TERA	ASLAB	Gray	Extra(5)		

OFFICE	ORB#2	ORB#2	ORB#2	DL:AD:OR	OELD	FR Notice + Amendment only	
SURNAME	SNorris	VRooney:pob	TIppolito	TNovak	R. Bachmann		
DATE	11/27/81	11/27/81	11/27/81	11/27/81	11/27/81		

8201050094 811127
PDR ADOCK 05000271
P PDR

cc:

Mr. W. F. Conway
President & Chief Operating Officer
Vermont Yankee Nuclear Power Corp.
411 Western Avenue
Drawer 2
West Brattleboro, Vermont 05301

Mr. Louis Heider, V. P.
Vermont Yankee Nuclear Power Corp.
25 Research Drive
Westboro, Massachusetts 05181

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

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

Mr. Raymond H. Puffer
Chairman
Board of Selectman
Vernon, Vermont 05354

W. P. Murphy, Plant Superintendent
Vermont Yankee Nuclear Power Corp.
P.O. Box 157
Vernon, Vermont 05354

Brooks Memorial Library
224 Main Street
Brattleboro, Vermont 05301

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

Mr. E. W. Jackson
Manager of Operations
Vermont Yankee Nuclear Power Corp.
411 Western Avenue
Drawer 2
West Brattleboro, Vermont 05301

U.S. Environmental Protection Agency
Region I Office
Regional Radiation Representative
JFK Federal Building
Boston, Massachusetts 02203

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

Vermont Yankee Decommissioning
Alliance
53 Frost Street
Brattleboro, Vermont 05301

Vermont Yankee Decommissioning
Alliance
5 State Street
Box 1117
Montpelier, Vermont 05602

Resident Inspector
c/o U.S. NRC
P.O. Box 176
Vernon, Vermont 05453

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

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. 70
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 September 2, 1981 as supplemented, 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:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B as revised through Amendment No. 70 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8201050098 811127
PDR ADOCK 05000271
P PDR

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 27, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 70

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 pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
2	2
2a	
3	3
4	4
39a	39a
72	72
	72a
73	73
76	76
180d	180d
180f	180f
180h	180h
180n	180n
180n3	180n3
180n5	180n5
180-01	180-01
197	197

G. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor, to verify the proper instrument channel response, alarm, and/or initiating action.

H. Log System Functional Test - A logic system functional test means a test of all relays and contacts of a logic circuit from sensor to activated device to insure all components are operable per design intent. Where possible, action will go to completion, i.e., pumps will be started and valves opened.

I. Minimum Critical Power Ratio - The Minimum Critical Power Ratio is defined as the ratio of that power in a fuel assembly which is calculated to cause some point in that assembly to experience boiling transition as calculated by application of the GEXL correlation to the actual assembly operating power. (Reference NEDO-10958)

J. Mode - The reactor mode is that which is established by the mode-selector-switch.

K. Operable - A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

L. Operating - Operating means that a system or component is performing its intended functions in its required manner.

M. Operating Cycle - Interval between the end of one refueling outage and the end of the next subsequent refueling outage.

N. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure supression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or deactivated in the isolated position.

4. All blind flanges and manways are closed.

O. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

P. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady state power level of 1593 thermal megawatts.

Q. Rated Thermal Power - Rated thermal power means a steady state power level of 1593 thermal megawatts.

R. Reactor Power Operation - Reactor power operation is any operation with the mode switch in the "Startup/Hot Standby" or "Run" position with the reactor critical and above 1% rated thermal power.

1. Startup/Hot Standby Mode - In this mode the low turbine condenser volume trip is bypassed when condenser vacuum is less than 12 inches Hg and both turbine stop valves and bypass valves are closed; the low pressure and the 10 percent closure main steamline isolation valve closure trips are bypassed; the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service and APRM neutron monitoring system operable.

2. Run Mode - In this mode the reactor system pressure is equal to or greater than 850 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

S. Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

T. Refueling Outage - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

U. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All reactor building automatic ventilation system isolation valves are operable or are secured in the isolated position.

V. Shutdown - The reactor is in a shutdown condition when the reactor mode switch is in the shutdown mode position and no core alterations are being performed. When the mode switch is placed in the shutdown position a reactor scram is initiated, power to the control rod drives is removed, and the reactor protection system trip systems are de-energized.

1. Hot Shutdown means conditions as above with reactor coolant temperature greater that 212°F.

2. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

3. Shutdown means conditions as above such that the effective multiplication factor (K_{eff}) of the core shall be less than 0.99.

W. Simulated Automatic Actuation - Simulated automatic actuation means applying a simulated signal to the sensor to actuate circuit in question.

X. Transition Boiling - Transition boiling means the boiling regime between nucleate and film boiling. Transition boiling is the regime in which both nucleate and film boiling occur intermittently with neither type being completely stable.

Y. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted plus 25%. The total maximum combined interval time for any three consecutive tests shall not exceed 3.25 times the specified interval. The operating cycle interval is considered to be 18 months and the tolerances stated above are applicable.

Z. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests unless otherwise stated in these specifications may be waived when the instrument, component, or system is not required to be operable, but these tests shall be performed on the instrument, component, or system prior to being required to be operable.

AA. Vital Fire Suppression Water System - The vital fire suppression water system is that part of the fire suppression system which protects those instruments, components, and systems required to perform a safe shutdown of the reactor. The vital fire suppression system includes the water supply, pumps, and distribution piping with associated sectionalizing valves, which provide immediate coverage of the Reactor Building, Control Room Building, and Diesel Generator Rooms.

TABLE 3.2.1

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

Recirculation Pump Trip - A & B (Note 1)

Minimum Number of Operable Instrument Channels per Trip System	Trip Function	Trip Level Setting	Required Action When Minimum Conditions for Operation are not satisfied
2	Low-Low Reactor Vessel Water Level	≥ 6' 10.5" above top of active fuel	Note 2
2	High Reactor Pressure	≤ 1150 psig	Note 2
2	Time Delays	≤ 10 seconds	Note 2
1	Trip Systems Logic	---	Note 2

Amendment No. ~~58~~, ~~68~~, 70

3.3 LIMITING CONDITIONS FOR OPERATION

C. Scram Insertion Times

1.1 The average scram time, based on the de-energization of the scram pilot valve solenoids of all operable control rods in the reactor power operation condition shall be no greater than:

Drop-Out of Position	%Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	0.358
36	25.34	0.912
26	46.18	1.468
06	87.84	2.686

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Drop-Out of Position	%Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	0.379
36	25.34	0.967
26	46.18	1.556
06	87.84	2.848

4.3 SURVEILLANCE REQUIREMENTS

C. Scram Insertion Times

1. After refueling outage and prior to operation above 30% power with reactor pressure above 800 psig all control rods shall be subject to scram-time measurements from the fully withdrawn position. The scram times for single rod scram testing shall be measured without reliance on the control rod drive pumps.

2. During or following a controlled shutdown of the reactor, but not more frequently than 16 weeks nor less frequently than 32 weeks intervals, 50% control rod drives in each quadrant of the reactor core shall be measured for scram times specified in Specification 3.3.C. All control rod drives shall have experienced scram-time measurements each year. Whenever 50% of the control rod drives scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drives performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the start up test report.

3.3 LIMITING CONDITIONS FOR OPERATION | 4.3 SURVEILLANCE REQUIREMENTS

1.2 If Specification 3.3.C.1.1 cannot be met, the average scram time, based on the de-energization of the scram pilot valve solenoids of all operable control rods in the reactor power operation condition shall be no greater than:

Drop-Out of Position	%Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	.358
36	25.34	1.096
26	46.18	1.860
06	87.84	3.419

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

Drop-Out of Position	%Inserted From Fully Withdrawn	Avg. Scram Insertion Time (sec)
46	4.51	.379
36	25.34	1.164
26	46.18	1.971
06	87.84	3.624

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

3.3 LIMITING CONDITIONS FOR OPERATION

3. If Specification 3.3.C.1.2 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be considered inoperable, fully inserted into the core, and electrically disarmed.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

4.3 SURVEILLANCE REQUIREMENTS

D. Control Rod Accumulators

Once a shift check the status of the pressure and level alarms for each accumulator.

3.3 (cont'd)

B. Control Rods

1. Control rod dropout accidents as discussed in the FSAR can lead to signifcant core damage. If coupling integrity is maintained, the possiblity of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive.

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.

Bases:

3.11 Fuel Rods

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

Refer to Section 5.5.2 of NEDE-24011P, Amendment 3, dated March 1978.

(Note: All exposure increments in this Technical Specification Section are expressed in terms of megawatt-days per short ton.)

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 1.

Table 1

SIGNIFICANT INPUT PARAMETERS TO THE
LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core Thermal Power	1664 MWt, which corresponds to 105% of rated steam flow
Vessel Steam Output	6.75×10^6 lbm/h, which corresponds to 105% of rated steam flow
Vessel Steam Dome Pressure	1055 psia
Recirculation Line Break Area for Large Breaks - Discharge	2.26 ft^2 (DBA)
- Suction	4.14 ft^2
Number of Drilled Bundles	220

Fuel Parameters:

	Fuel Type	Fuel Bundle Geometry	Peak Technical Specification Linear Heat Generation Rate (kW/ft)	Design Axial Peaking Factor	Initial Minimum Critical Power Ratio*
A.	7D230	7 x 7	18.5	1.4	1.2
B.	8D219	8 x 8	13.4	1.4	1.2
C.	8D274L	8 x 8	13.4	1.4	1.2
D.	8D274H	8 x 8	13.4	1.4	1.2
E.	8D274 (High Gd)	8 x 8	13.4	1.4	1.2
F.	LTA	8 x 8	13.4	1.4	1.2
G.	8DPB289 & P8DPB289	8 x 8	13.4	1.4	1.2

*To account for the 2% uncertainty in bundle power required by Appendix K, the SCAT calculation is performed with an MCPR of 1.18 (i.e., 1.2 divided by 1.02) for a bundle with an initial MCPR of 1.20.

Bases:

3.11C Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

1. The MCPR Operating Limit is a cycle dependent parameter which can be determined for a number of different combinations of operating modes, initial conditions, and cycle exposures in order to provede reasonable assurance against exceeding the fuel cladding integrity safety limit (FCISL) for potential abnormal occurences. The MCPR operating limits are presented in Appendix A of the current cycle's Core Performance Analysis report.

2. In order to counteract the postulated thermal margin degradation for the worst-case Fuel Loading Error accident, a higher MCPR operating limit is applied in the event air ejector off-gas radiation exceeds levels that could be associated with a mis-load fuel assembly.

VYNPS

Table 3.11-2

MCPR Operating Limits (5)

Value of "N" in RBM Equation(1)	Average Control Rod Scram Time	Cycle Exposure Range	MCPR Operating Limit for Fuel Type(2) 8X8	8X8R	P8X8R
42%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.29	1.29	1.29
		EOC-2 GWD/T to EOC-1 GWD/T	1.29	1.29	1.29
		EOC-1 GWD/T to EOC	1.29	1.29	1.29
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.29	1.29	1.29
		EOC-2 GWD/T to EOC-1 GWD/T	1.30	1.30	1.30
		EOC-1 GWD/T to EOC	1.33	1.32	1.32
41%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.25	1.25	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.25	1.25	1.25
		EOC-1 GWD/T to EOC	1.27	1.27	1.27
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.25	1.25	1.25
		EOC-2 GWD/T to EOC-1 GWD/T	1.30	1.30	1.30
		EOC-1 GWD/T to EOC	1.33	1.32	1.32
≤40%	Equal or better than L.C.O. 3.3 C.1.1	BOC to EOC-2 GWD/T	1.24	1.24	1.24
		EOC-2 GWD/T to EOC-1 GWD/T	1.24	1.24	1.24
		EOC-1 GWD/T to EOC	1.27	1.27	1.27
	Equal or better than L.C.O. 3.3 C.1.2	BOC to EOC-2 GWD/T	1.24	1.24	1.24
		EOC-2 GWD/T to EOC-1 GWD/T	1.30	1.30	1.30
		EOC-1 GWD/T to EOC	1.33	1.32	1.32
75%	Special Testing at Natural Circulation (Note 3, 4)		1.30	1.31	1.31

(1) The Rod Block Monitor (RBM) trip setpoints are determined by the equation shown in Table 3.2.5 of the Technical Specifications.

(2) The current analyses for MCPR Operating Limits do not include 7x7 fuel. On this basis further evaluation of MCPR operating limits is required before 7x7 fuel can be used in Reactor Power Operation.

(3) For the duration of pump trip and stability testing.

(4) K_f factors are not applied during the pump trip and stability testing.

(5) Until further NRC approval is obtained, the following restrictions apply:
a) RBM value of "N" ≥ 41% shall be used
b) Operation shall only be allowed until EOC-2 GWD/T.

180-01

Table 3.11-1B

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee Fuel Type: 8D219

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200.0	11.4	2053.	0.021
1,000.0	11.5	2061.	0.021
5,000.0	11.9	2117.	0.023
10,000.0	12.1	2164.	0.026
15,000.0	12.3	2192.	0.029
20,000.0	12.1	2189.	0.029
25,000.0	11.3	2077.	0.020
30,000.0	10.2	1933.	0.012
35,000.0	9.6	1704.	0.004

Source: NEDO-21697, August 1977 (revised)

Amendment No. ~~30~~, ~~41~~, 70

VYNPS

Table 3.11-1E

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee   Fuel Type: 8D274 (High Gd)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (^{o}F)	Oxidation Fraction
200.0	11.1	2053.	0.019
1,000.0	11.1	2044.	0.018
5,000.0	11.6	2092.	0.021
10,000.0	12.1	2141.	0.024
15,000.0	12.2	2165.	0.026
20,000.0	12.1	2170.	0.027
25,000.0	11.6	2119.	0.023
30,000.0	10.6	1993.	0.015
35,000.0	10.0	1751.	0.005
40,000.0	9.4	1671.	0.004

Source: NEDO-21697, August 1977 (revised)

Table 3.11-1G

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee Fuel Type: 8DPB289 & P8DPB289

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200.0	11.2	2126.	0.027
1,000.0	11.2	2119.	0.026
5,000.0	11.8	2178.	0.030
10,000.0	12.0	2185.	0.030
15,000.0	12.1	2200.	0.032
20,000.0	11.8	2187.	0.031
25,000.0	11.3	2120.	0.025
30,000.0	11.1	2095.	0.023
35,000.0	10.4	1862.	0.008
40,000.0	9.8	1784.	0.006

Source: NEDO-21697, August 1977 (revised)

d. Power Plant Design

e. Reactor Engineering

f. Radiation Safety

g. Safety Analysis

h. Instrumentation and Control

i. Metallurgy

3. Meeting Frequency: Semi-annually and as required on call of the Chairman.

4. Quorum: Chairman or Vice Chairman plus four members or designated alternates.

5. Responsibilities:

a. Review proposed changes to the operating license including Technical Specifications.

b. Review minutes of meetings of the Plant Operation Review Committee to determine if matters considered by that committee involve unreviewed or unresolved safety questions.

c. Review the safety evaluations for changes to equipment or systems completed under the provisions of Section 50.59 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

d. Periodic audits of plant operations, and audits of the facility fire protection program and implementing procedures shall be performed under cognizance of the Committee.

e. Investigate all reported instances of violations of Technical Specifications, reporting findings and recommendations to prevent recurrence to the Manager of Operations.

f. Perform special reviews and investigations and render reports thereon as requested by the Manager of Operations.

g. Review proposed tests and experiments and results thereof when applicable.

h. Review abnormal performance of plant equipment and anomalies.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 70 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 September 2, 1981 (Reference 1) Vermont Yankee Nuclear Power Corporation (VYNPC or licensee) has proposed changes to the Technical Specifications of the Vermont Yankee Nuclear Power Station (VY), as supported by Reference 2. The proposed changes relate to the core for Cycle 9 operation. Although the Cycle 9 reload involves the replacement of a number of irradiated fuel assemblies, the application does not involve physical changes to the fuel system design. The safety analysis of Cycle 9 reload does, however, involve the first-time application of a number of analytical methods. These methods are described and supported in references 3-5, 9, 13-15, 17, 19, 22-24, 26 and 27.

2.0 Evaluation

2.1 Fuel

2.1.1 Fuel Mechanical Design

The Vermont Yankee Cycle 9 reload involves the insertion of 120 new fuel bundles of the pressurized retrofit 8x8 design. These assemblies will be irradiated with a number of other pressurized retrofit 8x8, unpressurized retrofit 8x8, and unpressurized standard 8x8 fuel assemblies already resident in the core. All fuel assemblies were fabricated by the General Electric Company. These three interchangeable fuel designs have been approved for the previous cycle of operation at Vermont Yankee as well as at other boiling water reactors (BWRs). Cycle 9 involves no physical changes to the fuel design and is, therefore, acceptable.

2.1.2 Fuel Thermal Design

The Cycle 9 fuel thermal performance analysis was performed with a new Yankee Atomic computer code called FROSSTEY (refs. 3-5). This code was used to calculate (a) incipient fuel centerline melt limits, (b) 1 percent cladding strain limits, (c) core average gap conductance and core average fuel temperature for initializing non-LOCA transient analyses, and (d) average gap conductance and average fuel temperature of the peak bundle for initializing hot channel calculations. Previous analyses provided by General Electric were used for the remainder of the fuel thermal analysis, including LOCA initial conditions.

Our review of the FROSSTEY code is not yet complete. Because this code was used to only a limited extent in the Vermont Yankee Cycle 9 analysis, we have reviewed the specific Cycle 9 fuel conditions predicted by the FROSSTEY code, rather than the generic methods employed by the code to predict these conditions. This has allowed us to make a finding on the Cycle 9 submittal without completing our review of the generic methods to be employed in later cycles.

We have compared local fuel integrity limits calculated with FROSSTEY with those reported (ref. 6) by General Electric for the same conditions and find them to be similar, with slightly higher centerline melt and slightly lower cladding strain limits being reported by Yankee Atomic. Based on the similarity of the Yankee Atomic and General Electric (previously approved) results, and on the fact that these fuel integrity conditions are not limiting for Cycle 9 operation, we find these results acceptable.

The FROSSTEY code was also used to calculate core average and hot channel average gap conductance and fuel temperature. We have audited these calculations using the results of an NRC fuel performance code called GAPCON-THERMAL-2 (ref. 7) as reported in NUREG-0559 (ref. 8).

The results of our calculations are very similar to those predicted by the FROSSTEY code, our code predicting slightly lower fuel temperatures and a slightly greater variation in gap conductance as a function of burnup. As a result, we conclude that the average gap conductance and fuel temperature predictions used in the Cycle 9 analysis are acceptable. Because our calculations were based on Cycle 9 specific conditions, our approval does not apply to subsequent cycles of operation at Vermont Yankee.

Other fuel performance analyses used in the Vermont Yankee Cycle 9 Reload Report rely on General Electric generic analyses. These include fuel mechanical design, maximum linear heat generation rate (MLHGR, which remains at 13.4 KW/ft) and LOCA initial conditions. These analyses are unchanged from those previously accepted for Vermont Yankee with the exception of the maximum average planar linear heat generation rate (MAPLHGR) operating limits. The Cycle 9 MAPLHGR limits, which are documented in Reference 9, have been extended to 40,000 MWd/t for some fuel types.

The General Electric Company has requested (refs. 10-11) that credit for approved, but unapplied, ECCS evaluation model changes be used to avoid MAPLHGR penalties on operating reactors due to high burnup fission gas release. Such penalities would apply to Vermont Yankee for burnups beyond 30,000 MWd/t. However, we have conditionally accepted (ref. 12) the GE proposal and, by letter of November 9, 1981 (ref. 13), the licensee has subscribed to the conditions of our approval. We, therefore, find the Vermont Yankee Cycle 9 MAPLHGR limits acceptable as submitted.

2.1.3 Conclusions

The NRC staff has reviewed those sections of the reload report for Vermont Yankee Cycle 9 dealing with the fuel system design and we find those portions of the application acceptable.

2.2 Physics Considerations

The nuclear design analysis of the core was performed with the SIMULATE code (ref. 14) with input parameters calculated with the CASMO code (ref. 15). VYNPC has provided reports which describe these codes and the analyses carried out to verify them for use. These reports are currently in review by the staff and evaluation reports will be issued. Sufficient review has been completed to permit the conclusion that they are acceptable for use in the analysis of the Vermont Yankee Plant. This conclusion is based on comparisons of the results of calculations using these codes with measured data including that obtained from the first seven cycles of the Vermont Yankee reactor. These comparisons showed that power distributions, hot and cold reactivities, shutdown margins and reactivity parameters were calculated to within accuracies and precisions comparable to those of other codes and techniques employed in the industry.

The licensee has provided a description of the core loading for Cycle 9 as well as analyses of anticipated power distributions, end of cycle exposure distributions, shutdown margin values and cycle kinetics parameters. Burnup calculations have been performed for both rodded and unrodded depletions. The end of cycle power distribution shows acceptable peaking factors for both rodded and unrodded depletion. The minimum shutdown margin during the cycle was calculated to be 0.78 percent Δk, an acceptable value. The cycle kinetics parameters are similar to those for earlier cycles and are acceptable.

We have reviewed the analyses of the rod withdrawal error, fuel misloading event and the rod drop accident. The analysis procedures employed by VYNPC are the same as those in current use for other operating boiling water reactors and are acceptable.

A bounding analysis of the rod withdrawal error is performed. A fully inserted high worth rod is assumed to be withdrawn continuously. An assembly near the withdrawn rod is assumed to be operating on Technical Specification limits at the time of the withdrawal. For the analysis, the maximum number of LPRMs (which make up the inputs to the Rod Block Monitor) permitted by the Technical Specification is assumed to be inoperable. The response of the Rod Block Monitor is then calculated as a function of the distance the rod is withdrawn. When the rod block setpoint is reached, the rod is assumed to travel an additional two inches and then to stop at the next notch. The resulting change in CPR is then added to the safety limit value and the required MCPR operating limit for this event is obtained. For Cycle 9 of Vermont Yankee this value (1.29) is the most limiting value of this quantity (for the "measured" scram time) and establishes the operating MCPR value for the cycle. We conclude that the analysis of the rod withdrawal error is acceptable.

Two types of fuel misloading events are analyzed - misorienting an assembly in its proper location and mislocating a properly oriented bundle. In the first (misoriented)event the bundle may be rotated by 90^{o} or 180^{o} from its normal orientation. The worst case is chosen and the increase in linear heat generation rate (due to the presence of higher enrichment rods near the wide water gap) and the decrease in critical power ratio (due to the effect of the tilting of the assembly and the change in local power distribution on the R factor) is determined for a large number of core locations. The limiting case is chosen and the operating MCPR limit required to prevent violating the MCPR safety limit when the misoriented bundle is placed on operating limits is obtained. For Cycle 9 this value is 1.24 which is smaller than that required for the rod withdrawal event. The resulting linear heat generation limit is 17.5 kw/ft which is less than the 1 percent strain limit. We conclude that the analysis of the assembly misoriented event is acceptable.

The analysis of the mislocated bundle follows procedures used for other operating boiling water reactors and is acceptable. The procedure starts by substituting the higher enrichment reload bundle for various high burnup bundles throughout the core to obtain the highest change in CPR produced by the substitution. The ΔCPR is then added to the CPR values of all the bundles in the core at several times in the cycle. Some bundles in the core (the least reactive ones) will not violate the MCPR safety limit when the core is at full power. These are dropped from consideration. The procedure is repeated with the high enrichment bundle being substituted for the least reactive of the remaining bundles. Because these bundles have higher reactivity than the first group, the resulting ΔCPR will be smaller. The addition procedure is then repeated, additional bundles are dropped from consideration and the whole process is repeated. This iteration continues until all locations are shown to be above the MCPR safety limit or until a limiting location is identified. For Cycle 9 all locations are shown to be above the safety limit (1.07) if the operating MCPR limit is 1.24. We conclude that the analysis of this event is acceptable.

The analysis procedure for the rod drop accident is the same as that used for other boiling water reactors and is acceptable. The Vermont Yankee reactor employs the banked position withdrawal sequence which is enforced with the Rod Worth Minimizer. This sequence was examined for Cycle 9 and the maximum worth for a potential dropped rod was 0.86 percent. The generic analysis performed by General Electric for this event (which is applicable to the Vermont Yankee reactor) would yield a maximum fuel enthalpy of about 140 calories per gram for this rod worth - an acceptable value. We conclude that the analysis of this event is acceptable.

2.3 System Analysis

The licensee has provided its own analysis for Vermont Yankee cycle 9, independent of some parts of the GE modeling package, but making use of other parts. In particular, YAEC has developed a methodology for the analysis of Vermont Yankee full core transients making use of the RETRAN code in place of the ODYN code.

Ample comparisons of RETRAN with experimental results exist (ref. 16). The NRC staff regards RETRAN as a substantially correct coupling of thermal hydraulics and neutronics packages but will require more experience with its use before approving it generically (expected about September 1982). In the meantime the use of RETRAN in reload transient analyses is being reviewed on a case-by-case basis.

We have reviewed the use of RETRAN by VYNPC in Vermont Yankee Cycle 9 analysis to determine if the results and theory are sufficiently conservative when compared with experimental data and with previously accepted methods.

1. We have compared experimental results from the Peach Bottom turbine trip without bypass test and calculations using the Vermont Yankee methodology (ref. 17). The comparisons are close enough to produce a reasonable degree of confidence in the ability of the Vermont Yankee methods to reproduce this sort of transients and are best at the highest of the test powers (2275 MWt).

2. We have compared experimental results from a generator load rejection test performed at Vermont Yankee with calculations of the same transient using RETRAN. The overall effect on core power appears to be conservative.
3. We have considered comparisons based on a transient which simulates a turbine trip from full power without bypass flow, modeled on the Peach Bottom Unit 2 (ref. 18). The licensee has calculated this transient using Vermont Yankee methods (ref. 19) and compared the result with ODYN and BNL-TWIGL calculations.

4. We have compared Cycle 9 ΔCPR values with Cycle 8 ΔCPR values (ref. 20) which were calculated using the REDY code (no longer approved). The Cycle 9 ΔCPR values are more conservative than the corresponding calculated Cycle 8 values.

5. We have considered the results of comparisons between calculations using RETRAN and RELAP4/MOD 6. In general, good agreement between the RELAP4/MOD 6 results and VYNPC results using RETRAN 01/Cycle 15G and ANL results using RETRAN (ref. 21) was demonstrated.

In the course of this comparison, some of the steam line and steam dome modeling effects were studied. The passage of the pressure waves along the steam line is clearly apparent from the RETRAN calculation, indicating an acceptable nodalization and computation of compressibility effects. The nonequilibrium model of the steam dome chosen in the VYNPC calculation was also determined to be more conservative than an equilibrium model.

The theoretical basis for incorporating three-dimensional physics effects in the RETRAN point kinetics model (ref. 22) has been reviewed. The VYNPC calculations of the steady state core physics employ "state-of-the-art" methods for the 3D physics and for the collapse to equivalent 1D representation and then to a "0" D point kinetics representation for use in the

RETRAN transient analysis. Their use of the collapsed representation is quasi-1D, not truly 1D as a comparable ODYN calculation would be, nor is it simply point kinetics as the old REDY calculation was.

The comparison of the Peach Bottom tests results with the VYNPC calculations, mentioned above, indicates the conservative nature of the VYNPC methods.

The VYNPC method includes an implicit assumption that the set of steady state analyses adequately covers all possible transient conditions. Because of the strong feedback from the moderator density (boiling void) distribution and the changes that local voids can undergo during a transient, this general area requires further examination in the long term. VYNPC has not missed something that others have included, however, as this is potentially a problem that must also be faced by GE. VYNPC has provided assurances that for the transients analyzed in the Cycle 9 submittal, the snapshots used in their 3D calculations are representative of each specific transient analyzed. In comparison to the Peach Bottom tests, VYNPC methods produce results that are more conservative than those of ODYN and agree well with the data.

VYNPC has not at this time submitted ΔCPR values from their Peach Bottom transient calculations. There are difficulties in producing a ΔCPR calculation for Peach Bottom using the VYNPC models, but a calculation of this value is being considered by VYNPC. The NRC staff believes that this could be a useful comparison with the GE calculations and should be pursued.

The licensee has proposed to use the measured scram as a basis in the Technical Specifications for determining Operating Limit MCPR's. Technical Specifications require that these times are to be measured periodically. Sensitivity studies of the effect of scram time on ΔCPR have been included in their analysis supporting these Technical Specification changes. We find the methodology, as described above, and the proposed Technical Specification changes acceptable.

2.4 Core Thermal Hydraulics

In support of the Cycle 9 reload application, the licensee has submitted YAEC-1275, Vermont Yankee Cycle 9 Core Performance Analysis (ref. 2), which utilizes results from new core thermal hydraulic computer codes. These codes are FIBWR (YAEC-1234, ref. 23) for steady-state core flow distribution calculations and MAYU04-YAEC (YAEC-1235, Ref. 24) for rod bundle transient thermal hydraulic calculations. In addition, the EPRI void model is used for the two-phase void fraction calculation. The GEXL critical quality-boiling length correlation is used for critical power calculation. The staff's review evaluations are described as follows.

FIBWR Code

FIBWR is a steady-state thermal hydraulic analysis code which determines the flow and void distributions for a given power distribution and inlet flow conditions in a BWR core. The staff has reviewed the FIBWR code and concluded it is acceptable for Vermont Yankee reload analyses. The review of FIBWR will be addressed in a separate SER.

MAYUO4-YAEC Code

MAYUO4-YAEC (ref. 24) is a modified version of MAYUO4 (ref. 25) which analyzes one-dimensional single channel hydraulic and heat transfer transients in rod bundles. The modifications made to the original MAYUO4 include the use of the EPRI void model and the GEXL critical quality-boiling length correlation. MAYUO4-YAEC is used to calculate hot channel thermal margins under transient conditions with the transient input provided by the RETRAN system response analysis. MAYUO4-YAEC coupled with the GEXL correlation performs critical power ratio calculation for the reactor transients.

The staff has reviewed the MAYUO4-YAEC code and requested the licensee to provide comparisons using existing transient ATLAS 4x4 data and MAYUO4-YAEC with the EPRI-void model and GEXL correlation. For these comparisons the code predicted poorly with data in the high void fraction range. The licensee has determined that more work has to be done to identify the problem and fix the code. In the interim, the staff has concluded that MAYUO-4YAEC is not acceptable for thermal margin analysis and has informed the licensee that core wide transients using an acceptable code should be submitted by March 31, 1982.

In order to support reactor operation without relying on the results of MAYUO4-YAEC, the licensee by letter dated November 23, 1981 (ref. 26) demonstrated that the most limiting transient prior to EOC-2000 MWD/t is the local control rod withdrawal error transient. A comparison is made for the ΔCPR values between rod withdrawal error and loss of 100°F feedwater heating transient for fuel burnup from BOC to EOC-2000 MWD/t for the previous Cycles 6, 7, and 8. In all cases, the ΔCPR's for RWE with rod block monitoring (RBM) trip setpoint "N" values of 41 and 42 percent are always greater than the loss of feedwater heating transient. Since the loss of feedwater heating transient is the most limiting among the core wide transients prior to EOC-2000 MWD/t, this comparison shows that RWE is the most limiting transient for the previous cycles. Since Cycle 9 fuel loading is similar to the Cycle 8, it is reasonable to assume that the RWE and RBM setpoint N value no less than 41 percent is the most limiting transient prior to EOC-2000 MWD/t in Cycle 9.

The RWE transient analysis is performed with the three-dimensional SIMULATE code (ref. 14) using the GEXL correlation in a quasi-steady state approach. The review of the SIMULATE code is not complete, but has progressed sufficiently to conclude that SIMULATE is acceptable for Vermont Yankee Cycle 9 reload analysis. In order to justify the validity of the SIMULATE MCPR prediction, the licensee has provided a comparison (ref. 27) between SIMULATE and FIBWR calculations on the core flow distribution in various bundles. The results show excellent agreement between the two codes in bundle flow predictions. This demonstrates the adequacy of the SIMULATE code in predicting the thermal hydraulic conditions which are used in calculating MCPR's associated with the RWE transient. Therefore, the ΔCPR calculated by SIMULATE for the RWE transient is acceptable.

We have issued Technical Specifications which require that until further NRC approval is obtained, an RBM value of "N" greater than or equal to 41 percent shall be used, and operation shall only be allowed until EOC-2 GWD/t.

Based on the above observations, the staff concludes that the operating limit MCPR based on the RWE transient analysis with RBM trip setpoint "N" value no less than 41 percent is acceptable until EOD-2000 MWD/t.

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 or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, 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: November 27, 1981

REFERENCES

1. L. H. Heider (VYNPC) letter to the Office of Nuclear Reactor Regulation (NRC) dated September 2, 1981.

2. "Vermont Yankee Cycle 9 Core Performance Analysis," Yankee Atomic Electric Company Report YAEC-1275, August 1981.

3. S. P. Schultz and K. E. St. John, "Methods for the Analysis of Oxide Fuel Rod Steady-State Thermal Effects (FROSSTEY) Code/Model Description Manual," Yankee Atomic Electric Company Report YAEC-1249, April 1981.

4. S. P. Schultz and K. E. St. John, "Methods for the Analysis of Oxide Fuel Rod Steady-State Thermal Effects (FROSSTEY) code Qualification and Application," Yankee Atomic Electric Company Report YAEC-1265, June 1981.

5. D. C. Albright, "H2ODA: An Improved Water Properties Package," Yankee Atomic Electric Company Report YAEC-1237, March 1981.

6. "Generic Reload Fuel Application," General Electric Company Report NEDE-24011-P-A, July 1981.

7. C. E. Beyer et al, "GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Road," Battelle Pacific Northwest Laboratories Report BNWL-1898, November 1975.

8. D. L. Acey and J. C. Voglewede, "A Comparative Analysis of LWR Fuel Designs," U.S. Nuclear Regulatory Commission Report NUREG-0559, July 1980.

9. L. H. Heider (VYNPC) letter to Office of Nuclear Reactor Regulation (NRC) dated November 13, 1981.

10. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 6, 1981.

11. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 28, 1981.

12. L. S. Rubenstein (NRC) memorandum to T. M. Novak (NRC) on "Extension of General Electric Emergency Core Cooling Systems Performance Limits" dated June 25, 1981.

13. R. L. Smith (VYNPC) to T. A. Ippolito, "Additional Information on the Extension of Emergency Core Cooling System Performance Limits for Vermont Yankee," dated November 19, 1981.

14. D. M. Ver Planck, "Methods for the Analysis of Boiling Water Reactor Steady State Core Physics," Yankee Atomic Electric Company Report YAEC-1238, March 26, 1981.

15. E. E. Pilat, "Methods for Analyses of Boiling Water Reactors Lattice Physics," Yankee Atomic Electric Company Report YAEC-1232, December 31, 1980.

16. J. H. McFadden, et al, RETRAN Computer Code Manual, EPRI CCM - 5 Volume 4, Energy Incorporated, December 1978.

17. A. Ansari, et al, Methods for the Analysis of Boiling Water Reactors, A Systems Transient Analysis Model (RETRAN), YAEC-1233, April 1981.

18. M. S. Lu, et al, Analysis of Licensing Basis Transients for a BWR/4, BNL - NUREG-26684, September 1979.

19. J. T. Cronin, et al, Yankee Atomic Boiling Water Reactor Analysis Methods: Analysis of a Typical BWR/4 Turbine Trip Without Bypass Transient, YAEC-1280, October 1981.

20. Supplemental Reload Licensing Submittal for Vermont Yankee Nuclear Power Station - Reload #7, General Electric Company Y1003J01A02, July 1980.

21. Letter, P. Abramson, Argonne National Laboratory, to T. P Speis, NRC, Subject: Review of Vermont Yankee Cycle 9 Reload Analysis, with enclosure.

22. J. M. Holzer, et al, Methods for the Analysis of Boiling Water Reactors Transient Core Physics YAEC-1239P, August 1981 (proprietary).

23. YAEC-1234, "Methods for the Analysis of Boiling Water Reactors: Steady-State Core Flow Distribution Code (FIBWR)," letter from R. L. Smith to U.S. NRC dated December 31, 1980.

24. YAEC-1235, "Methods for the Analysis of Boiling Water Reactors: Transient Thermal Margin Analysis Code (MAYU04-YAEC)," dated December 1980.

25. W. C. Punches, "MAYU04: A Method to Evaluate Transient Thermal Hydraulic Conditions in Rod Bundles," GEAP-23517 dated March 1977.

26. Letter from D. E. Vandenburgh to U.S. NRC, "Justification for the Vermont Yankee MCPR Operating Limits, BOC to EOC-200 MWD/t," dated November 23, 1981.

27. Letter from D. E. Vandenburgh (VY) to T. A. Ippolito (NRC), "Validation of SIMULATE Thermal Hydraulics and Thermal Margin Calculations by Comparison to the FIBWR Code," dated November 23, 1981.

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. 70 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 changes the Technical Specifications to incorporate the limiting conditions for operation and surveillance requirements associated with Cycle 9 operation.

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 regulations. The Commission has made appropriate findings as required by the Act and the Commission's 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 Section 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 this action, see (1) the application for amendment dated September 2, 1981 as supplemented October 28 and 30 and November 6, 13, 23 and 23, 1981, (2) Amendment No. 70 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 27th day of November 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing