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Docket No. 50-271

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Mr. Robert L. Smith
Licensing Engineer
Vermont Yankee Nuclear Power
Corporation
25 Research Drive
Westboro, Massachusetts 01581

Dear Mr. Smith:

The Commission has issued the enclosed Amendment No. 61 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment is in response to your application of August 19, 1980, and supplemental information dated October 7 and 23, and November 21, 1980.

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The amendment changes the Technical Specifications to incorporate the limiting conditions for operation associated with Cycle 8 operation, a change in the factor for adjusting fuel power density calculations, a surveillance requirement for faster response from the reactor protective system, and certain administrative changes.

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

Sincerely,

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

Enclosures:
1. Amendment No. 61 to DPR-28
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

8101100712

MWH
DSI:RSB
M.W. Hodges
12/12/80

Core Performance Branch
W. Johnston
L.E. Phillips
12/12/80

legal review only of amendment + FR notice

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DATE	12/12/80	12/12/80	12/12/80	12/12/80	12/17/80	12/12/80

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

January 6, 1981

Docket No. 50-271

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Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: Vermont Yankee Nuclear Power Station
Vermont Yankee Nuclear Power Corp.

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

XX Other: Referenced docuemnts have veen provided to the PDR.

Enclosure:
As Stated

Office of Nuclear Reactor Regulation
Division of Licensing, ORB#2

OFFICE	ORB#2:DL					
SURNAME	SNorris:kf					
DATE	1-6-81					

NRC FORM 102 (1-76)

Docket
50-271

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

December 18, 1980

Docket No. 50-271

Mr. Robert L. Smith
Licensing Engineer
Vermont Yankee Nuclear Power
Corporation
25 Research Drive
Westboro, Massachusetts 01581

Dear Mr. Smith:

The Commission has issued the enclosed Amendment No. 61 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. The amendment is in response to your application of August 19, 1980, and supplemental information dated October 7 and 23, and November 21, 1980.

The amendment changes the Technical Specifications to incorporate the limiting conditions for operation associated with Cycle 8 operation, a change in the factor for adjusting fuel power density calculations, a surveillance requirement for faster response from the reactor protective system, and certain administrative changes.

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

Sincerely,

T. Ippolito

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

Enclosures:
1. Amendment No. 61 to DPR-28
2. Safety Evaluation
3. Notice

cc w/encl:
See next page

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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. 61
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 August 19, 1980, as supplemented October 7 and 23, and November 21, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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 Appendix A, as revised through Amendment No. 61 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

8101100717

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 18, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 61

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A as follows:

Remove the pages listed and replace with revised pages.

i	i
2	2
-	2a
5a	5a
6	6
14a	14a
14b	14b
18	18
25	25
31	31
70R	70R
71	71
76	76
77	77
135	135
136	136
179	179
180-01	180-01

TABLE OF CONTENTS

Amendment No. 61

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 activiated 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. Peaking Factor - The ratio of the fuel rod heat flux to the heat flux of an average rod in an identical geometry bundle operating at the average core power.

O. 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.

P. 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

1.1 SAFETY LIMIT

B. Core Thermal Power Limit (Reactor Pressure 800 psia or Core Flow 10% of Rated)

When the reactor pressure is 800 psia or core flow 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

2.1 LIMITING SAFETY SYSTEM SETTING

In the event of operation with the ratio of MFLPD to FRP greater than 1.0, the APRM gain shall be increased by the ratio: $\frac{MFLPD}{FRP}$

where: MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7 x 7 fuel and 13.4 KW/ft for 8 x 8 fuel.

FRP = fraction of rated power (1593 Mwt)

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

For no combination of loop recirculation flow rate and core thermal power shall the APRM flux scram trip setting be allowed to exceed 120% of rated thermal power.

b. Flux Scram Trip Setting (Refuel or Startup and Hot Standby Mode)

When the reactor mode switch is in the REFUEL or STARTUP position, average power range monitor (APRM) scram shall be set down to less than or equal to 15% of rated neutron flux. The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

B. APRM Rod Block Trip Setting

The APRM rod block trip setting shall be as shown in Figure 2.1.1 and shall be:

$$S_{RB} = 0.66W + 42\%$$

1.1 SAFETY LIMIT

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1A and and 1.1B is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 LIMITING SAFETY SYSTEM SETTING

where:

S_{RB} = Rod block setting in percent of rated thermal power 1593 MWt

W = percent rated drive flow where 100% rated drive flow is that flow equivalent to 48×10^6 lbs/hr core flow.

In the event of operation with the ratio of MFLPD to FRP greater than 1.0, the APRM gain shall be increased by the ratio: $\frac{MFLPD}{FRP}$

where: MFLPD = maximum fraction of limiting power density where the limiting power density is 18.5 KW/ft for 7 x 7 fuel and 13.4 Kw/ft for 8 x 8 fuel.

FRP = fraction of rated power (1593 Mwt)

In the event of operation with the ratio of MFLPD to FRP equal to or less than 1.0, the APRM gain shall be equal to or greater than 1.0.

APRM Flux Scram Trip Setting (Run Mode)

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of MFLPD and reactor core thermal power. If the scram requires a change due to an abnormal peaking condition, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1.A.1.a, thus assuring a reactor scram at lower than design overpower conditions.

Analyses of the limiting transients show that no scram adjustment is required to assure fuel cladding integrity when the transient is initiated from the operating limit MCPR (Specification 3.11C).

Flux Scram Trip Setting (Refuel or Startup & Hot Standby Mode)

For operation in the startup mode while the reactor is at low pressure, the reduced APRM scram setting to 15 percent of rated power provides adequate thermal margin between the setpoint and the safety limit, 25 percent of the rated. The margin is adequate to accomodate anticipated maneuvers associated with station startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimzer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5 percent of rated power per minute, and the APRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The reduced APRM scram remains active until the mode switch is placed in the RUN position. This switch can occur when reactor pressure is greater than 850 psig.

The IRM system consists of 6 chambers, 3 in each of the reactor protection system logic channels. The IRM is a 5-decade instrument which covers the range of power level between that covered by the SRM and the APRM. The 5 decades are covered by the IRM by means of a range switch and the 5 decades are broken down into 10 ranges, each being one-half of a decade in size. The IRM scram trip setting of 120/125 of full scale is active in each range of the IRM. For example, if the instrument were on range 1, the scram setting would be a 120/125 of full scale fo that range; likewise, if the instrument were on range 5, the scram would be 120/125 of full scale on that range. Thus, as the IRM is ranged up to accomodate the increase in power level, the scram trip setting is also ranged up. The most significant sources of reactivity change during the power increase are due to control rod withdrawal. For insequence control rod withdrawal, the rate of change of power is slow enough due to the physical limitation of withdrawing control rods, that heat flux is in equilibrium with the neutron flux and an IRM scram would result in a reactor shutdown well before any Safety Limit is exceeded.

In order to ensure that the IRM provided adequate protection against the single rod withdrawal error, a range of rod withdrawal accidents was analyzed. This analysis included starting the accident at various power levels. The most severe case invloves an initial condition in which the reactor is just subcritical and the IRM system is not yet on scale. This condition exists at quarter rod density. additional conservatism was taken in this analysis by assuming that the IRM channel closest to the withdrawn rod is bypassed. The results of this analysis show that the reactor is scrammed and peak power limited to one percent of rated power, thus maintaining MCPR above the fuel cladding integrity safety limit. Based on the above analysis, the IRM provides protection against local control rod withdrawal errors and continuous withdrawal of control rods in sequence.

B. APRM Rod Block Trip Setting

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at the fuel cladding integrity safety limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to contrl rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship, therefore the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power, because of the APRM rod block trip setting. The actual power distribuiton in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with the APRM scram trip setting, the APRM rod block trip setting must be adjusted downward if the ratio of MFLPD to FRP exceeds the specified value. If the APRM rod block requires a change due to abnormal peaking conditions, it will be accomplished by increasing the APRM gain by the ratio in Specification 2.1B, thus ensuring a rod block at lower than design overpower conditions.

C. Reactor Low Water Level Scram

The reactor low water level scram is set at a point which will prevent reactor operation with the steam separators uncovered, thus limiting carry-under to the recirculation loops. In addition, the safety limit is based on a water level below the scram point and therefore this setting is provided.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the operbility of plant instrumentation and control systems required for reactor safety.

Objective:

To specify the limits imposed on plant operation by those instrument and control systems required for reactor safety.

Specification:

A. Plant operation at any power level shall be permitted in accordance with Table 3.1.1. The system response time from the opening of the sensor contact up to and including the opening of the scram solenoid relay shall not exceed 50 milliseconds.

B. During operation with the ratio of MFLPD to FRP greater than 1.0 either:

   a. The APRM System gains shall be adjusted by the ratios given in Technical Specifications 2.1.A.1 and 2.1.B or

   b. The power distribution shall be changed to reduce the ratio of MFLPD to FRP.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the plant instrumentation and control systems required for reactor safety.

Objective:

To specify the type and frequency of surveillance to be applied to those instrument and control systems required for reactor safety.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

B. Once a day during reactor power operation the maximum fraction of limiting power density and fraction of rated power shall be determined and the APRM system gains shall be adjusted by the ratios given in Technical Specifications 2.1.A.1.a and 2.1.B.

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 equiv full pwr hr
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	A	Standard Pressure Source	Every 3 months
High Water Level in Scram Discharge Volume	A	Water Level	Refueling Outage
Low Reactor Water Level	B	Standard Pressure Source	Once/Operating Cycle
Turbine Stop Valve Closure	A	(6)	Refueling Outage
High Main Steamline Radiation	B	Appropriate Radiation Source (3)	Refueling Outage
First Stage Turbine Pressure Permissive	A	Pressure Source	Every 6 months and after refueling
Maine Steamline Isolation Valve Closure	A	(6)	Refueling Outage

Bases:

4.1 REACTOR PROTECTION SYSTEM

A. The scram sensor channels listed in Tables 4.1.1 and 4.1.2 are divided into three groups: A, B and C. Sensors that make up Group A are of the on-off type and will be tested and calibrated at the indicated intervals. Initially the tests are more frequent than Yankee experience indicates necessary. However, by testing more frequently, the confidence level with this instrumentation will increase and testing will provide data to justify extending the test intervals as experience is accrued.

Group B devices utilize an analog sensor followed by an amplifier and bi-stable trip circuit. This type of equipment incorporates control room mounted indicators and annunciator alarms. A failure in the sensor or amplifier may be detected by an alarm or by an operator who observes that one indicator does not track the others in similar channels. The bi-stable trip circuit failures are detected by the periodic testing.

Group C devices are active only during a given portion of the operating cycle. For example, the IRM is active during start-up and inactive during full-ppower operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

B. The ratio of MFLPD to FRP shall be checked once per day to determine if the APRM gains require adjustment. Because few control rod movements or power changes occur, checking these parameters daily is adequate.

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 controls rods except that:

   (a) If after withdrawal of at least twelve 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 pressure, are fully inserted, no more than two rods may be moved.

4. Control rod patterns and the sequence of withdrawal or insertion shall be established such that the rod drop accident limit of 280 cal/g is not exceeded.

4.3 SURVEILLANCE REQUIREMENTS

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.

   (c) Out-of-sequence control rods in each distinct RWM group shall be selected and the annunciatior of the selection errors verified.

   (d) An out-of-sequence control rod shall be withdrawn no more than three notches and the rod block function verified.

4. The control rod pattern and sequence of withdrawal or insertion shall be verified to comply with Specification 3.3.B.4.

3.3 LIMITING CONDITIONS FOR OPERATION

5. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate greater than or equal to three counts per second.

6. During operation with limiting control rod patterns either:

   (a) Both RBM channels shall be operable; or

   (b) Control rod withdrawal shall be blocked; or

   (c) The operating power level shall be limited so that the MCPR will remain above the fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of any single operable control rod.

4.3 SURVEILLANCE REQUIREMENTS

5. Prior to control rod withdrawal for startup or during refueling, verification shall be made that at least two source range channels have an observed count rate of at least three counts per second.

6. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

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 responsbility 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.

3.3 (cont'd)

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 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 section 5.5.1 of NEDE 24011P-A, latest revision, "Control Rod Drop Accident Evaluation".

TABLE 4.7.2.a

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time(Sec)	Normal Position	Action on Initiating Signal
1	Main Steam Line Isolation (2-80A, D & 2-86A, D)	4	4	5(note 2)	Open	GC
1	Main Steam Line Drain (2-74, 2-77)	1	1	35	Closed	SC
1	Recirculation Loop Sample Line (2-39, 2-40)	1	1	5	Closed	SC
2	RHR Discharge to Radwaste (10-57, 10-66)		2	25	Closed	SC
2	Drywell Floor Drain (20-82, 20-83)		2	20	Open	GC
2	Drywell Equipment Drain (20-94, 20-95)		2	20	Open	GC
3	Drywell Air Purge Inlet (16-19-9)		1	10	Closed	SC
3	Drywell Air Purge Inlet (16-19-8)		1	10	Open	GC
3	Drywell Purge & Vent Outlet (16-19-7A)		1	10	Closed	SC
3	Drywell Purge & Vent Outlet Bypass (16-19-6A)		1	10	Closed	SC
3	Drywell & Suppression Chamber Main Exhaust (16-19-7)		1	10	Closed	SC
3	Suppression Chamber Purge Supply (16-19-10)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet (16-19-7B)		1	10	Closed	SC
3	Suppression Chamber Purge & Vent Outlet Bypass (16-19-6B)		1	10	Open	GC
3	Exhaust to Standby Gas Treatment System (16-19-6)		1	10	Open	GC
3	Containment Purge Supply (16-19-23)		1	10	Open	GC
3	Containment Purge Makeup (16-20-20, 16-20-22A, 16-20-22b)		3	NA	Closed	SC
5	Reactor Cleanup System (12-15, 12-18)	1	1	25	Open	GC
5	Reactor Cleanup System (12-68)		1	45	Open	GC
6	HPCI (23-15, 23-16	1	1	55	Open	GC
6	RCIC (13-15, 13-16)	1	1	20	Open	GC
	Primary/Secondary Vacuum Relief (16-19-11A, 16-19-11B)		2	NA	Closed	SC
	Primary/Secondary Vacuum Relief (16-19-12A, 16-19-12B)		2	NA	Closed	Process
	Control Rod Hydraulic Return Check Valve (3-181)			NA	Open	Process
3	Containment Air Sampling (VG 23, VG 26, 109-76A&B)		4	5	Open	GC

Table 4.7.2.b

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES NOT SUBJECT TO TYPE C LEAKAGE TESTS

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time(sec)	Normal Position	Action on Initiating Signal
2	RHR Return to Suppression Pool (10-39A,B)		2	70	Closed	SC
2	RHR Return to Suppression Pool (10-34A,B)		2	120	Closed	SC
2	RHR Drywell Spray (10-26A, B & 10-31A,B)		4	70	Closed	SC
2	RHR Suppression Chamber Spray (10-38A,B)		2	45	Closed	SC
3	Containment Air Compressor Suction (72-38A,B)		2	20	Open	GC
4	RHR Shutdown Cooling Supply (10-18, 10-17)	1	1	28	Closed	SC
4	RHR Reactor Head Cooling (10-32, 10-33)	1	1	25	Closed	SC
	Feedwater Check Valves (2-28 A,B)	2	2	NA	Open	Proc.
	Reactor Head Cooling Check Valve (10-29)	1		NA	Closed	Proc.
	Standby Liquid Control Check Valves (11-16, 11-17)	1	1	NA	Closed	Proc.
*	Hydrogen Monitoring (109-75 A, 1-4; 109-75 B-D, 1-2) Sampling Valves - Inlet		10	NA	NA	NA
*	Hydrogen Monitoring (VG-24, 25, 33, 34)		4	NA	NA	NA

*These valves are remote manual sampling valves which do not receive an isolation signal. Only one valve in each line is required to be operable.

In the event that both startup transformers are lost, adequate power is available to operate the emergency safeguards equipment from either of the emergency diesel generators or from either of the delayed-access offsite power sources. Also, in the event that both emergency diesel generators are lost, adequate power is available immediately to operate the emergency safeguards equipment from at least one of the startup transformers or from either of the delayed-access offsite power sources within six hours. The plant is designed to accept one hundred percent load rejection without adverse effects to the plant or the transmission system. Network stability analysis studies indicate that the loss of the Vermont Yankee unit will not cause instability and consequent tripping of the connecting 345 kv and 115 kv lines. The Vernon feed is an independent source. Thus, the availability of the delayed-access offsite power sources is assured in the event of a turbine trip. Therefore, reactor operation is permitted with the startup transformers out of service and with one diesel generator out of service provided the NRC is notified immediately of the event and restoration plans.

Either of the two station batteries has enough capacity to energize the vital buses and supply d-c power to the other emergency equipment for 8 hours without being recharged. In addition, two 24 volt ECCS Instrumentation batteries supply power to instruments that provide automatic initiation of the ECCS and some reactor pressure and level indication in the control room.

Due to the high reliability of battery systems, one of the two batteries may be out of service for up to three days. This minimizes the probability of unwarranted shutdown by providing adequate time for reasonable repairs. A station battery, ECCS Instrumentation battery, or an Uninterruptible Power System battery is considered inoperable if more than one cell is out of service. A cell will be considered out of service if its float voltage is below 2.13 volts and the specific gravity is below 1.190 at 77°F.

The battery room is ventilated to prevent accumulation of hydrogen gas. With a complete loss of the ventilation system, the accumulation of hydrogen would not exceed 4 percent concentration in 16 days. Therefore, on loss of battery room ventilation, the use of portable ventilation equipment and daily sampling provide assurance that potentially hazardous quantities of hydrogen gas will not accumulate.

C. The minimum diesel fuel supply of 25,000 gallons will supply one diesel generator for a minimum of seven days of operation satisfying the load requirements for the operation of the safeguards equipment. Additional fuel can be obtained and delivered to the site from nearby sources within the seven-day period.

4.10 AUXILIARY ELECTRICAL POWER SYSTEMS

Bases:

A. The monthly tests of the diesel generators are conducted to check for equipment failures and deterioration. The test of the undervoltage automatic starting circuits will prove that each diesel will receive a start signal if a loss of voltage should occur on its emergency bus. The loading of each diesel generator is conducted to demonstrate proper operation at less than the continuous rating and at equalibrium operating conditions. Generator experience at other generator stations indicates that the testing frequency is adequate to assure a high reliability of operation should the system be required.

Table 3.11-2

MCPR OPERATING LIMITS

Exposure Range	Value of "N" in RBM Equation (1)	Fuel Type 8 x 8	8 x 8R	P8 x 8R
BOC to EOC-2 GWd/t	42%	1.21	1.26	1.27
	41%	1.21	1.22	1.23
	40%	1.21	1.21	1.22
	39%	1.21	1.21	1.21
EOC-2 GWd/t to EOC-1 GWD/t	42%	1.26	1.26	1.28
	41%	1.26	1.26	1.28
	≤40%	1.26	1.26	1.28
EOC-1 GWd/t to EOC	42%	1.29	1.29	1.31
	41%	1.29	1.29	1.31
	≤40%	1.29	1.29	1.31

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 Introduction

Vermont Yankee Nuclear Power Corporation (VYNPC or licensee) has proposed changes to the Technical Specifications of the Vermont Yankee Nuclear Power Station (VY) in Reference 1 and as supplemented by Reference 2. The proposed changes relate to the core for Cycle 8 operation at power levels up to 1593 MWt (100% power). In support of the reload application, the licensee has enclosed proposed Technical Specification changes in Reference 1 and the GE BWR supplemental licensing submittal (Reference 3).

This reload involves loading of prepressurized GE 8x8 retrofit (P8x8R) fuel. This is the same type of fuel as was loaded during the last reload. The description of the nuclear and mechanical designs of 8x8 retrofit is contained in References 4 and 5. Reference 4 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing a mixture of 8x8 and 8x8R fuel. The use and safety implications of prepressurized fuel are presented in Appendix D to Reference 3 and have been found acceptable per Reference 5. The conclusions of Reference 6 found that the methods of Reference 4 were generally applicable to prepressurized fuel. Therefore, unless otherwise specified, Reference 4, as supported by Reference 6, is adequate justification for the current application of prepressurized fuel. This report also addresses proposed Technical Specification changes submitted by VYNPC at the request of the NRC staff in Reference 7.

2.0 Evaluation

2.1 Reactor Physics

The reload application follows the procedure described in NEDE-24011-P, "Generic Reload Fuel Application." We have reviewed this application and the consequent Technical Specification changes. The transient analysis input parameters are typical for BWRs and are acceptable. Core wide transient analysis results are given for the limiting transients and the required operating limit values for MCPR are given for each fuel type. The revised MCPR limits are the only changes in Technical Specifications required by the reload and they are acceptable.

Other changes in the Technical Specifications include a change in the allowable delay time in the protection system from 100 milliseconds to 50 milliseconds to correspond to the value used in the licensing analyses, a change in the maximum total peaking factor Technical Specification to reflect the presence of fuels of different designs (number of rods per bundle and fuel length) in the core, and a change in the control rod pattern and sequence requirement to reflect the revised analysis procedure of NEDE-24011-P-A.

We have reviewed these changes and conclude that they are acceptable.

2.2 Thermal Hydraulics

As stated in Reference 4, for BWR cores which reload with GE's retrofit 8x8R fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient or fuel misloading, the most limiting events have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for the exposed 8x8 fuel and the exposed and fresh 8x8R fuel. Addition of the largest reductions in critical power ratio to the SLMCPR was used to establish the operating limits for each fuel type.

We have found the methods used for this analysis consistent with previously approved past practice (Reference 4). We have found the results of this analysis and the corresponding Technical Specification changes acceptable.

2.3 Administrative Changes

The change in the definition of "operable" proposed by the licensee is in response to a request from the NRC staff (Reference 7). This change improves the enforceability of Technical Specifications, and is acceptable . Also typographical corrections have been made on pages 136 and 179.

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: December 18, 1980

REFERENCES

1. Letter, L.H. Heider (VYNPC) to Office of Nuclear Reactor Regulation (USNRC), dated August 19, 1980.

2. Letter, L.H. Heider (VYNPC) to Office of Nuclear Reactor Regulation (USNRC), dated October 7, 1980

3. "Supplemental Reload Licensing Submittal for Vermont Yankee Nuclear Power Station Reload 7" Y1003J01A02, July 1980.

4. "General Electric Boiling Water Reactor Generic Reload Application," NEDE-24011-P-A, May 1977.

5. Letter, R.E. Engel (GE) to U.S. Nuclear Regulatory Commission, dated January 30, 1979.

6. Letter, T.A. Ippolito (USNRC) to R. Gridley (GE), April 16, 1979, and enclosed SER.

7. Letter, D. Eisenhut (USNRC) to L.H. Heider (VYNPC), April 10, 1980.

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. 61 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 associated with Cycle 8 operation, a change in the factor for adjusting fuel power density calculations, a surveillance requirement for faster response from the reactor protective system, and certain administrative changes.

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 the 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 August 19, 1980, as supplemented October 7 and 23, and November 21, 1980, (2) Amendment No. 61 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, NW, Washington, D.C. and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont. 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 18th day of December 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

TAIppolito

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