OCTOBER 1 0 1978

Docket No. 50-271

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Mr. Robert H. Groce
Licensing Engineer
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Dear Mr. Groce:

The Commission has issued the enclosed Amendment No. 47 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This Amendment changes the Technical Specifications to incorporate the limiting conditions for operation associated with cycle 6 operation. These changes are in response to your submittals dated June 21, 1978, July 12, 1978, August 30, 1978, and September 20, 1978. To meet our requirements, certain changes to the Technical Specifications which you proposed were necessary. These changes have been discussed with and concurred in by your staff.

Because the analysis on which your proposed Technical Specifications is based did not address operation beyond EOC-2 GWD/T, we regard operation beyond EOC-2 GWD/T as an unreviewed safety question. In order to permit us to review analysis and approve operation beyond EOC-2 GWD/T, please submit supporting analysis at least 90 days prior to EOC-2 GWD/T. Should the analysis be based on unapproved codes or input parameters, additional lead time should be provided.

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

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 47 to DPR-28
2. Safety Evaluation
3. Notice

cc w/enclosures: See page 2

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NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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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. 47
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 June 21, 1978, as supplemented July 12, August 30, and September 20, 1978, 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. 47, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 10, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 47

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A as follows:

Remove	Replace
iv	iv
5	5
5-a	5-a
6	6
9	9
10	10
11	11
12-a through 12-e (deleted)	--
13	13
14	14
14-a	14-a
14-b	14-b
18	18
31	31
47	47
48	48
180-a through 180-l	180-a through 180-l
--	Add 180-n5
180-01	180-01
180-r (deleted)	---

TABLE OF CONTENTS (CONT)

1.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variable associated with fuel thermal behavior.

Objective:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Bundle Safety Limit (Reactor Pressure >800 psia and Core Flow >10% of Rated)

When the reactor pressure is >800 psia and core flow is >10% of rated, the existence of a Minimum Critical Power Ratio (MCPR) less than 1.07 shall constitute violation of the fuel cladding integrity safety limit.

2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the nuclear system safety limits from being exceeded.

Objective:

To define the level of the process variable at which automatic protective action is initiated.

Specification:

A. Trip Settings

The limiting safety system trip settings shall be as specified below:

1. Neutron Flux Trip Settings

   a. APRM Flux Scram Trip Setting (Run Mode)

   When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be as shown on Figure 2.1.1 and shall be:

   $$S \leq 0.66W + 54\%$$

   where:

   S = 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.

1.1 SAFETY LIMIT

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

When the reactor pressure is $\leq$800 psia or core flow $\leq$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 a maximum total peaking factor (MTPF) greater than the design value of A, the APRM gain shall be increased by the ratio:

$$\frac{MTPF}{A}$$

where:

A = 2.62 for 7x7 fuel
= 2.44 for 8x8 fuel

MTPF = The value of the existing maximum total peaking factor.

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:

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

1.1 SAFETY LIMIT

C. Power Transient

To ensure that the Safety Limit established in Specification 1.1A 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 x 10^6 lbs/hr core flow.

In the event of operation with a maximum total peaking factor (MTPF) greater than the design value of A, the APRM gain shall be increased by the ratio

$$\frac{MTPF}{A}$$

where:

A = 2.62 for 7x7 fuel
= 2.44 for 8x8 fuel

MTPF = The value of the existing maximum total peaking factor.

Bases:

1.1 Fuel Cladding Integrity

Refer to Section 5.1 of General Electric Licensing Topical Report, "Generic Reload Fuel Application", NEDE-24011P, Amendment 3, March 1978.

This page has been deleted.

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

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.1A or 1.1.1B will not be exceeded. Scram times are checked periodically

This page has been deleted.

2.1 FUEL CLADDING INTEGRITY

A. Trip Settings

The bases for individual trip settings are discussed in the following paragraphs.

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1593 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120 percent scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is substantial margin from fuel damage. Therefore, the use of flow referenced scram trip provides even additional margin.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity Safety Limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity Safety Limit yet allows operating margin that reduces the possibility of unnecessary scrams.

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 MTPF 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 minimizer. 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 for 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 involves 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 control 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 distribution 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 maximum total peaking factor exceeds the specified values. 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 over power 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 operability 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 100 milliseconds.

B. During operation with a maximum total peaking factor (MTPF) greater than the design value (A) 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 maximum total peaking factor (MTPF) to or less than the design value (A).

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 peak heat flux and total peaking factor 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.

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-power operation. Testing of these instruments is only meaningful within a reasonable period prior to their use.

B. The peak heat flux and total peaking factor shall be checked once per day to determine if the APRM gains require adjustment. This will normally be done by checking LPRM readings. Because few control rod movements or power changes occur, checking these parameters daily is adequate.

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

Minimum Number of Operable Instrument Channels per Trip System (Note 1)	Trip Function	Modes in Which Function Must be Operable: Refuel	Startup	Run	Trip Setting
	Start up Range Monitor				
2	a. Upscale (Note 2)	X	X		$\leq 5 \times 10^5$ cps (Note 3)
2	b. Detector not fully inserted	X	X		
	Intermediate Range Monitor				
2	a. Upscale	X	X		$\leq$108/125 full scale
2	b. Downscale (Note 4)	X	X		$\geq$5/125 full scale
2	c. Detector not fully inserted.	X	X		
	Average Power Range Monitor				
2	a. Upscale (Flow Bias)			X	$\leq$0.66W + 42% (Note 5)
2	b. Downscale			X	$\geq$2/125 full scale
	Rod Block Monitor (Note 6)				
1	a. Upscale (Flow Bias) (Note 7)			X	$\leq$0.66W + N (Note 5)
1	b. Downscale (Note 7)			X	$\geq$2/125 full scale
1	Trip System Logic	X	X	X	
1	Scram Discharge Volume	X	X	X	$\leq$12 gallons

TABLE 3.2.5 NOTES

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

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

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

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

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

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

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

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to these parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

During steady state power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting values shown in Tables 3.11-1A through G. If at any time during steady state operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within prescribed limits within two (2) hours, the reactor shall be brought to the shutdown conditions within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective:

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications:

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at >25% rated thermal power.

LIMITING CONDITIONS FOR OPERATION

B. Linear Heat Generation Rate (LHGR)

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR of 18.5 kw/ft for 7x7 and 13.4 kw/ft for 8x8 and 8x8R.

If at any time during steady state operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power.

C. Minimum Critical Power Ratio

MCPR shall be determined daily during reactor power operation at $\geq$25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.6.

3.11 LIMITING CONDITIONS FOR OPERATION

SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

1. During steady state power operation the MCPR Operating Limit shall be equal or greater than the values shown on Table 3.11-2. For core flows other than rated MCPR the Operating MCPR Limit shall be the above value multiplied by K_f where K_f is given by Figure 3.11-2. If at any time during steady-state operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor power shall be brought to shutdown condition, within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

2. If during steady state operation, the off-gas activity as measured at the SJAE's exceeds 236,000 μCi/sec for fifteen (15) minutes or 1.18 Ci/sec for one (1) minute immediate operator action shall be taken to change the operating MCPR to $\geq$1.31 times the appropriate K_f. Subsequent operation shall be per paragraph 3.11.C.1 above with all MCPR values on Table 3.11-2 equal to 1.31.

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.

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

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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	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:

3.11 B. Linear Heat Generation Rate (LHGR)

Refer to Section 2.4 of NEDE-24011P, Amendment 3, dated March 1978.

Bases:

3.11C Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

1. Refer to Section 5.2 of NEDE-24011P, Amendment 3, dated March 1978.

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 to the event air ejector off-gas radiation exceeds levels that could be associated with a mis-load fuel assembly.

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3.11 FUEL RODS (Continued)

D. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values of MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specification 3.11.A, B, and C that if at any time during steady state power operation, it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded, action is then initiated within fifteen minutes to restore operation to within the prescribed limits. Each event involving steady state operation beyond a specified limit shall be reported as a reportable occurrance. However, if the corrective action is taken as described, a thirty day written report will meet the requirement of this specification.

Table 3.11-1G

MAPLHGR, PCT, Oxidation Fraction Versus Exposure, Fuel Type 8DPB289

Average Planar Exposure (MWd/t)	MAPLHGR (kw/ft)	P.C.T. (Deg-F)	Oxidation Fraction
200.0	11.2	2126	0.027
1000.0	11.2	2119	0.026
5000.0	11.8	2178	0.030
10000.0	12.0	2185	0.030
15000.0	12.1	2200	0.032
20000.0	11.8	2187	0.031
25000.0	11.3	2120	0.025
30000.0	11.1	2095	0.023

Table 3.11-2

MCPR OPERATING LIMITS

Value of "N" in RBM Equation(1)	MCPR Operating Limits Over Exposure Range Noted		
	BOC to EOC - 2 GWD/T		
	7x7	8x8	8x8R
40%	1.23	1.22	1.24
39%	1.23	1.22	1.22

Notes:

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 47 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

Introduction

By letter dated June 21, 1978, Vermont Yankee Nuclear Power Corporation (the licensee) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station (VYNPS). The proposed changes relate to the replacement of 96 fuel assemblies constituting refueling of the core for sixth cycle operation at power levels up to 1593 Mwt out to end of cycle conditions minus 2 GWD/T.

In support of the reload application, the licensee has provided the Reload 5 licensing submittal and the proposed Technical Specification changes (Reference 1), information on the VYNP Loss of Coolant Accident (LOCA) analysis (References 1 and 3), and responses to NRC requests for additional information (Reference 4).

This reload involves loading of General Electric (GE) 8x8 fuel and GE Retrofit 8x8R fuel. The description of the nuclear and mechanical design of the 8x8R fuel and the 8x8 fuel is contained in GE's licensing topical report for BWR reloads (Reference 5). Reference 5 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 7x7, 8x8 and 8x8R fuel.

Values for plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other various design parameters are provided in Reference 5. Additional plant and cycle dependent information are provided in the reload application, (Reference 1), which closely follows the outline of Appendix A of Reference 5.

Reference 7, describes the staff's review, approval, and conditions of approval for the plant-specific data addressed in Reference 5. The above mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

Our safety evaluation (Reference 7) of the GE generic reload licensing topical report concluded that the nuclear and mechanical design of the 8x8R fuel, and GE's analytical methods for nuclear, thermal-hydraulic, and transient and accident calculations as applied to mixed cores containing 7x7, 8x8 and 8x8R fuel are acceptable. Approval of the nuclear and mechanical design of 8x8 fuel was determined based on information in Reference 6 and expressed in the staff's status report (Reference 8) on that document.

Because of our review of a large number of generic considerations related to use of 8x8R fuel in mixed loadings with 8x8 and 7x7 fuel, and on the basis of the evaluations which have been presented in Reference 7, only a limited number of additional areas of review have been included in this safety evaluation. For evaluations of areas not specifically addressed in this safety evaluation, the reader is referred to Reference 7.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 6 operation of VYPNS, 36 fresh 8x8 fuel bundles of type 8D274H and 60 fresh 8x8R bundles of type 8DPB289 will be loaded into the core (Reference 1). The remainder of the 368 fuel bundles in the core will be 7x7 and 8x8 fuel exposed during the previous cycles. The fresh fuel will be loaded in a core pattern as shown in Figure 3.2 of Reference 1, which is acceptable.

Based on the data presented in section 5 of Reference 1, both the control rod systems and the standby liquid control system will have acceptable shutdown capability during Cycle 6.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit

As stated in Reference 5, the minimum critical power ratio (MCPR) which may be allowed to result from core-wide or localized transients is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling.

The safety limit MCPR for VYNPS is being raised from 1.06 to 1.07 because the distribution of fuel rod power within the 8x8R fuel bundles is flatter than that of the 8x8 fuel. The reason for the flatter power distribution is the presence of two rather than one water rods in 8x8R fuel. The issue has been addressed in Reference 7 and the 1.07 limit has been found acceptable to BWRs with uncertainties in flux monitoring and operational parameters no greater than those listed in Table 5-1 of Reference 5, for which the CPR distribution is within the bounds of Figures 5.2 and 5.2a of Reference 5. It has been proposed in Table 6.1 of Reference 1 that these conditions are applicable for VYNPS up to EOC-2 GWD/T. The applicability will be verified in the physics startup tests discussed in Section 3.0.

2.2.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the CPR below the intended operating limit MCPR during Cycle 6 operation of VYNPS. The most limiting of these operational transients up to EOC-2 GWD/T have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the critical power ratio (ΔCPR) during the earlier part of Cycle 6.

The transients evaluated were the feedwater controller failure at maximum demand, loss of a 100°F feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Tables 6.3, 7.2 and 7.4 of Reference 1 were assumed. The most limiting transients for the later part of Cycle 6 (EOC-2 GWD/T to EOC) will be completed in the next several months. The results of the EOC analysis will be then used to establish the operating limit MCPRs for the later part of Cycle 6.

The calculated systems responses and ΔCPRs for the above listed operational transients and conditions have been analyzed by the licensee. Table 1 lists the ΔCPRs for the various fuel types at the specified cycle exposure. Also included in Table 1 are the results of the maximum vessel pressure discussed in Section 2.4.

TABLE 1

Transient	Limiting Exposure Time	ΔCPR 7x7/8x8/8x8R	Operating Limit MCPR 7x7/8x8/8x8R
Load Rejection without Bypass	EOC-2 GWD/T to EOC	+++	+++
Turbine Trip without Bypass	EOC-2 GWD/T to EOC	+++	+++
Loss of 100°F Feedwater Heater	BOC-EOC	.16/.15/.15	1.23/NA/NA
Feedwater Controller Failure	BOC-EOC	.05/.07/.07	NA

+++ Operating limit MCPRs for these transients which are limiting during the later part of Cycle 6 shall be determined prior to EOC-2 GWD/T.

values

Condition			
Rod Withdrawal	BOC-EOC	NA/.15/.17	NA/1.22/1.24
Overpressurization (MSIV Closure)	Peak vessel pressure assuming one failed SRV is 1307 psi		
Fuel Loading Error	BOC-EOC	NA/.24/.24	**

2.3 Fuel Loading Error

The potential fuel loading errors (FLE) involving misoriented bundles and mislocated bundles have been evaluated. The analysis of the fuel loading error is discussed in Reference 5 and approved in Reference 7.

The limiting fuel loading error (a mislocated bundle) ΔCPR was calculated by the more conservative older GE analysis. For the VY plant, FLE analyses for an earlier cycle showed conservatisms of approximately 50% in ΔCPR when the old approved analysis is compared to the newly approved GE analysis (Reference 9). Even though we recognize the large conservatisms in the calculated valves of the older analysis methods cannot be approved since it was not for this specific cycle. In the interim, until a new analysis is provided for Cycle 6, the ΔCPR value for the fuel loading error is accepted as 0.24 which gives credit for 25% of the previously identified conservatisms.

This ΔCPR value, when added to the safety limit MCPR of 1.07, would result in an operating limit MCPR of 1.31 for the 8x8 and 8x8R fuel. If a FLE occurs during this cycle some of the fuel rods in the bundle could experience boiling transition and fail.

In this event, one means for detection of abnormal fuel degradation at VYNPS will be accomplished by measurements of off-gas radioactivity levels at the steam jet air ejector. To assure that further fuel degradation as a result of a fuel loading error will not occur, VY has proposed the installation of a new alarm on SJAE activity. This alarm will serve to warn the operator that there could be a FLE event in the core. If an activity level of 0.236 Ci/sec persists for 15 minutes or 1.18 Ci/sec persists for one minute Technical Specifications have been added which require that the operator take action to increase the operating limit MCPRs to ≥1.31. This action assures that the worst mislocated bundle would remain above the safety limit MCPR. The activity levels are chosen to correspond to the maximum activity release expected from a single mislocated bundle. Continued operation of the plant would then be determined by the most limiting condition relative to the MCPR value of 1.31 or the Technical Specification limit of offgas listed in Technical Specification Table 3.2.4.

**See Section 2.3

In addition to the detection capabilities and Technical Specification requirements, the core verification procedures have been augmented by an independent verification by Yankee Atomic Electric Company personnel and the NRC Office of Inspection and Enforcement.

In summary, we find the above procedures which require operating limit MCPRs as shown in Table 1 be raised to 1.31, in the event of indicated fuel degradation, in addition to the Technical Specification limits on offgas acceptable for this cycle of operation.

2.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 7. As specified in Reference 7, the sensitivity of peak vessel pressure to failure of one SRV has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure (1375 psi) to allow for the failure of at least one valve. Therefore the limiting overpressure events as analyzed by the licensee is acceptable.

2.5 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 1) show that the channel hydrodynamic and reactor core decay ratios at the Natural Circulation - 105% Rod Line intersection (which is the least stable physically attainable point of operation) are below the stability limit.

Because operation in the natural circulation mode is prohibited by Technical Specifications, there will be added margin to the stability limit. We find this is acceptable.

2.6 Accident Analysis

2.6.1 ECCS Appendix K Analysis

Input data and results for the ECCS analysis have been given in References 1 and 3. The information presented fulfills the requirements for such analyses outlined in Reference 7.

We have reviewed the analyses and information submitted for the reload and conclude that the VYNPS will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: (1) it is operated with the "MAPLHGR VERSES AVERAGE PLANAR EXPOSURE" values given in Tables 3.11-1A through 3.11-1G of the Technical Specifications, (2) is it operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss-of-Coolant-Accident, as described in Section 2.2.) and 2.3)

2.6.2 Control Rod Drop Accident

The worst case control rod drop accident (CRDA) can occur under startup conditions when the characteristic parameters for the accident meet the requirements for bounding analyses described in Reference 5, this is adequate to show that the design basis of 280 cal/gm peak fuel enthalpy for a startup CRDA is met (Reference 7).

For VYNPS, the characteristic accident parameters for the worst startup CRDA satisfy the requirements for bounding analyses as described in Reference 5. Therefore the postulated CRDA would be $\leq$280 cal/gm which is acceptable.

3.0 Physics Startup Testing

The licensee in accordance with Technical Specification requirements and Reference 2 will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed in the transient and accident analysis calculations will be met during Cycle 6. The tests will also check that the core is loaded as intended and that the incore monitoring system and control rod worths and operations are functioning as expected. A written report of the startup tests will be provided to NRC within approximately 45 days as discussed in Reference 2.

4.0 Technical Specifications

The changes to the Technical Specification as proposed by the licensee are acceptable with the following exceptions:

1. The operating limit MCPR for the 7x7 fuel shall be changed to 1.23. Other changes in MCPR have been required upon evidence of high gas activity so that the core wide safety limit will not be violated for the worst case fuel loading error. This is discussed in Section 2.3.

2. The proposed wording in the Technical Specifications relating to action if limiting values of ALHGR, LHGR and MCPR are exceeded was not included in this amendment. As discussed with the licensee, these matters will be considered separately for a possible later license amendment, after the licensee has provided further supporting arguments.

5.0 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 §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.

6.0 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: October 10, 1978

References

1. Letter, D. E. Vanderburgh, to NRC WVY 78-59, dated June 21, 1978.
2. Letter, R. H. Groce, to NRC, WVY 78-64, dated July 12, 1978.
3. Letter, W. P. Johnston, to NRC, WVY 77-71, dated August 12, 1977.
4. Letter, D. E. Vanderburgh, to NRC, WVY 78-89, dated September 20, 1978.
5. General Electric Boiling Water Reactor Generic Reload Fuel Application, NEDE-24011-P, May 1977.
6. General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel, NEDO-20360, Rev. 1, Supplement 4, April 1, 1976.
7. NRC Safety Evaluation of the GE Generic Reload Fuel Application (NEDE-20411-P), April 1978.
8. Status Report on the Licensing Topical Report "General Electric Boiling Water Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, April 1975.
9. Letter, Ronald Engel, GE to Darrell Eisenhut, NRC, Fuel Assembly Loading Error, June 1, 1977.

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. 47 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.

This amendment changes the Technical Specifications to permit operation of the facility in the sixth fuel cycle, following the first refueling, during which 96 of the 368 fuel assemblies were replaced.

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

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

For further details with respect to this action, see (1) the application for amendment dated June 21, 1978, as supplemented July 12, August 30, and September 20, 1978, (2) Amendment No. 47 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.

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 Operating Reactors

Dated at Bethesda, Maryland this 10th day of October, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors