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MAR 11 1981

Docket No. 50-271

RECEIVED MAR 17 1981 U.S. NUCLEAR REGULATORY COMMISSION

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

Dear Mr. Smith:

The Commission has issued the enclosed Amendment No. 64 to Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This Amendment changes the Technical Specifications to permit the performance of stability and recirculation pump trip tests and are in response to your submittal dated February 12, 1981.

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 #2
Division of Licensing

Enclosures:
1. Amendment No. 64 to DPR-28
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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No legal objection to amendment + FR Notice

OFFICE	ORB #2	ORB #2	AD:OR	OELD	ORB #2		
SURNAME	SNorris:mjf	VRooney	TNovak	BACHMANN	TIppolito		
DATE	3/9/81	3/9/81	3/9/81	3/9/81	3/9/81		

NRC FORM 318 (10/80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1980—329-824

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

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 64
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 February 12, 1981, 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. 64 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 11, 1981.

cc:
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ATTACHMENT TO LICENSE AMENDMENT NO. 64

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-331

1. Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
5	5
5a	5a
-	5b
6	6
-	6a
19	19
21a	21a
47	47
48	48
110b	110b
-	110c
180a	180a
180b	180b
-	180b1
180-01	180-01

1.1 SAFETY LIMIT

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 LIMITING SAFETY SYSTEM SETTING

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 (except as permitted in 2.1.A.1.b for special stability testing) as shown on Figure 2.1.1 and shall be:

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

1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

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.

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 13.4 kW/ft for 8x8 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. <u>APRM Flux Scram Trip Setting (Run Mode, Special Testing)</u>

2.1 LIMITING SAFETY SYSTEM SETTING

1.1 SAFETY LIMIT

For the purpose of performing special stability testing when the Mode Switch is in in the RUN position, the APRM flux scram trip setting shall be:

$$S \leq 0.66W + 85\%$$

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.

During operation under the provisions of 2.1.A.1.b, the special MAPLHGR limits of 3.11.A shall apply and such operation shall be limited to the duration of pump trip and stability tests. Adjustments for the ratio of MFLPD to FRP greater than 1.0 are not required while conducting special testing under this provision.

c. 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. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia or Core Flow $\leq$ 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.

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 means other than the expected scram signal.

B. APRM Rod Block Trip Setting

1. The APRM rod block trip setting shall as shown in Figure 2.1.1 and shall be (except as permitted in 2.1.B.2 for special stability testing):

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

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 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 13.4 kW/ft for 8x8 fuel.

FRP = fraction of rated power (1593 MWt)

.1 SAFETY LIMIT

2.1 LIMITING SAFETY SYSTEM SETTING

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.

2. For the purpose of performing special stability tests, the APRM rod block trip setting shall be:

$$S_{RB} \leq 0.66W + 75\%$$

During operation under the provision of 2.1.B.2, the special MAPLHGR limits of 3.11.A shall apply and such operation shall be limited to the duration of pump trip and stability tests. Adjustments for the ratio of MFLPD to FRP greater than 1.0 are not required while conducting special tests under this provision.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Trip Settings	Modes in Which Functions Must be Operating			Minimum Number Operating Instrument Channels Per Trip System (2)	Required Conditions When Minimum Conditions For Operation Are Not Satisfied (3)
		Refuel(1)	Startup	Run		
1. Mode Switch in Shutdown		X	X	X	1	A
2. Manual Scram		X	X	X	1	A
3. IRM						
High Flux	$\leq$120/125	X	X	X(11)	2	A
Inop		X	X	X(11)	2	A
4. APRM						
High Flux (flow bias)	$\leq$0.66W+54%(4)(12)			X	2	A or B
High Flux (reduced)	$\leq$15%	X	X		2	A
Inop				X	2(5)	A or B
Downscale	$\geq$2/125			X	2	A or B
5. High Reactor Pressure	$\leq$1055 psig	X	X	X	2	A
6. High Drywell Pressure	$\leq$2.5 psig	X	X	X	2	A
7. Reactor Low Water Level	$\geq$1.0 inch(6)	X	X	X	2	A
8. Scram Discharge Volume High Level	$\leq$24 gallons	X	X	X	2	A

9. Channel signals for the turbine control valve fast closure trip shall be derived from the same event or events which cause the control valve fast closure.

10. A turbine stop valve closure and generator load rejection bypass is permitted when the first stage turbine pressure is less than 30 percent of normal (220 psia).

11. The IRM scram is bypassed when the APRM's are on scale and the mode switch is in the run position.

12. For special stability tests, the APRM flux scram shall be $\leq$0.66W + 85% for the duration of testing. Adjustments for the ratio of MFLPD to FRP greater than 1.0 are not required while conducting special tests.

TABLE 3.2.5

CONTROL ROD BLOCK INSTRUMENTATION

nimum Number of erable Instrument annels per Trip System (Note 1)	Trip Function	Modes in Which Function Must be Operable Refuel	Startup	Run	Trip Setting
	Startup 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)(Note 8)
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.

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

3.6 LIMITING CONDITIONS FOR OPERATION

4.6 SURVEILLANCE REQUIREMENTS

2. All hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability every 31 days.

3. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.I.1, it shall be assumed that the facility had been on a 6-month inspection interval.

4. Once each refueling cycle, a representative sample of approximately 10% of the snubbers shall be functionally tested for operability including verification of proper piston movement, lockup and bleed. For each unit and subsequent unit found inoperable, an additional 10% shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capacity greater than 50,000 lbs need not be functionally tested.

J. Thermal Hydraulic Stability

1. When the reactor mode switch is in RUN, the reactor shall not intentionally be operated in a natural circulation

J. Thermal Hydraulic Stability

Operation in the natural circulation mode shall be timed and recorded for special tests. Also, during special tests loop

3.6 LIMITING CONDITIONS FOR OPERATION

mode, except as permitted in 3.6.J.2 below, nor shall an idle recirculation pump be started with the reactor in a natural circulation mode, except as permitted in 3.6.J.2.

2. For the purpose of performing special tests, operation in the natural circulation mode is permitted. For the purpose of recovering forced circulation operation during and after special tests at natural circulation, startup of an idle recirculation pump is permitted if:

   a. The ΔT between the idle loop and vessel saturation temperature is $\leq 50^{\circ}F$.

   b. The ΔT between the idle loop and an operating loop is $\leq 50^{\circ}F$.

   c. The ΔT between the vessel top head and the vessel bottom head is $\leq 145^{\circ}F$.

4.6 SURVEILLANCE REQUIREMENTS

temperatures, vessel saturation temperature (pressure), vessel top head temperature, and vessel bottom head temperature shall be monitored and recorded.

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 (except as specified in Section 3.11.A.1 for Special Stability Testing). 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

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

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.

1. The flow-biased APRM Rod Block normally provides the LOCA margin required for operation at reduced flow. This assurance will be diminished when the APRM Rod Block trip is increased, therefore special restrictions on MAPLHGR's are required for operation in that mode. After adjustment of the APRM Rod Block setting, the MAPLHGR's shall be 80% of the limits specified in Tables 3.11-1A through G.

B. <u>Linear Heat Generation Rate (LHGR)</u>

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 13.4 kW/ft for 8x8, 8x8R, and P8X8R.

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

SURVEILLANCE REQUIREMENTS

B. <u>Linear Heat Generation Rate (LHGR)</u>

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

C. <u>Minimum Critical Power Ratio</u>

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
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.	control rod pattern as described in the bases for Specification 3.3.B.6.

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Table 3.11-2

MCPR Operating Limits

Exposure Range	Value of "N" in RBM Equation (1)	Fuel Type(2) 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
Special Testing at Natural Circulation (Note 3,4)	75%	1.30	1.31	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 7x7 fuel. On this basis further evaluation of MCPR operating limits is required before 7x7 fuel can be used in Reactor Power Operation.

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

I. Introduction

By letter dated February 12, 1981 (reference 1), Vermont Yankee Nuclear Power Corporation (the licensee) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating License No. DPR-28, (reference 2) for the Vermont Yankee Nuclear Power Station (VY). The proposed changes permit the licensee to conduct stability and recirculation pump trip tests at Vermont Yankee during Cycle 8. The objectives of the tests are to obtain data for licensing support of Vermont Yankee stability performance and for qualification of its stability and operational transient models.

The stability tests will be performed at one point under minimum pump speed and three points under natural circulation conditions. The approach will be to perform tests with vessel and core pressure perturbation introduced through the turbine control system at steps of 10 psi. The resulting neutron flux response of the core will be measured and used to determine a core transfer function. The recirculation pump trip test will be performed after the stability test at minimum pump speed is completed and before the natural circulation tests. The total actual time of the tests will be less than 48 hours within a time span of seven days.

The Technical Specification changes are required in order to allow the operation under natural circulation during the tests. The tests also require bypassing of any affected trip functions while the test instrumentation is being installed and removed. The APRM flow-biased rod block line and scram and rod block monitoring setting are to be raised, and special MCPR and MAPLHGR limits are to be used during the tests.

II. Evaluation

The test procedures are presented in Reference 3.

In order to ensure that the stability test will be within acceptable limits, two levels of criteria have been established as follows. Level 1 criteria, if exceeded, require that the test be suspended immediately until corrective action can be taken. The Level 1 criteria are that the APRM response to pressure perturbation must be within $\pm 20\%$ of the rated power, the decay ratio

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must not exceed 1.0, the offgas and reactor coolant solubles must remain within the site administrative limit, and if the decay ratio reaches 1.0, the APRM oscillation must not exceed ± 15% of the rated power. Level 2 criteria are the expected test results which, if exceeded, the responsible test engineer must evaluate the situation and determine if the test should be suspended. The criteria are that the APRM response is within ± 15% of the rated power and the decay ratio is less than 0.5.

The tests are to be conducted in orderly fashion so that one test result provides guidance for the next test. The minimum pump speed test point, VPT1, will be performed first to allow determination of the sensitivity between minimum pump speed and natural circulation on stability of the systems. Data from lower power test points, VPT2 and VPT3, at natural circulation condition will be extrapolated to establish a power level of VPT4, which is the least stable condition and will be limited to 70% of the rated power and a decay ratio of less than 1.0. Since VPT4 is above the design APRM flow-biased rod block line and very close to the APRM flow-biased scram, higher settings of the rod block and scram are necessary in order for the test to be conducted without an inadvertent scram during the test. Special LOCA and MCPR limits will be used during the test in lieu of the APRM flow-biased scram. The limits are established through the reanalysis of the Cycle 8 reload safety analysis(4) using the natural circulation condition at VPT4 with the General Electric (GE) standard reload analysis method(5). The analysis describes local rod withdrawal error transient and core-wide transients such as load rejection without bypass, loss of 100°F feedwater heating, inadvertent HPCI start and feedwater control failure. The results of the analysis are reported in Reference 3. The most severe ΔCPR is 0.19 for the inadvertent HPCI start transient. An additional 0.05 is added to the ΔCPR to account for lack of operating experience at the high power natural circulation condition. With the acceptance criteria for the design-limit MCPR of 1.06 for 8x8, 1.07 for 8x8R and P8x8R bundles, the overall operating-limit MCPR's will be 1.30, 1.31 and 1.31, respectively, for 8x8, 8x8R and P8x8R bundles. These MCPR limits will be used during the natural circulation tests to replace the flow factor K_f which is normally used as a multiplier of the operating-limit MCPR for the reduced flow. Since the M-G sets are off at natural circulation conditions and special test procedure will ensure that only one pump will be started at natural circulation conditions, protection against a flow increase transient is provided.

The flow-biased APRM rod block which normally provides the LOCA margin required for operation at reduced flow is replaced by a special MAPLHGR limit. The MAPLHGR limit will be 80% of the limits specified in the Technical Specification.

The recirculation pump trip is to be conducted between the stability test points VPT1 (at minimum pump speed) and VPT2 (lower power test point at natural circulation). Since the reload analysis for natural circulation conditions, which is the least stable on power-flow map, has shown a decay ratio of less than 1.0 (4), the pump trip test should result in no adverse safety effect.

As for the evaluation of the stability test results, VY is scheduled to record the parameters listed on table 1. However, the staff feels it is important to record additional data in order to provide complete and convincing test results. The following parameters should be recorded:

(1) water level in the reactor vessel;
(2) core inlet temperature;
(3) steam flow at the turbine admission valves used to generate the test perturbations, and
(4) steam pressure upstream of these same valves.

It is also desirable, for some recorded parameters, to read not only the dc values, but noise components. For purposes of providing stochastic comparison analyses, the noise components of the reactor vessel pressure signal, the core inlet temperature, core inlet flow rate, and as many APRM and LPRM outputs as possible should be recorded both during the test and for at least ten minutes prior to the beginning of each of the tests. For the neutron noise data, a minimum of one APRM signal, plus 4 LPRM signals all from the same channel, plus an additional LPRM signal from a nearby channel are required. Noise components from any or all of the other output parameters listed above would be valuable, but are not required for the stability evaluation. If the parameter recording is done digitally, a minimum sampling rate of 10 Hz is needed for core stability analysis.

In order to perform audit calculations, the raw test data, with appropriate signal identification and scaling factors will be desirable. The following data should also be provided.

A. Data specific to each test at steady state conditions prior to the beginning of the test:

1. Power: Thermal power generated in the core.

Map of the power generated in each fuel assembly. Map of the vertical power distribution in each characteristic bundle type according to the position of the neighboring control rods.

2. Flow: Mass flow rate entering the reactor core. Fraction of the core mass flow rate which goes to the bypass region. Mass flow rate in each of the recirculation loop drive flows. Feedwater mass flow rate. Steam flow rate.

3. Temperature: Feedwater temperature. Core inlet temperature.

4. Pressure: Core inlet pressure. Core outlet pressure. Steam separator exit pressure. Vessel outlet pressure. Recirculation pump inlet pressure. Downcomer pressure at the jet pump entrance. Jet pump throat pressure.

5. Water level in the reactor vessel.

6. Map of control rod position and degree of insertion.

B. Plant data common to all proposed tests.

(1) Core description: Fuel assembly dimensional and material data; core map including control rod and LPRM locations and support plate orifices; reference design data including fraction of the thermal power which is transferred to the coolant by convection and fraction of the thermal power which is deposited in the bypass coolant flow by radiation.

(2) Recirculation pump data: Characteristic equations relating pump head and pump torque to mass flow rate and pump speed. Also, the moment of inertia of the pump and pump efficiency.

(3) Friction coefficients: At the core entrance orifice; at the core exit; between the core exit and the steam separator exit; between the steam separator exit and the steam dryers exit; at the jet pump suction entrance; at the drive flow nozzles; at the recirculation loop piping; at the jet pump diffusers; at the lower plenum; at the upper dome plenum. Also, the friction factor multiplier in the core fuel bundles.

(4) Length of area flow ratios (L/A) for the steam separators as a function of steam quality, for the recirculation loops, the jet pumps and the downcomer region.

(5) Neutronic parameters: Sets of 2 group cross-sections (i.e. Σ_a, $\nu\Sigma_f$, D, Σ_R) as a function of local bundle void fraction, burnup, gadolinium content and control rod position for each type of fuel bundle.

Also the doppler reactivity coefficient as a function of burn-up, void fraction and average fuel temperature.

(6) Fuel gap conductance.

The staff has reviewed the procedure, Technical Specification changes and safety analysis report of the proposed stability and recirculation pump trip tests at VY. Based on the above evalution, the staff concludes that the thermal hydraulic acceptance criteria, where MCPR must be greater than 1.06 and decay ration must be less than 1.0, will not be violated during the tests. Therefore, the Technical Specification change for VY is acceptable.

III. 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.

IV. 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: March 11, 1981.

References:

1. Letter from J. H. Heider (VYNPC) to Office of Nuclear Reactor Regulation of U. S. Nuclear Regulatory Commission, February 12, 1981.

2. License No. DPR-28 (Docket No. 50-271).

3. General Electric Company, "Vermont Yankee Nuclear Power Station Proposed Stability and Recirculation Pump Trip Test," NEDO-24279 80 NED 283, January 1981.

4. General Electric Company, "Supplemental Reload Licensing Submittal for Vermont Yankee Nuclear Power Station Reload No. 7," Y1003J01A02, July 1980.

5. General Electric Company, "General Electric Reload Application," NEDE-24011-p-1.

TABLE 1

Scheduled Parameters to be Recorded During VY Stability Test

LPRM readings
APRM readings
Jet Pump Pressure Differentials
Jet Pump Loop Flows
Core Pressure Differential
Reactor Vessel Pressure
Reactor Vessel Pressure Differential
Core Exit Pressures
Reactor Feedpump Flows
Reactor Feedwater Temperature
Recirculation Loop Drive Flows
Recirculation Loop Temperatures
Total Core Flow (Minimum Filtering)
Total Steam Flow
EPR (Electrical Pressure Regulator)
Pressure Controller

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. 64 to Facility Operating License No. DPR-28 issued to Vermont Yankee Nuclear Power Corporation which revises the Technical Specifications for operation of the Vermont Yankee Nuclear Power Station located in Windham County, Vermont. The amendment is effective as of the date of its issuance.

This amendment changes the Technical Specifications to permit the performance of stability and recirculation pump trip tests.

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 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 February 12, 1981, (2) Amendment No. 64 to License No. DPR-28.

DPR-28, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont 05301. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 11th day of March, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

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