November 30, 1977

Docket No. 50-271

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Yankee Atomic Electric Company
ATTN: Mr. Robert H. Groce
Licensing Engineer
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Westboro, Massachusetts 01581

Gentlemen:

The Commission has issued the enclosed Amendment No. 41 to Facility Operating License No. DPR-28 for the Vermont Yankee Nuclear Power Station. This amendment consists of changes to the Technical Specifications in response to your application dated October 12, 1977.

This amendment incorporates into the Technical Specifications higher Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the facility. We have evaluated your request for new MAPLHGR limits in conjunction with your reanalysis of the Emergency Core Cooling System performance submitted with your letter of August 12, 1977, in response to the Commission's Order of March 11, 1977. We have found your reanalysis to be acceptable. Effective upon issuance of this amendment, the Commission's Order for Modification of License dated March 11, 1977, relative to Facility Operating License No. DPR-28, is terminated.

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

Sincerely,

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 41
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

subject to Note 11/28/77

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DATE	11/19/77	11/ /77	11/22/77	11/28/77	11/30/77	

NRC FORM 318 (9-76) NRCM 0240

Yankee Atomic Electric Company

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filing dtd.: 10/12/77
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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. 41
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 October 12, 1977, 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. 41 , 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

Robert W. Reid for

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 30, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
180-a	180-a
180-f	180-f
180-m - 180-n3	180-m - 180-n4

The changed areas on the revised pages are shown by marginal lines.

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLIES

Applicability:

The Limiting Conditions for Operation associated with the fuel rods apply to those 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 F. 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 cold 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 $\geq$25% rated thermal power.

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

*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.

Table 3.11-1A

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee Fuel Type: 7D230

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200.0	14.7	2199.	0.031
1,000.0	14.8	2197.	0.030
5,000.0	15.0	2199.	0.029
10,000.0	14.6	2193.	0.030
15,000.0	14.1	2198.	0.072
20,000.0	13.8	2199.	0.073
25,000.0	13.7	2198.	0.071
30,000.0	13.8	2198.	0.071

Source: NEDO-21697, August 1977

Table 3.11-1B

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee

Fuel Type: 8D219

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

Source: NEDO-21697, August 1977

Table 3.11-1C

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

PLANT: Vermont Yankee

Fuel Type: 8D274L

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200.0	11.2	2060.	0.019
1,000.0	11.3	2064.	0.019
5,000.0	11.9	2133.	0.024
10,000.0	12.1	2129.	0.023
15,000.0	12.2	2159.	0.025
20,000.0	12.1	2167.	0.026
25,000.0	11.6	2118.	0.023
30,000.0	10.9	2028.	0.017

Source: NEDO-21697, August 1977

Table 3.11-1D

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee   Fuel Type: 8D274H

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200.0	11.1	2052.	0.019
1,000.0	11.2	2050.	0.018
5,000.0	11.8	2113.	0.022
10,000.0	12.1	2131.	0.023
15,000.0	12.2	2161.	0.025
20,000.0	12.0	2164.	0.026
25,000.0	11.5	2112.	0.022
30,000.0	10.9	2029.	0.017

Source: NEDO-21697, August 1977

Table 3.11-1E

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee

Fuel Type: 8D274 (High Gd)

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200.0	11.1	2053.	0.019
1,000.0	11.1	2044.	0.018
5,000.0	11.6	2092.	0.021
10,000.0	12.1	2141.	0.024
15,000.0	12.2	2165.	0.026
20,000.0	12.1	2170.	0.027
25,000.0	11.6	2119.	0.023
30,000.0	10.6	1993.	0.015

Source: NEDO-21697, August 1977

Table 3.11-1F

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE

Plant: Vermont Yankee

Fuel Type: LTA

Average Planar Exposure (MWd/t)	MAPLHGR (kW/ft)	PCT (°F)	Oxidation Fraction
200.0	11.4	2126.	0.026
1,000.0	11.5	2132.	0.026
5,000.0	12.1	2192.	0.031
10,000.0	12.2	2188.	0.030
15,000.0	12.2	2198.	0.031
20,000.0	12.0	2198.	0.032
25,000.0	11.7	2164.	0.029
30,000.0	11.4	2130.	0.026

Source: NEDO-21697, August 1977

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 41 TO FACILITY LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

Introduction

By letter dated October 12, 1977, Vermont Yankee Nuclear Power Corporation (the licensee) requested approval of proposed Technical Specification changes for the Vermont Yankee Nuclear Power Station (the facility or VYNPS) relating to the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the cycle 5 core configuration.

Discussion

In December of 1976, the NRC staff was informed that certain input errors and computer code errors had been made in the facility's Emergency Core Cooling System (ECCS) analysis that was provided under the requirements of 10CFR50.46. An Order was issued to the Vermont Yankee Nuclear Power Corporation on March 11, 1977, requiring that "corrected revised calculations fully conforming to the requirements of 10CFR50.46 are to be provided for the (Vermont Yankee) facility as soon as possible." Our Order allowed VYNPS to continue operating with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values then existing in the Technical Specifications because the NRC staff was aware that revised modeling changes in the General Electric Company (GE) ECCS evaluation model would offset the effect of the errors. On April 12, 1977, the NRC staff approved the revised GE ECCS calculational model which incorporated the model changes.

The licensee submitted the new ECCS evaluation by letter dated August 12, 1977, in conjunction with information submitted in support of the facility's cycle 5 of operation. We concluded in our September 30, 1977 letter, which authorized cycle 5 operation, that having reviewed the revised ECCS evaluation, continued plant operation with the linear heat generation rates specified in our March 11, 1977 Order provided continued assurance that the ECCS would conform to the performance requirements of 10CFR50.46(b).

Evaluation

The revised ECCS analyses include a correction of all input errors previously made and correction of all computer code errors. The analyses were performed using the calculational model which contains model changes approved by the NRC staff in its Safety Evaluation issued April 12, 1977. The analyses were performed for the present cycle 5 fuel loading. Approximately 60 percent of the fuel assemblies in the cycle 5 reactor core are drilled fuel assemblies. Drilled fuel represents a major pathway for core spray to reach the lower plenum following a Loss of Coolant Accident (LOCA) thereby providing earlier reflooding and lower calculated peak cladding temperatures.

The concept of a lead plant analysis was used in determining the most limiting break size and location in the VYNPS ECCS analysis. The lead plant analysis provides an expanded documentation base to provide added insight into evaluation of the details of particular phenomena.

The VYNPS ECCS analysis references, as a lead plant analysis, the FitzPatrick Nuclear Power Plant (FitzPatrick) ECCS analysis which also included a correction of the input errors and incorporated the revised GE ECCS model changes. FitzPatrick is similar to VYNPS in that both plants are BWR/4 reactors with the low pressure coolant injection (LPCI) system modification (our FitzPatrick Safety Evaluation, dated September 16, 1977, discusses in detail the nature of the LPCI modification). We discuss below the results of our review of the non-lead plant analysis for VYNPS.

The VYNPS analysis represented the first non-lead plant analysis (that references FitzPatrick as the lead plant) to be submitted with the corrected model. The analysis provides all information requested in our letter to GE on June 30, 1977, on the number of breaks to be analyzed, documentation to be provided, etc. for the new analysis. Since this analysis references FitzPatrick as the lead plant analysis for BWR/4 plants with the low-pressure-coolant-injection (LPCI) system modification, the following description is provided of particular features of the FitzPatrick analysis as compared to the VYNPS analysis.

The FitzPatrick break spectrum (i.e., peak cladding temperature (PCT) versus break size) shows that the particular break producing the highest PCT for FitzPatrick is a recirculation pump discharge line break having an area approximately 80% as large as the largest discharge line break. The particular reasons why that size and location break is limiting for FitzPatrick are stated in detail in the FitzPatrick Evaluation. For the same reasons that are stated in the FitzPatrick Evaluation, the limiting break location for VYNPS is the recirculation discharge line.

However, unlike FitzPatrick where the worst size break at that location was 80% of the maximum pipe area, for VYNPS the worst size break at that location is the design basis accident (DBA), or 100% of the maximum pipe area. As explained in the FitzPatrick Evaluation:

1. FitzPatrick has a relatively small peripheral bypass area due to a relatively large number of fuel bundles in a relatively small reactor vessel. This makes FitzPatrick more likely than other plants to experience counter-current flooding (CCFL) effects in the bypass region.

2. FitzPatrick has holes drilled in the lower tie plates of all fuel bundles to enhance flow in the bypass area. These holes, at the bottom of the bypass region, are a major pathway for core spray water to reach the lower plenum following a LOCA and thereby contribute to the reflooding inventory, providing earlier reflooding and lower calculated PCT's. Any CCFL effects in the bypass area will delay such reflooding, causing a higher calculated PCT.

The FitzPatrick Evaluation explains how the above two effects combine with the effect of slower depressurization for smaller breaks (i.e., more lower plenum flashing steam is produced later for smaller breaks, which is when spray water is trying to go down through the bypass region). These effects combine to make the 0.8 times DBA break limiting for FitzPatrick.

VYNPS is much less sensitive to steam CCFL effects in the bypass region than is FitzPatrick. VYNPS has a larger bypass region which makes it less sensitive to CCFL effects. Also, VYNPS has only 220 of a total of 368 fuel bundles drilled (60%), whereas FitzPatrick has all 560 drilled (100%). Therefore, a smaller fraction of the spray water goes through the bypass region in VYNPS than in FitzPatrick due to the lesser number of holes, and the water that does go through the region is less affected by steam due to the larger area present at the top of the peripheral bypass area, where CCFL effects occur.

Therefore, the effects of more steam being produced later for smaller breaks do not dominate in VYNPS to produce a smaller-than-maximum size limiting break. Instead, the predominate effects for VYNPS are earlier departure from nucleate boiling and earlier high power plane uncovery for large break analyses compared to smaller break analyses. These effects cause the largest size discharge line break to be limiting for VYNPS. For the above reasons, we concur with the analysis provided by VYNPC that the limiting break for VYNPS is the largest recirculation discharge line break.

We have reviewed the corrected ECCS analysis for the cycle 5 core configuration at VYNPS. We conclude that the facility will be in conformance with all requirements of 10CFR50.46 and Appendix K to 10CFR50.46 when it is operated in accordance with the MAPLHGR limits given in the licensee's August 12, 1977 letter. These MAPLHGR limits are higher than the MAPLHGR limits specified in our March 11, 1977 Order.

Technical Specifications

The licensee's August 12, 1977, submittal showed the increased MAPLHGR limits by use of Tables 4A through 4F. By letter of October 12, 1977, the licensee redesignated these tables 3.11-1A through 1F for use in the Technical Specifications. We conclude that the proposed Technical Specifications are acceptable and consistent with those of other facilities operating with similar systems and found acceptable by the NRC Staff.

VYNPS is much less sensitive to steam CCFL effects in the bypass region than is FitzPatrick. VYNPS has a larger bypass region which makes it less sensitive to CCFL effects. Also, VYNPS has only 220 of a total of 368 fuel bundles drilled (60%), whereas FitzPatrick has all 560 drilled (100%). Therefore, a smaller fraction of the spray water goes through the bypass region in VYNPS than in FitzPatrick due to the lesser number of holes, and the water that does go through the region is less affected by steam due to the larger area present at the top of the peripheral bypass area, where CCFL effects occur.

Therefore, the effects of more steam being produced later for smaller breaks do not dominate in VYNPS to produce a smaller-than-maximum size limiting break. Instead, the predominate effects for VYNPS are earlier departure from nucleate boiling and earlier high power plane uncovery for large break analyses compared to smaller break analyses. These effects cause the largest size discharge line break to be limiting for VYNPS. For the above reasons, we concur with the analysis provided by VYNPC that the limiting break for VYNPS is the largest recirculation discharge line break.

We have reviewed the corrected ECCS analysis for the cycle 5 core configuration at VYNPS. We conclude that the facility will be in conformance with all requirements of 10CFR50.46 and Appendix K to 10CFR50.46 when it is operated in accordance with the MAPLHGR limits given in the licensee's August 12, 1977 letter. These MAPLHGR limits are higher than the MAPLHGR limits specified in our March 11, 1977 Order.

Technical Specifications

The licensee's August 12, 1977, submittal showed the increased MAPLHGR limits by use of Tables 4A through 4F. By letter of October 12, 1977, the licensee redesignated these tables 3.11-1A through 1F for use in the Technical Specifications. We conclude that the proposed Technical Specifications are acceptable and consistent with those of other facilities operating with similar systems and found acceptable by the NRC Staff.

Environmental Consideration

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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: November 30, 1977

7590-01

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. 41 to Facility Operating License No. DPR-28, issued to Vermont Yankee Nuclear Power Corporation (the licensee), 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 modifies the Technical Specifications relating to an increase in the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the facility. This increase in the MAPLHGR limits is based on the results of a new evaluation of the Emergency Core Cooling System (ECCS) performance submitted in compliance with our Order for Modification of License dated March 11, 1977. This amendment terminates the March 11, 1977 Order.

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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on October 17, 1977 (42 F.R. 55507). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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 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 October 12, 1977, (2) Amendment No. 41 to License No. DPR-28 and the Commission's related Safety Evaluation, (3) Amendment No. 39 to License No. DPR-28 and the Commission's related Safety Evaluation, (4) the Commission's Order for Modification of License dated March 11, 1977, and (5) the licensee's ECCS Submittal dated August 12, 1977. 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), (3) and (4) 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 30th day of November 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors