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OCTOBER 26 1979

Docket No. 50-271

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. 55 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 7 operation, and the surveillance requirements associated with your control rod hydraulic return line isolation valves. These changes are in response to your submittals dated August 21, 1979, October 5, 1979 and October 5, 1979. 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.

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. 55 to DPR-28
2. Safety Evaluation
3. Notiee

cc w/enclosures: see next page

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SURNAME ▶	SSheppard	VRooney:acr		TIppolito	WGammill	
DATE ▶	10/22/79	10/22/79	10/24/79	10/23/79	10/23/79	

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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. 55
License No. DPR-28

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Vermont Yankee Nuclear Power Corporation (the licensee) dated August 21, 1979, as supplemented October 5, 1979 and October 5, 1979, 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.

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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. 55, 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. Ippolito

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 26, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 55

FACILITY OPERATING LICENSE NO. DPR-28

DOCKET NO. 50-271

Revise Appendix A as follows:

Remove the pages listed and replace with revised pages.

135
136
180-n5
180-01

TABLE 4.7.2.a

PRIMARY CONTAINMENT ISOLATION VALVES
VALVES SUBJECT TO TYPE C LEAKAGE TESTS

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

Table 4.7.2.b

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

Isolation Group (Note 1)	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec)	Normal Position	Action on Initiating Signal
2	RHR Return to Suppression Pool (10-39A, B)		2	70	Closed	SC
2	RHR Return to Suppression Pool (10-34A, B)		2	120	Closed	SC
2	RHR Drywell Spray (10-26A, B & 10-31A, B)		4	70	Closed	SC
2	RHR Suppression Chamber Spray (10-38A, B)		2	45	Closed	SC
3	Containment Air Compressor Suction (72-38A, B)		2	20	Open	GC
3	Containment Air Sampling System (109-75A, D; 1, 2 109- 76A, B)		10	5	Open	GC
4	RHR Shutdown Cooling Supply (10-18, 10-17)	1	1	28	Closed	SC
4	RHR Reactor Head Cooling (10-32, 10-33)	1	1	25	Closed	SC
	Feedwater Check Valves (2-28 A, B)	2	2	NA	Open	Proces
	Reactor Head Cooling Check Valve (10-29)	1		NA	Closed	Process
	Standby Liquid Control Check Valves (11-16, 11-17)	1	1	NA	Closed	Proces

Table 3.11-1G

MAPLHGR, PCT, Oxidation Fraction Versus Exposure,
Fuel Type 8DPB289 and P8DPB289

Average Planar Exposure (MWDd/t)	MAPHLGR (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

Exposure Range	Value of "N" in RBM Equation (1)	7x7 (2)	8x8	8x8R	P8x8R
BOC to EOC-2 GWd/t	42%		1.27	1.22	1.27
	41%		1.24	1.22	1.24
	≤40%		1.23	1.22	1.22
EOC-2 GWd/t to EOC-1 GWd/t	42%		1.27	1.23	1.27
	41%		1.24	1.23	1.24
	≤40%		1.23	1.23	1.24
EOC-1 GWd/t to EOC	42%		1.28	1.28	1.30
	41%		1.28	1.28	1.30
	≤40%		1.28	1.28	1.30

(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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. DPR-28

VERMONT YANKEE NUCLEAR POWER CORPORATION

VERMONT YANKEE NUCLEAR POWER STATION

DOCKET NO. 50-271

1.0 Introduction

Vermont Yankee Nuclear Power Corporation (VYNPC or licensee) has proposed changes to the Technical Specifications of the Vermont Yankee Nuclear Power Station (VY) in Reference 1 and as supplemented by Reference 2. The proposed changes relate to the replacement of fuel assemblies constituting refueling of the core for Cycle 7 operation at power levels up to 1665 MWt (100% power). In support of the reload application, the licensee has enclosed proposed Technical Specification changes in Reference 1 and the GE BWR supplemental licensing submittal (Reference 3).

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

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 design parameters are provided in Reference 4. Additional plant and cycle dependent information is provided in the reload application (Reference 3) which closely follows the outline of Appendix A of Reference 4.

Appendix C of Reference 4 includes a description of the staff's review, approval, and conditions of approval for the plant-specific data addressed in Reference 4. The above-mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

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Our safety evaluation (Reference 4) of the GE generic reload licensing topical report has also concluded that the nuclear and mechanical design of the 8x8R fuel, and GE's analytical methods for nuclear and thermal-hydraulic calculations as applied to mixed cores containing 8x8 and 8x8R fuel, are acceptable. Approval of the application of the analytical methods did not include plants incorporating a prompt recirculation pump trip (RPT) or Thermal Power Monitor (TPM).

Because of our review of a large number of generic considerations related to use of 8x8R fuel in mixed loadings, and on the basis of the evaluations which have been presented in Reference 4, only a limited number of additional areas of review have been included in this safety evaluation report. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 4.

This report also addresses proposed Technical Specification changes submitted by VYNPC in reference 7. These changes, which concern surveillance of control rod hydraulic return line isolation valves, are discussed in Section 2.6 of this report.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 7 operation of Vermont Yankee, 96 fresh P8x8R fuel bundles of type P8DPB289 will be loaded into the core (Ref. 3). The remainder of the 368 fuel bundles in the core will be 68 8DB274L bundles, 124 8DB274H bundles, 120 8DB219L bundles, and 60 8DPB289 bundles. These are all previously irradiated bundles.

Based on the data provided in Reference 3, both the control rod system and the standby liquid control system will have acceptable shutdown capability during Cycle 7.

2.2 Thermal Hydraulics

2.2.1 Fuel Cladding Integrity Safety Limit MCPR

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

The 1.07 SLMCPR to be used for Cycle 7 is unchanged from the SLMCPR previously approved for Cycle 6. The basis for this safety limit is addressed in Reference 4, while our generic approval of the limit is given in the staff evaluation included in Reference 4.

2.2.2 Operating Limit MCPR

Various transient events can reduce the MCPR from its normal operating level. To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient, the most limiting transients have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for the exposed 8x8 fuel and the exposed and fresh 8x8R fuel. Addition of the largest reductions in critical power ratio to the SLMCPR establishes the operating limits for each fuel type.

2.2.2.1 Transient Analysis Methods

The generic methods used for these calculations, including cycle-independent initial conditions and transient input parameters, are described in Reference 4. The staff evaluation, included as Appendix C of Reference 4, contains our acceptance of the cycle-independent values. Additionally, Appendix C contains our evaluation of the transient analysis methods, together with a description and summary of the outstanding issues associated with these methods. Supplementary cycle-independent initial conditions and transient input parameters used in the transient analyses appear in the tables in Sections 6 and 7 of Reference 3. Our evaluation of the methods used to develop these supplementary input values is also included in Appendix C of Reference 4.

2.2.2.2 Transient Analysis Results

The transients evaluated were the limiting pressure and power increase transient (turbine trip without bypass in this case), the limiting coolant temperature decrease transient (loss of a feedwater heater), the feedwater controller failure transient, and the control rod withdrawal error transient. Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 3 were assumed.

The results of these analyses are outlined in Reference 3 sections 9 and 10. On this topic, Reference 6 found it acceptable if fuel specific operating limits are established for prepressurized fuel as has been done for VY. On this basis, the transient analysis results are acceptable for use in the evaluation of the operating limit MCPR. Based on this, the proposed Technical Specification modifications to operating limit MCPR are acceptable.

2.3 Accident Analyses

2.3.1 ECCS Appendix K Analysis

For the previous cycle, the licensee re-evaluated the adequacy of VY's ECCS performance in connection with the retrofit 8x8 reload fuel design. The methods used in this analysis were previously approved by the staff. For that cycle, we reviewed the ECCS analysis results submitted by the licensee and concluded that VY would be in conformance with all the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 when operated in accordance with the MAPLHGR 8x8R versus Average Planar Exposure values which appeared in the proposed plant Technical Specifications. In Reference 6, we have concluded that MAPLHGR limits for prepressurized fuel is conservatively bound by the values for the non-prepressurized fuels. VY has conservatively used the non-prepressurized values. Therefore, based on our conclusions of Reference 6, the proposed MAPLHGR limits are acceptable.

2.3.2 Control Rod Drop Accident

For VY Cycle 7, the accident reactivity shape function (cold) does not satisfy the requirements for the bounding analyses described in Reference 4. Therefore, it was necessary for the licensee to perform a plant and cycle specific analysis for the control rod drop accident. The results of this analysis indicated that the peak fuel enthalpy for this event would be at most 135 calories per gram.(3) Since this is well below the criterion of 280 calories per gram, we find the results of this analysis to be acceptable.

2.3.3 Fuel Loading Error

Potential fuel loading errors involving misoriented bundles and bundles loaded into incorrect positions have been analyzed. This GE method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff and is part of the Reference 4 methodology. In order to address our concerns on the fuel loading errors for the previous cycle, the licensee and we agreed to an MCPR adjustment on radiological indications of potential fuel loading errors. The licensee has proposed similar requirements for this cycle which we find acceptable.

2.3.4 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 4. As specified in the staff evaluation included in Reference 4, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable.

2.4 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis(3) 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 at greater than 1% rated thermal power will be prohibited by Technical Specifications, there will be added margin to the stability limit and this is acceptable.

2.5 Startup Test Program

The licensee has not changed his startup test program from that approved for the previous cycle. This program therefore remains acceptable.

2.6 Technical Specifications

The only change to the VY Technical Specifications involving core refueling that has not yet been discussed is the elimination of operating limit MCPR for 7x7 fuel and a statement that 7x7 fuel MCPR limits have not been established and that future use of 7x7 fuel would require further evaluation. We and the licensee have agreed to such a specification.

In reference 7 the VYNPC requested that Table 4.7.2.6 of the Technical Specifications be changed to delete valves V 3-110 and V 3-113 and add valve V 3-181 to the listing of valves subject to Type C leakage tests. This change corrects Table 4.7.2.6 to delete two valves which no longer exist in the control rod hydraulic return line and adds an additional valve to the table. This change is administrative in nature and is acceptable.

3.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: October 26, 1979

References

1. Letter, D. E. Vandenburgh (VYNPC) to Office of Nuclear Reactor Regulation (USNRC), dated August 21, 1979.

2. Letter, D. E. Vandenburgh (VYNPC) to Office of Nuclear Reactor Regulation (USNRC), dated October 5, 1979.

3. "Supplemental Reload Licensing Submittal for Vermont Yankee Nuclear Power Station Reload 6," NEDO-24208, August 1979.

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

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

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

7. Letter, D. E. Vandenburgh (VYNPC) to Office of Nuclear Reactor Regulation dated October 5, 1979, WVY 79-113.

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. 55 to Facility Operating License No. DPR-28, issued to Vermont Yankee Nuclear Power Corporation which revised Technical Specifications for operation of the Vermont Yankee Nuclear Power Station (the facility) located near Vernon, Vermont. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to incorporate the limiting conditions for operation associated with cycle 7 operation, and the surveillance requirements associated with the control rod hydraulic return line isolation valves.

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

For further details with respect to this action, see (1) the application for amendment dated August 21, 1979, as supplemented October 5, 1979 and October 5, 1979, (2) Amendment No. 55 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 26th day of October 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

T. Ippolito

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