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Docket No. 50-271

Yankee Atomic Electric Company
ATTN: Mr. G. Carl Andognini
Assistant to the Vice President
20 Turnpike Road
Westboro, Massachusetts 01581

Gentlemen:

Enclosed is a signed original of the "Order for Modification of License" issued by the Commission for the Vermont Yankee Nuclear Power Station. The Order revises, in its entirety, Appendix A to the Order for Modification of License dated December 27, 1975. However, all other provisions of the Order shall remain in full force and effect. The enclosed Order also authorizes operation of the facility with plugged bypass flow holes in accordance with the restrictions set forth in the revised Appendix A. A copy of the Order is being filed with the Office of the Federal Register for publication.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

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

Enclosures:
1. Order for Modification of License
2. Safety Evaluation

cc: See next page

Dispatched 9/2

	ORB#4	OELD	RL:AD/ORs	RL	NRR	NRR
OFFICE	ORB#4	OELD	RL:AD/ORs	RL	NRR	NRR
SURNAME	RIngram/dg; RWReid/ DiBenedetto	TR VStello	KRGoller	RBoyd	ECase	BCRusche
DATE	8/20/75	8/22/75	8/22/75	8/22/75	8/22/75	8/22/75

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cc w/enclosures:
Mr. James E. Griffin, President
Vermont Yankee Nuclear Power Corporation
77 Grove Street
Rutland, Vermont 05701

Mr. Donald E. Vandenburgh, Vice President
Vermont Yankee Nuclear Power Corporation
Turnpike Road, Route 9
Westboro, Massachusetts 01581

John A. Ritsher, Esquire
Ropes and Gray
225 Franklin Street
Boston, Massachusetts 02110

Gregor I. McGregor, Esquire
Assistant Attorney General
Department of the Attorney General
State House, Room 370
Boston, Massachusetts 02133

Richard E. Ayres, Esquire
Natural Resources Defense Council
1710 N Street, N. W.
Washington, D. C. 20036

Honorable M. Jerome Diamond
Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

John A. Calhoun
Assistant Attorney General
State of Vermont
109 State Street
Pavilion Office Building
Montpelier, Vermont 05602

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
1712 N Street, N. W.
Washington, D. C. 20036

John R. Stanton, Director
Radiation Control Agency
Hazen Drive
Concord, New Hampshire 03301

John W. Stevens
Conservation Society of Southern Vermont
P. O. Box 256
Townshend, Vermont 05353

Mr. David M. Scott
Radiation Health Engineer
Agency of Human Services
Division of Occupational Health
P. O. Box 607
Barre, Vermont 05641

New England Coalition on Nuclear Pollution
Hill and Dale Farm
West Hill - Faraway Road
Putney, Vermont 05346

Mr. Raymond H. Puffer
Chairman
Board of Selectman
Vernon, Vermont 05354

cc w/enclosures and cy of VY's filing dtd. 8/6/75
Mr. Martin K. Miller, Chairman
State of Vermont
Public Service Board
120 State Street
Montpelier, Vermont 05602

Mr. Wallace Stickney
Environmental Protection Agency
JFK Federal Building
Boston, Massachusetts 02203

additional cc: see next page

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cc w/enclosures and cy of VY's
filing dtd. 8/6/75:
Chief, TIRB
Technology Assessment Division
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 647A East Tower
Waterside Mall
401 M. Street, S. W.
Washington, D. C. 20460

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
VERMONT YANKEE NUCLEAR POWER CORPORATION	)	Docket No. 50-271
	)	
(VERMONT YANKEE NUCLEAR POWER STATION)	)	

ORDER FOR MODIFICATION OF LICENSE

I.

Vermont Yankee Nuclear Power Corporation (the licensee) is the holder of Facility Operating License No. DPR-28 which authorizes operation of the Vermont Yankee Nuclear Power Station (the facility) at steady-state reactor core power levels not in excess of 1593 megawatts thermal (rated power). The facility is a boiling water reactor (BWR) located near Vernon, Vermont.

II.

1. On July 23, 1975, the Nuclear Regulatory Commission (the Commission) issued an "Order for Modification of License" (40 Fed. Reg. 32180, July 31, 1975) which confirmed a plan for limited additional operation of the facility. As detailed in the Order, the facility's channel box wear, as indicated by the noise-to-signal ratio recorded by the traversing incore probe (TIP), had exceeded the remedial action threshold. The remedial plan confirmed by the Order contemplated operation of the facility for a limited period of time (until August 3, 1975) at not more than 80% of rated core power and 70% of rated core flow, provided the TIP noise-to-signal ratio at those levels did not exceed 0.05. In addition, the Order permitted operation up to full flow and power for a brief period of time as necessary to obtain baseline TIP data.

2. On August 1, 1975, the Commission issued an "Order for Modification of License" (40 Fed. Reg. 33739, August 11, 1975) which modified the July 23, 1975 Order to extend operation for an additional three days until August 6, 1975. The basis for this action was the licensee's request dated July 31, 1975. The licensee's letter states that the request is made at the behest of the New England Power Exchange based upon a serious power shortage resulting from the unscheduled outage of several units and forecasted weather conditions. Our evaluation of the request concluded that the recently obtained TIP traces did not show any accelerated channel box wear, and that operation of Vermont Yankee for an additional three days beyond the period contemplated by our previous safety evaluation was acceptable since no appreciable additional wear would be incurred.

3. By its letter dated July 17, 1975, the licensee formally proposed a plan, previously discussed with the Commission, setting forth a course of remedial action. The plan, as modified by the licensee's letter dated July 31, 1975, entailed continuation of operation at 80% of rated core power and 70% of rated flow until a shutdown not later than August 6, 1975, with the exception of a brief period of operation at full flow and power immediately prior to shutdown as necessary to obtain baseline TIP data for use in connection with the shutdown inspection and in connection with future operations. During the shutdown, worn channel boxes were to be replaced as necessary, and plugs to be inserted in the bypass holes. The reactor was shutdown on August 6, 1975, for visual inspection of the channel boxes and the necessary repairs.

4. By letter dated July 30, 1975[1/], the licensee provided details relating to the inspection and repair program for correction of channel box wear and to the installation of core bypass flow plugs in the lower core plate and supplied analyses to demonstrate the adequacy of such plugs and the adequacy of the procedures for plug installation.

5. On August 15, 1975, the Commission issued an "Order for Modification of License" that approved the repair program and authorized the installation of bypass hole plugs in the facility's lower core plate. As discussed in the August 15, 1975 Order, the NRC staff concluded that the plugs will reduce the vibration of the instrument thimbles caused by flow through the bypass holes. By telecon dated August 22, 1975, Vermont Yankee has confirmed that the licensee's inspection and repair program has been completed. This resulted in the rejection of 24 channel boxes with unacceptable wear as defined in the repair program. Thirty-eight channel boxes with indications of wear, but within the criteria of the repair program, were reinstalled in the reactor in locations which are not next to instrument channels in accordance with the repair program. Vermont Yankee also confirmed that all flow bypass holes in the core plate were plugged.

6. The licensee's July 30, 1975 letter and their letter dated August 6, 1975 provided analyses, including a emergency core cooling performance analysis, for reactor power operation with the plugs installed in the bypass holes.

1/ Copies of (1) the licensee's letters of July 30 and August 6, 1975, and (2) the NRC staff Safety Evaluation of Vermont Yankee Nuclear Power Station Operation with Plugged Bypass Flow Holes and the documents referenced therein, are available for public inspection in the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and are being placed in the Brooks Memorial Library, 224 Main Street, Brattleboro, Vermont.

7. The Commission's staff has reviewed the analyses submitted by the licensee on July 30, and August 6, 1975, to support operation with bypass flow plugs installed. As discussed in the Commission's concurrently issued Safety Evaluation, Vermont Yankee Nuclear Power Station Operation with Plugged Bypass Flow Holes, the proposed operation with plugs will require that certain modifications be made to earlier restrictions set forth in the December 27, 1974 Order for Modification of License (40 Fed. Reg 1778, Jan. 9, 1975) relating to the emergency core cooling performance. In this regard, it is appropriate to replace the original Appendix A to the December 27, 1974 Order with a revised Appendix A listing restrictions for operation with bypass flow plugs installed. All other provisions of the December 27, 1974 Order remain in full force and effect. It should also be noted that plugs identical to those installed in the Vermont Yankee reactor have previously been installed in both the Vermont Yankee and Pilgrim reactors in 1973 and 1974, respectively, to eliminate the vibration of temporary control curtains that caused channel box wear in those reactors. They have also been installed in the Duane Arnold reactor to mitigate channel box wear. After ten months of successful service the previous plugs in the Vermont Yankee reactor were removed at the time that the temporary curtains were removed.

8. Based on a review of the licensee's submittals of July 30 and August 6, 1975, and the prior related experience at the Pilgrim and Vermont Yankee reactors, the NRC staff concluded in its concurrently issued Safety Evaluation that operation of the Vermont Yankee reactor in accordance with the additional restrictions set forth in Appendix A to the Safety Evaluation will provide reasonable assurance that the public health and safety will not be endangered. These additional restrictions are set forth as Appendix A to this Order.

III.

Accordingly, pursuant to the Atomic Energy Act of 1954, as amended, and the Commission's Rules and Regulations in 10 CFR Parts 2 and 50, IT IS ORDERED THAT:

1. The Order for Modification of License dated December 27, 1974 be amended by replacing Appendix A of that Order with Appendix A attached to this Order. All other provisions of the December 27, 1974 Order shall remain in full force and effect.

2. Operation of the Vermont Yankee Nuclear Power Station with plugged bypass flow holes is hereby authorized subject to the restrictions set forth in the Order for Modification of License, dated December 27, 1974 as amended by paragraph 1 above.

FOR THE NUCLEAR REGULATORY COMMISSION

Ben C. Rusche

Ben C. Rusche, Director
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 22nd day of August, 1975.

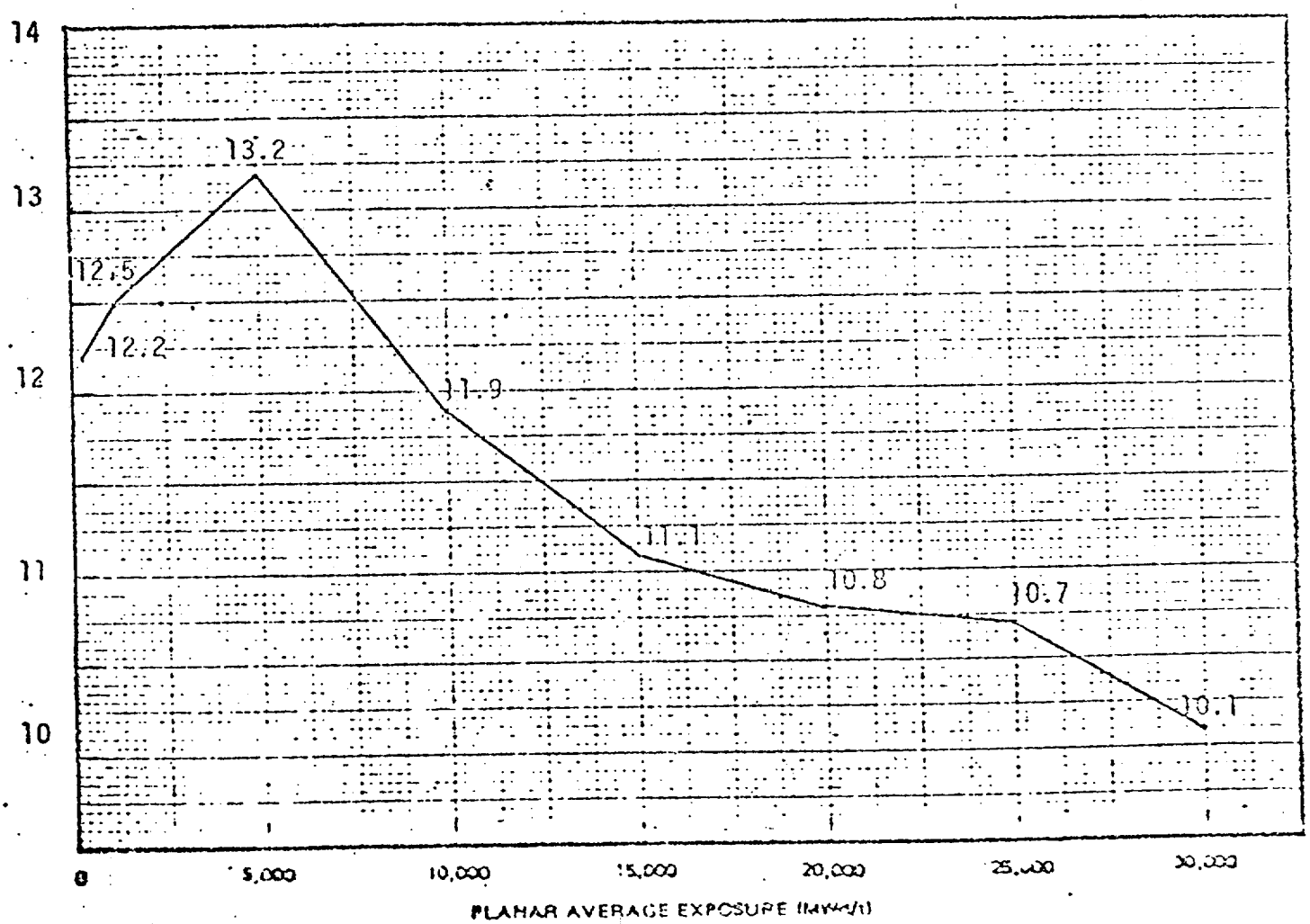
Attachment:
VYNPS Operating Restrictions

ATTACHMENT

APPENDIX A

VERMONT YANKEE NUCLEAR POWER STATION
OPERATING RESTRICTIONS

There are two limitiations on the continued operation of the reactor for the remainder of this fuel cycle. These are the limiting assembly maximum average planar linear heat generation rate, MAPLHGR, and the minimum critical power ratio limit related to boiling crisis, MCPR. Operation shall conform to a MCPR value of equal to or greater than 1.28 as proposed by the licensee. The limiting value of MAPLHGR included with the proposed Technical Specifications submitted on October 31, 1974 have been revised to account for the staff requirements of December 27, 1974 and the proposed operation with plugged bypass holes. The revised values are given in Figures 1 and 2 for Generic B 7x7 Fuel and 8D219 8x8 Fuel types. A maximum value linear heat generation rate of 13.4 $kw/_{ft}$ for 8x8 fuel shall remain in effect until the Commission completes its review of the licensee's proposed increase in linear heat generation rate.



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE

Vermont Yankee Nuclear Power Station
ECCS Analysis Using December 28, 1974 Methods
Generic B 7x7 Fuel Type

Figure 1

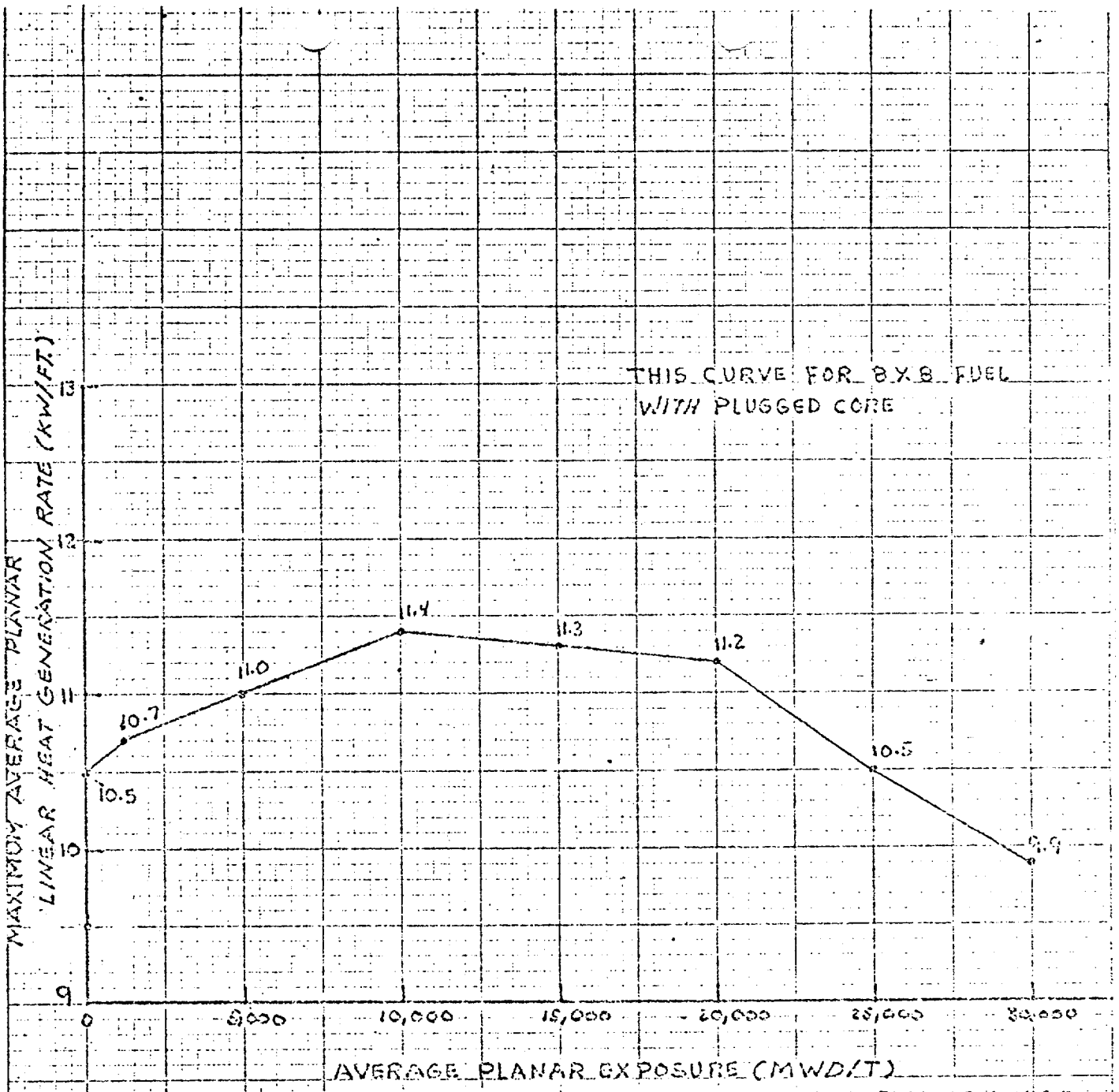


FIGURE 2 MAXIMUM ALLOWABLE PLANAR LHGR

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

SAFETY EVALUATION REPORT

DOCKET NO. 50-271

VERMONT YANKEE NUCLEAR POWER STATION

OPERATION WITH PLUGGED BYPASS FLOW HOLES

1. Introduction

Vermont Yankee Nuclear Power Corporation submitted References 1 through 4 to support continued operation of the Vermont Yankee Nuclear Power Station for the remainder of this fuel cycle. The principal change in operation is the plugging of the bypass flow holes in the core support plate in order to reduce instrument tube-fuel channel interaction.

2. Summary

Based on this review, we have drawn the following conclusions regarding the proposed operation of the Vermont Yankee Nuclear Power Station with plugged bypass holes.

a. The nuclear, mechanical and thermal-hydraulic characteristics of the core are acceptable.

b. The use of plugged bypass flow holes will significantly reduce instrument tube-channel interaction that has caused excessive wear of some channels.

c. The overpressurization protection satisfies ASME code requirements for the reactor coolant system.

d. Safety analyses show that the core will not violate limiting thermal margins if the plant is operated with a steady-state MCPR equal or greater than 1.28.

e. The MAPLHGR limits submitted October 31, 1975 have been revised to account for the staff requirements of December 27, 1974 and for operation with plugged hole

f. Continued surveillance during operation is required for monitoring any undesirable instrument tube-channel interaction.

Operating restrictions for the remainder of this cycle are presented in Appendix A.

3.0 Nuclear Design

The primary nuclear effect caused by plugging the bypass flow holes is an increased bypass void fraction and a reduction in the average in-channel void fraction. The in- and out-of-channel void fraction changes give a net increase in the core average void fraction.

At steady state conditions, the increased bypass void fraction results in a small reduction in the maximum local peaking factor within a fuel bundle and an increase in the local bundle power calculational uncertainty. Another consequence of the reduced bypass flow is a small reduction in the infinite multiplication factor of uncontrolled fuel.

The presence of voids in the bypass region affects the relationship between the travelling incore prope (TIP) signal and the local bundle power. The TIP signal is reduced by the presence of voids and could lead to an underprediction of the peak heat flux. The relationship of the power in the four bundles surrounding a TIP instrument tube and the TIP signal as a function of bypass voids was determined by the General Electric Company (GE) by performing three group, two-dimensional diffusion theory calculations. A correction factor was developed and algorithms for computing the bypass void fraction and for making appropriate corrections in the local bundle power have been incorporated in the process computer.

The uncertainty in the local bundle power caused by bypass voids is taken into account in determining the MCPR safety limit. The TIP uncertainty introduced by the bypass voids is zero in the bottom half of the core and increases from 3.95% at the core mid-plane to 4.53% at the core exit.

After the bypass flow holes are plugged, most of the fuel will be placed in its original core location. Seventy six of 136 fuel bundles will be moved to new positions in the core but quadrant symmetry will be maintained. The following observations can be made:

(1) the control rod worths are not significantly changed and, consequently, the previous results of the control rod drop analysis remain valid,

(2) the shutdown margin will remain the same as previously analyzed,

(3) the standby liquid control system reactivity insertion rate and magnitude will not be affected.

We have reviewed the proposed core configuration and find it to be a minor change from the original core. We conclude that the analysis of the nuclear performance of the plant with plugged bypass holes is acceptable.

4.0 Mechanical Design

The only mechanical design change in the reactor is the use of plugs to fill the bypass flow holes[1]. The plug consists of two stainless steel parts (body and shaft) which are connected by an Inconel spring. The shoulder of the body rests on the top of the core plate along the rim of a one-inch bypass hole and is pressed down by the spring. An equal and opposite force is applied on the shaft. A stainless steel latch is connected to the bottom of the shaft by means of a pin. This latch is free to rotate about the pin and latches the shaft to the core plate. The spring exerts a minimum of 38 pounds on the body and latch and a maximum of 46 pounds (with the worst tolerance combination).

Removal of a plug can be accomplished by applying about 500 pounds of force and deforming the latch plastically. More than 10 plugs were removed in tests performed at the GE test facility with consistent latch deformations without damaging other parts.

Plugs identical to those to be used in the Vermont Yankee reactor were installed once before in Vermont Yankee and recently in the Duane Arnold and Pilgrim reactors. The plugs installed previously in Vermont Yankee were removed during a refueling operation after 10 months of successful service. No abnormalities or loose pieces were reported.

Pressure differentials across the core plate during normal steady state operation and following a steam line break accident are expected to be on the order of 17 to 32 psi. These loads together with the spring preload will produce yielding of the latch in bending but will be significantly below the 500 pounds of force necessary for removing the plug. The 1973 GE full scale flow mockup test shows that, with up to 40 psi differential pressure, there is negligible leakage flow through the plugged holes. No plug vibration was

observed during the test and no apparent deformation on the latch was evident after the test. As previously mentioned, approximately 500 pounds were required to deform the latch plastically and remove it from the core plate. No fatigue and plastic strain ratcheting is expected since the plant power cycle during the anticipated service period will be minimal.

Stainless steel and Inconel are compatible with other reactor internals and are not expected to introduce any unusual oxidation and stress corrosion problems. The flux level at the core plate elevation is estimated to be quite low and an insignificant reduction in ductility due to irradiation is anticipated. GE has performed creep tests with both Inconel springs and stainless steel latches and found that stress relaxation or creep deformation were insignificant. The tests were performed at 550°F.

Vermont Yankee presented to the NRC staff a summary of channel inspections on BWR-2s and BWR-3s[1]. These older plants have instrument tubes similar to Vermont Yankee, but no bypass flow holes in the core support plate. The bypass flow for these plants enters through clearances in the assembly end fittings, which is similar to the proposed Vermont Yankee configuration with plugged bypass holes. Seventy-five channels (adjacent to instrument tubes and source tubes) were inspected during normal fuel outages in 7 plants. No significant channel wear was observed at the corners adjacent to the instrument tubes.

General Electric has a design criteria for channel box wastage of 0.010 inches for the lower 80 inches of the channel and 0.020 inches for the remaining length. All of the channels (new and old) in the core will meet this requirement. Channels with observed acceptable wear on the corner will not be reinserted in the core next to an in-core instrument where additional wear could occur during subsequent reactor operation.

Based on a review of the design, the test rig, the installation methods and primarily the previously successful operating experience at Vermont Yankee and Pilgrim, we conclude that the plugs will not fail so as to result in loose parts in the core or result in unplugging of the bypass flow holes. Also, we conclude that the installed plugs will substantially reduce the instrument tube vibration, due to flow through the bypass holes, sufficient to preclude any unacceptable wear for at least the proposed fuel cycle.

5.0 Thermal-Hydraulic Design

The fuel cladding integrity safety limit for Vermont Yankee has been changed to a minimum critical power ratio (MCPR) based on a thermal margin correlation, GEXL[3], which the staff previously has found acceptable[4]. The fuel cladding integrity safety limit MCPR for this fuel cycle is 1.06, based on a statistical analysis for which 99.9% of the fuel rods in the core are expected to avoid boiling transition. The input list of uncertainty effects of the core operating parameters and calculated parameters associated with the GEXL correlation plus the GETAB relative bundle power histogram used in the statistical analysis is acceptable to the staff.

The tabulated list of uncertainties[2] shows a standard deviation of 8.7% for the TIP readings plus a 3.95 to 4.53% standard deviation due to voids in the bypass region.

Conservatism was applied to the axial power shape because the axial power peak is assumed to be at the midplane of the core, (peaking factor of 1.5). Bottom peaked axial shapes, which are obtained during reactor operation would reduce the required safety limit MCPR.

As discussed in the following section, the operating MCPR requirement is 1.28 based on the most limiting transient, turbine trip without bypass from rated conditions.

The plugged bypass flow holes increase the core hydraulic resistance which reduces the recirculation flow rate by 2 percent. However, the assembly flow rates are increased while the total bypass flow is decreased.

The stability of the core was analyzed based on the most limiting conditions of natural circulation and 51.5% power. The analysis, which is similar to that reported in the FSAR, showed that the decay ratios for both the channel and the core decreased from the values presented in the FSAR. Based on the analyses presented, operation with plugged bypass holes results in improved stability for the channel performance and core performance.

The staff concludes that the steady state thermal-hydraulic design is acceptable for operation with plugged bypass flow holes based on the above considerations.

6.0 Safety Analyses

6.1 Abnormal Transients

The licensee reanalyzed three abnormal transients - turbine trip, loss of feedwater heater, and rod withdrawal error - as the most limiting events to be considered. The main factors affecting the plant transient analyses are the moderator void coefficient of reactivity, the Doppler coefficient of reactivity, and the full power scram reactivity function. The Doppler coefficient of reactivity is affected by the changes in the moderator density in the fuel channel and bypass region primarily through changes in the Dancoff Ginsburg rod shadowing effect. This effect is small and insignificantly affects the Doppler coefficient of reactivity. The full power scram reactivity function for the end-of-cycle with plugged bypass flow holes indicates a total scram worth of -37.05 dollars. This is more scram worth than the previously determined value of about -30 dollars and is due only to a recalculation of the Vermont Yankee end-of-cycle reactivity and not to any effects caused by changed void distributions.

The moderator void coefficient of reactivity used in the safety analysis of the Vermont Yankee plant with plugged bypass flow holes is more negative than used in the FSAR for two reasons. The first cause is a renormalization of the void coefficient calculations based on analyses of operating BWR data. This effect, of the order of 15 to 20 percent, is unrelated to the plugging of the bypass flow holes. The second cause is the increase in the amount of voids present in the bypass region after the bypass flow holes are plugged.

The limiting transient is a turbine trip with failure of bypass valves to open. The analyses was initiated from 103 percent design power and the scram was initiated by the position switch on the turbine stop valves. A peak pressure of 1235 psig was calculated at the bottom of the vessel. The decrease in MCPR is 0.22 which is the limiting change in thermal margin. As a result, the steady-state MCPR must be equal or greater than 1.28 to satisfy the safety limit MCPR of 1.06. The decrease in MCPR for a loss of feedwater heater (100°F in feedwater temperature) is only 0.14.

The licensee also provided an overpressure analysis assuming closure of all main steamline isolation valves with an indirect scram. For this case the plant was assumed to be operating at 104.5% power, no credit was taken for relief function, scram was initiated by high neutron flux, and failure of a single safety/relief valve was assumed. A pressure of 1306 psig was calculated at the bottom of the vessel which meets staff requirement that the pressure be below the 110% of design pressure limit specified in Section III of the ASME Boiler and Pressure Vessel Code.

The rod withdrawal error was analyzed for a limiting control rod pattern. The results of the analysis indicate that a Rod Block Monitor (RBM) setpoint of 106% of full power will provide, for the worst case failure of Local Power Range Monitor (LPRM) detectors, a rod block at approximately 6 feet of rod withdrawal for the withdrawing rod. The MCPR at this point will be about 1.12 and the cladding strain will be less than 1.0%.

The staff finds the responses of the abnormal transients acceptable and the overpressurization protection with plugged bypass flow holes meets the ASME Code criteria.

6.2 Loss-of-Coolant Accident

The licensee analyzed the design basis loss-of-coolant accident with the bypass flow holes plugged, applying methods used for the December 27, 1974 operating bases to determine the maximum average planar linear heat generation rate (MAPLHGR) versus exposure for Generic B 7 x 7 and 8D219 8 x 8 fuel types[2].

The calculation was performed using procedures described by General Electric in their December 20, 1974 letter from G. Gyorey to V. Stello, NRC. The licensee applied a MAPLHGR penalty or reduction to their October 31,1974 submittal as revised to account for the staff requirements of December 27, 1974 for the longer delayed flooding time which occurs when the bypass holes are plugged.

We modified the 8x8 type fuel MAPLHGR curve, Figure 2, using as a basis the most recent analytical methods in the licensee's submittal of July 30, 1975.

The staff finds the MAPLHGR's for this fuel cycle acceptable for interim operation until such time as the Appendix K submittal is reviewed by the staff. The Appendix K reanalysis was submitted for staff review July 30, 1975.

7.0 Surveillance

Excessive instrument tube-channel interaction previously has been determined from the noise level in the LPRM signals. The plugged bypass flow holes are expected to affect the noise content of the LPRM signals. The noise content in the 1.4 to 3 Hz frequency range caused by vibration of the LRPM instrument tube should be reduced relative to the power dependent noise content. Some increase in the boiling noise, 5 to 50 Hz range, is expected because of boiling in the bypass water region.

Before the plant was shutdown in early August 1975, extensive LPRM time traces, TIP traces, and power spectral density (PSD) calculations were obtained for a number of combinations of power and flow. These data will provide a basis for evaluating the efficiency of plugging the bypass flow holes. After reactor startup, comparison of similar measurements with pre-shutdown data will be made to confirm that the mechanical vibration of the instrument tubes has been substantially reduced.

The licensee has agreed to provide NRC with a plan for monitoring instrument tube-channel box interaction. The monitoring will be performed on a periodic basis using the available LPRM and TIP traces.

Conclusion

Based on our evaluation of the safety analyses submitted by the licensee, we conclude that the Vermont Yankee Nuclear Power Station can be operated without undue risk to the health and safety of the public, provided the facility is operated in accordance with the restrictions in Appendix A to this safety evaluation.

AUG 2 2 1975

References

1. Letter from D. E. Vandenburgh, Vice President, Vermont Yankee Nuclear Power Corporation to NRR dated July 30, 1975 enclosing GE Safety Analysis with Bypass Flow Holes Plugged.

2. Letter from D. E. Vandenburgh, Vice President, Vermont Yankee Nuclear Power Corporation to NRR, dated August 6, 1975 enclosing LOCA analysis with MAPLHGR limits.

3. General Electric BWR Thermal Basis (GETAB): Data, Correlation and Design Application, NEDO-10958 (Nov. 1973).

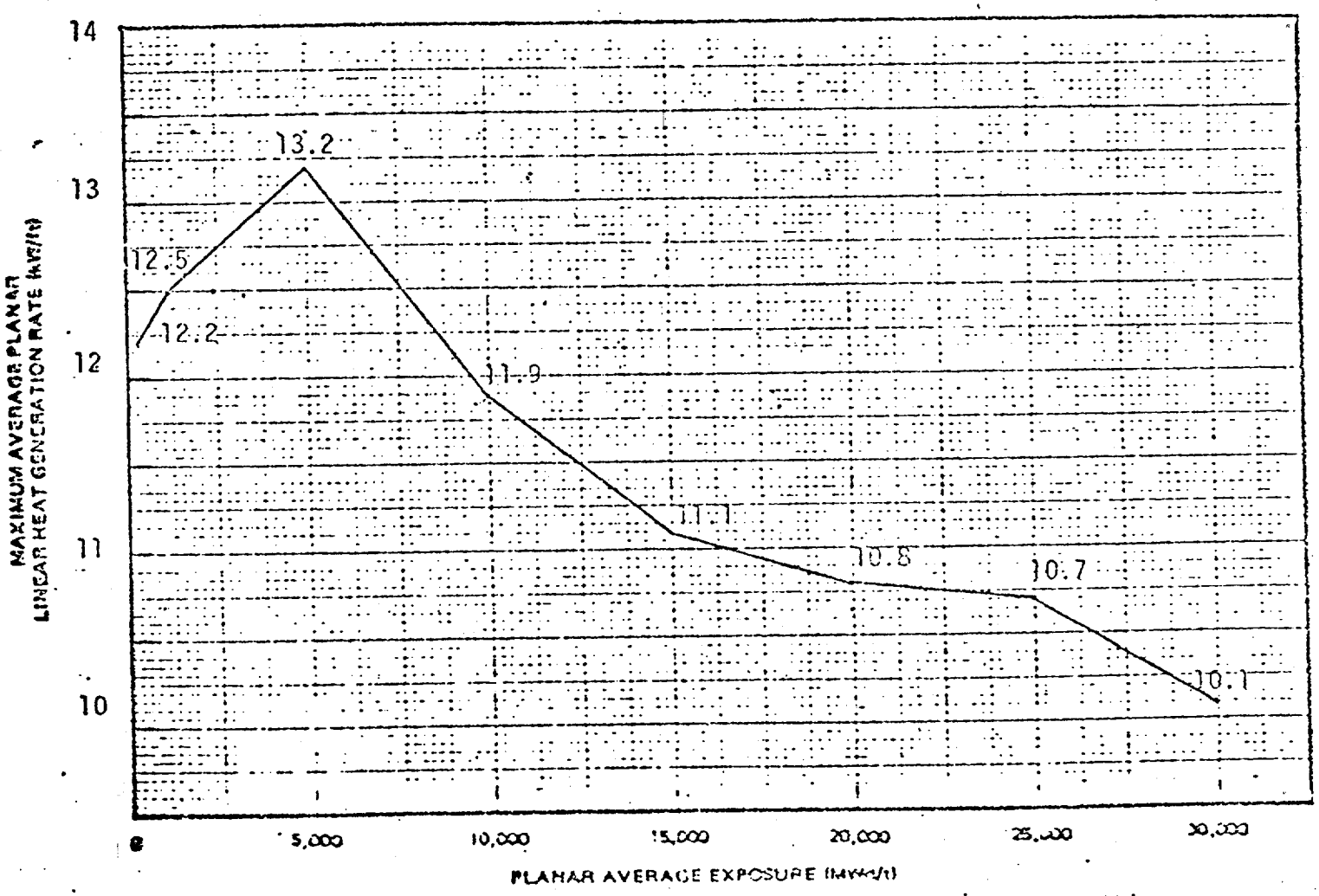
4. Review and Evaluation of GETAB for BWR's by Technical Review, Directorate of Licensing, U.S. AEC (September 1974).

ATTACHMENT

APPENDIX A

VERMONT YANKEE NUCLEAR POWER STATION
OPERATING RESTRICTIONS

There are two limitiations on the continued operation of the reactor for the remainder of this fuel cycle. These are the limiting assembly maximum average planar linear heat generation rate, MAPLHGR, and the minimum critical power ratio limit related to boiling crisis, MCPR. Operation shall conform to a MCPR value of equal to or greater than 1.28 as proposed by the licensee. The limiting value of MAPLHGR included with the proposed Technical Specifications submitted on October 31, 1974 have been revised to account for the staff requirements of December 27, 1974 and the proposed operation with plugged bypass holes. The revised values are given in Figures 1 and 2 for Generic B 7x7 Fuel and 8D219 8x8 Fuel types. A maximum value linear heat generation rate of 13.4 $kw/_{ft}$ for 8x8 fuel shall remain in effect until the Commission completes its review of the licensee's proposed increase in linear heat generation rate.



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS PLANAR AVERAGE EXPOSURE

Vermont Yankee Nuclear Power Station
ECCS Analysis Using December 28, 1974 Methods
Generic B 7x7 Fuel Type

Figure 1

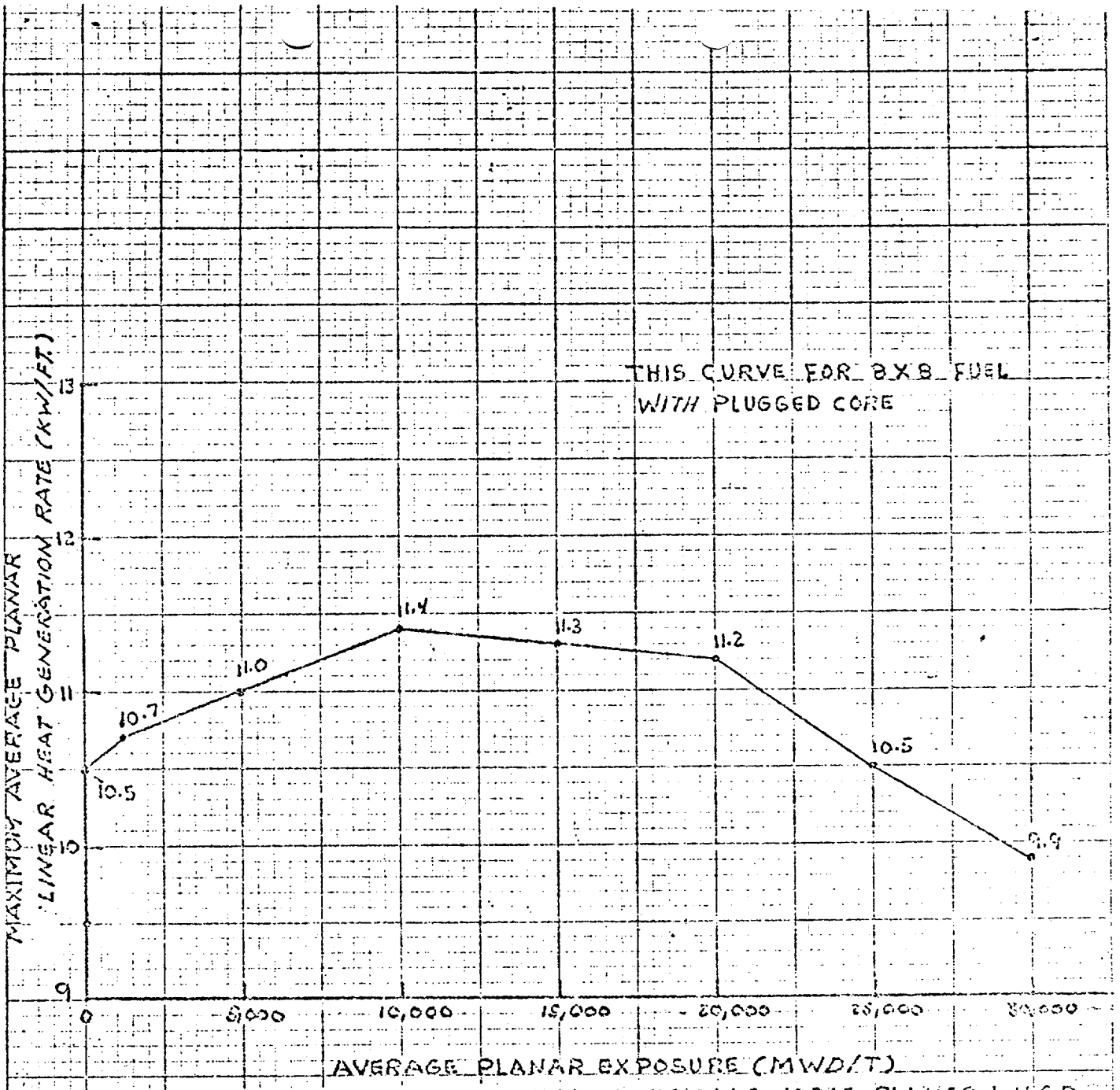


FIGURE 2 MAXIMUM ALLOWABLE PLANAR LHGR