Copies sent to applicant, PDR, local PDR, Cheney, Chalhoun, and Reisman 11-16-73.

NOV 16 1973

Docket No. 50-271

Vermont Yankee Nuclear Power Corporation
ATTN: Mr. Albert A. Cree, President
77 Grove Street
Rutland, Vermont 05701

Change No. 12
License No. DPR-28

Gentlemen:

Your letter dated November 6, 1973, proposed changes to the Technical Specifications of Facility License No. DPR-28 for the Vermont Yankee Nuclear Power Station that would increase the maximum average planar linear heat generation rate (MAPLHGR) for the initial core fuel with and without enrichment deviations and would add another curve for reload-1 fuel assemblies to Figure 3.5.1. These changes in the MAPLHGR are the result of modifications which were made to the core to preclude further wear between the fuel channels and temporary control curtains and reanalysis at a reactor thermal power level of 81 percent (or less) rated thermal power.

During our review, we informed your staff that certain modifications to the proposed changes were necessary to meet Regulatory requirements. These modifications have been made.

The MAPLHGR curves for the initial core fuel with and without enrichment deviations are based on computer calculations out to 30,000 MWd/t exposure as requested by our October 26, 1973 letter. The MAPLHGR curve for the reload-1 fuel assemblies are based on computer calculations with an assumed fuel planar average exposure of 10,000 MWd/t but have been extrapolated for higher exposures. Since the reload fuel assemblies will not be depleted to an average exposure greater than 10,000 MWd/t during the next operating cycle, the proposed curve is acceptable for this operating cycle. A revised MAPLHGR graph for the reload fuel assemblies, based on detailed heat up calculations at higher burnups, will be required for operation beyond this operating cycle.

cp.

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We have reviewed the "Summary Report on Vermont Yankee Channel Wear Investigation and Corrective Measures Taken" transmitted by your November 6, 1973 letter. We have concluded that plugging of the bypass flow holes in the lower core plate as described is an acceptable modification to the reactor vessel design described in the Vermont Yankee Final Safety Analysis Report and provides assurance that the observed fuel channel wear will not recur.

On the basis of our review, we have concluded that the proposed changes do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. Our related Safety Evaluation is enclosed.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to Facility License No. DPR-28 are hereby changed by replacing Figures 3.5.1 and 3.5.1A with the enclosed revised figures dated November 6, 1973, and by adding the enclosed Figure 3.5.1B.

Sincerely,

Original Signed by
D. J. Skovholt

Donald J. Skovholt
Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Figures 3.5.1, 3.5.1A and 3.5.1B

cc w/enclosures:
See next page

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UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

VERMONT YANKEE NUCLEAR POWER CORPORATION

DOCKET NO. 50-271

CHANGE NO. 12 TO THE TECHNICAL SPECIFICATIONS

Introduction

Following a shutdown of the Vermont Yankee Nuclear Power Station in September of this year, the Zircaloy channel boxes on some of the fuel assemblies were inspected and found to be damaged. The observations of damaged channels in the Vermont Yankee core and another reactor, the cause of the damage, and the consequences which might result have been discussed in the "Safety Evaluation by the Directorate of Licensing, U. S. Atomic Energy Commission, Relating to Channel Box Wear in the Vermont Yankee Nuclear Power Station and the Pilgrim Nuclear Power Station" dated October 26, 1973. Subsequently, the Vermont Yankee Nuclear Power Corporation has submitted a report entitled "Summary Report on Vermont Yankee Channel Wear Investigation and Corrective Measures Taken". This report describes the actions taken to repair the damage and prevent its recurrence. This Safety Evaluation sets forth the staff's reasons for concluding that the repairs and corrective measures which were made and are discussed in the Summary Report would permit the Vermont Yankee Nuclear Power Station to operate without undue hazard to the health and safety of the public.

Repair of Damage

All of the damaged fuel channels in the Vermont Yankee core have been replaced with new fuel channels. All of the fuel channels which are adjacent to a control curtain stiffener and therefore subject to possible damage by vibration of the curtain have been inspected. In addition, approximately 20% of the remaining fuel channels have been inspected. Based on these inspections, all channels which experienced wear of greater than 0.010 inch on the corners have been replaced with new fuel channels.

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Evaluation

The cause of channel wear, as identified in our previous Safety Evaluation, is the interaction of high velocity flow from the flow bypass holes with the temporary control curtains, causing the curtains to vibrate and damage adjacent channels. All of the observations and tests indicate that this is the sole cause of the observed channel damage in the Vermont Yankee core.

Of the many possible methods considered to prevent further damage, the most satisfactory would be to remove the source of the problem. In this core that would mean removal of either the control curtains themselves or blockage of the high velocity flow through the bypass holes. Although after approximately ten months of additional reactor operation the curtains are to be removed, immediate removal of the curtains is not practical at this time because of the reactivity consideration which would require design and installation of an entirely different core. The alternate solution of plugging the bypass holes in the lower core plate to eliminate the high velocity flow was determined to be the most practical, positive solution to the problem. This has been accomplished with a small plug whose design and analysis is discussed in a later section of this report.

Plugging the bypass flow holes in the Vermont Yankee core will reduce the bypass flow from approximately 10% of the total core flow to approximately 4%. Tests in the GE mockup facility show that flow through the plugged holes is insignificant, i.e., less than 0.3% of the total core flow. The remaining bypass flow results from several sources, but mainly by leakage between the channel and the lower tie plate and between the fuel support casting and the control rod guide. These sources of bypass flow are the same in the older BWRs which have no bypass flow holes. Therefore, the bypass flow patterns in the older reactors would be the same as expected in the Vermont Yankee core with the bypass holes plugged. Bypass flows in the old reactors are in the range of 4% to 10% of total core flow which is comparable with the expected flow in Vermont Yankee.

Approximately 5200 channels from 14 older BWR reactors have been examined and 51 channels from 4 of those reactors have been examined in greater detail. No damage attributable to control curtain vibration was observed on any of these channels. However, one crack of undetermined origin was observed.

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The core plate 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 load of 38 lbs on the body and latch and a maximum of 46 lbs (with the worst tolerance combination).

During installation the latch is in a position rotated 90 degrees from its installed position and is withdrawn into the body. The shaft is gripped by the installation tool and the plug is inserted into a bypass flow hole. First the body engages the rim of the hole and then the spring is compressed to push the shaft to its full extension. The latch then comes out of the body and rotates 90 degrees by means of an eccentric weight with respect to the pin. When the installation tool is relaxed, the latch bears against the bottom of the core plate. After insertion, the plug is pulled with about 30 lbs force to check its placement. At the end of the next fuel cycle (after approximately 10 months of service), the control curtains and bypass flow hole plugs will be removed. Removal of a plug will be accomplished by applying about 500 lbs 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. Actual plugs were latched on a 2-inch plate with 1-inch diameter holes.

Pressure differentials across the core plate during normal steady state operation and following a steam line break accident are expected to be 17 and 32 psi, respectively. These loads together with the spring preload will produce yielding on the latch in bending but will be significantly below the 500 lbs of force necessary for removing the plug. The 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 vibration was observed during the test and no apparent deformation on the latch was evident after the test. As previously mentioned, approximately 500 lbs 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 proposed 10 months 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 no 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 environment.

Effects on Operation

Plugging the bypass flow holes in the core support plate redistributes the bypass flow and increases the thermal margins slightly during normal operation, abnormal operational transients and accidents. However, in those regions of the core where the reduced bypass flow results in the formation of voids in the vicinity of neutron flux detectors, the uncertainty in the calculation of power is substantially increased. Therefore, plugging the bypass holes does not require any further restrictions on operating limits, but does require the development of appropriate procedures used to determine the variation of power within the core under the modified core conditions.

The effect of plugging the bypass flow holes is to increase the flow through the fuel assemblies and the void fraction in the bypass region. The increased flow will increase the critical heat flux while the increased voids will reduce the local power peaking and peak heat flux within the fuel assembly. During normal steady state operation and abnormal operational transients, both of these effects will increase the thermal margin, that is, the critical heat flux ratio.

Plugging the bypass flow holes does not affect any accident analysis other than the loss-of-coolant accident. In the design basis loss-of-coolant accident, redistribution of flow due to the plugging of the core plate bypass holes affects the course of the accident slightly. During the flow coastdown period of the blowdown, a portion of the flow which would have passed through bypass flow holes adds to the flow through the fuel assemblies and the remainder increases the reverse flow through the jet pumps of the broken recirculation loop. Although the increased flow through the fuel assemblies would increase the margin to critical heat flux, this effect will not result in any change in the calculated clad temperatures. However, the increased reverse flow through the jet pumps will decrease the net rate of drainage from the vessel-core shroud annulus, delay uncovering of the jet pumps, and extend the period of high heat removal rate. Stored energy removal is increased which tends to decrease peak clad temperatures.

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Although uncovering the jet pumps and the beginning of the period during which adiabatic heatup is assumed is delayed, the time when lower plenum flashing and the accompaning convective cooling begins is delayed even more. The slower draining of the annulus delays uncovering the recirculation loop suction nozzle as well as the uncovering of the jet pumps. Therefore, depressurization of the lower plenum is delayed. In addition, the pressure change required to reach the saturation pressure of the coolant in the plenum is greater because the core inlet subcooling is greater. The greater subcooling results from the higher core pressure drop resulting from the reduced core bypass area. The net effect of both delays is to increase the period during which adiabatic heatup occurs. This tends to result in higher calculated peak fuel clad temperatures.

During the period of lower plenum flashing and the accompanying convective cooling, a portion of the flow which would have passed through the bypass flow holes is redirected through the fuel assemblies. The increased flow increases heat transfer and tends to reduce the calculated peak fuel clad temperature.

The overall effect of plugging the bypass flow holes is to decrease the calculated peak clad temperature. The licensee has performed calculations which demonstrate that the maximum average planar linear heat generation rate resulting in a calculated peak clad temperature of 2300°F is 0.1 kW/ft greater without bypass flow than with bypass flow.

The major effect of plugging the bypass flow holes is the resultant increase in the uncertainty of measurements of power peaking factors. With plugged holes, the reduced bypass flow may result in the formation of steam voids in the bypass region where the neutron flux detectors are located. The presence of voids surrounding the detector perturbs the relation between the neutron flux detector signal and the power in the adjacent portions of the fuel assemblies. Voids reduce the fission rate in the detector which then results in an underprediction of the power in the adjacent fuel assemblies. The licensee has proposed that a correction factor be applied in the determination of the peak heat flux, minimum critical heat flux ratio, and maximum average planar linear heat generation rate. The correction factor consists of an algorithm to determine the bypass void fraction which is based on detailed thermal-hydraulic analysis of the bypass region and a constant determined from detailed physics calculation to relate bypass void fraction and detector response.

The proposed correction uses a value of 1.0 as an estimate of the ratio of the fractional change in fuel assembly local power as indicated by the neutron detector signal to the fractional change in the local bypass void fraction. GE has stated that the results of two dimensional, three group diffusion calculations for two different values of void fraction within a fuel assembly indicate that this ratio is between 0.4 and 0.5.

Based on the information available, we conclude that the use of a value of 1.0 for this ratio, which is twice the calculated value, would conservatively account for the uncertainties in calculating this ratio. A smaller value for this ratio may be suitable but cannot be determined unless additional analyses were available for our review.

The second element of the correction is the bypass void fraction which is to be calculated as a function of the various operating parameters, i.e., fuel assembly local power, control rod position, coolant bypass flow rate, inlet temperature and pressure. It is proposed that nominal values of these parameters be used in an algorithm which is based on detailed thermal-hydraulic calculations to determine the bypass void fraction. Some uncertainties are associated with each element of this method and the nominal value of each parameter. For example, GE has estimated that the uncertainty in the bypass flow is approximately $\pm$ 10%. In addition, the algorithm can reproduce the bypass void fraction calculated with the detailed thermal-hydraulic model within 0.01. Furthermore, the inaccuracy in the detailed thermal-hydraulic model is not taken into account. We conclude that the nominal calculated correction factor should be increased by 5% to account for these uncertainties. If further analyses and information are provided, review of this information might indicate that the uncertainties are smaller than currently estimated.

Conclusion

Based on the observation of the fuel channels and the control curtains at the Vermont Yankee Plant and flow tests performed at GE, we conclude that the observed damage was caused by flow induced vibration of the control curtains. We have concluded that such damage will be prevented from recurring in the Vermont Yankee reactor by plugging the bypass flow holes.

Based on a review of the design and installation methods of the bypass hole plugs, 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

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holes. We also have evaluated the effect that plugging the flow holes will have during normal operation, abnormal transients and accidents, and conclude that the only detrimental effect of the plugs is to increase the uncertainty in the measurement of power distribution. However, the correction factor used in the procedures proposed by the licensee, if increased by 5% to account for uncertainties in the input parameters, would adequately correct for the effects that bypass voids could have on the determination of local fuel assembly power.

We have evaluated the proposed changes to the maximum average planar linear heat generation rate (MAPLHGR) curves in the Technical Specifications. We have concluded that these MAPLHGR curves were determined using the Interim Acceptance Criteria calculational model, including the effects of fuel densification, and use of these MAPLHGR curves will not result in a peak clad temperature in excess of 2300°F during the loss-of-coolant design basis accident. Therefore, the Interim Acceptance Criteria will be satisfied during operation of the reactor in the proposed manner. The MAPLHGR curves are based on computer calculations out to 30,000 MWd/t exposure as requested by our October 26, 1973 letter for the initial core fuel with and without enrichment deviations. The MAPLHGR curve for the reload fuel is based on computer calculations out to 10,000 MWd/t exposure since the reload fuel will not be depleted to average exposure greater than this during the next operating cycle. Since the reanalysis to determine the MAPLHGR curves was based on a rated thermal power level of 81 percent or less, the proposed figures have been modified by adding a note to the figures in the Technical Specifications stating that the MAPLHGR curves must be reduced by 0.1 kW/ft if the stated power level is exceeded.

On the basis of our evaluation, we have concluded that the modification to the core by plugging the bypass holes in the lower grid plate does not present a significant hazards consideration and there is reasonable assurance that the health and safety of the public will not be endangered by operation of the reactor in the proposed manner. The Technical Specifications should therefore be changed as proposed by Vermont Yankee and modified by the AEC staff.

Date: November 16, 1973

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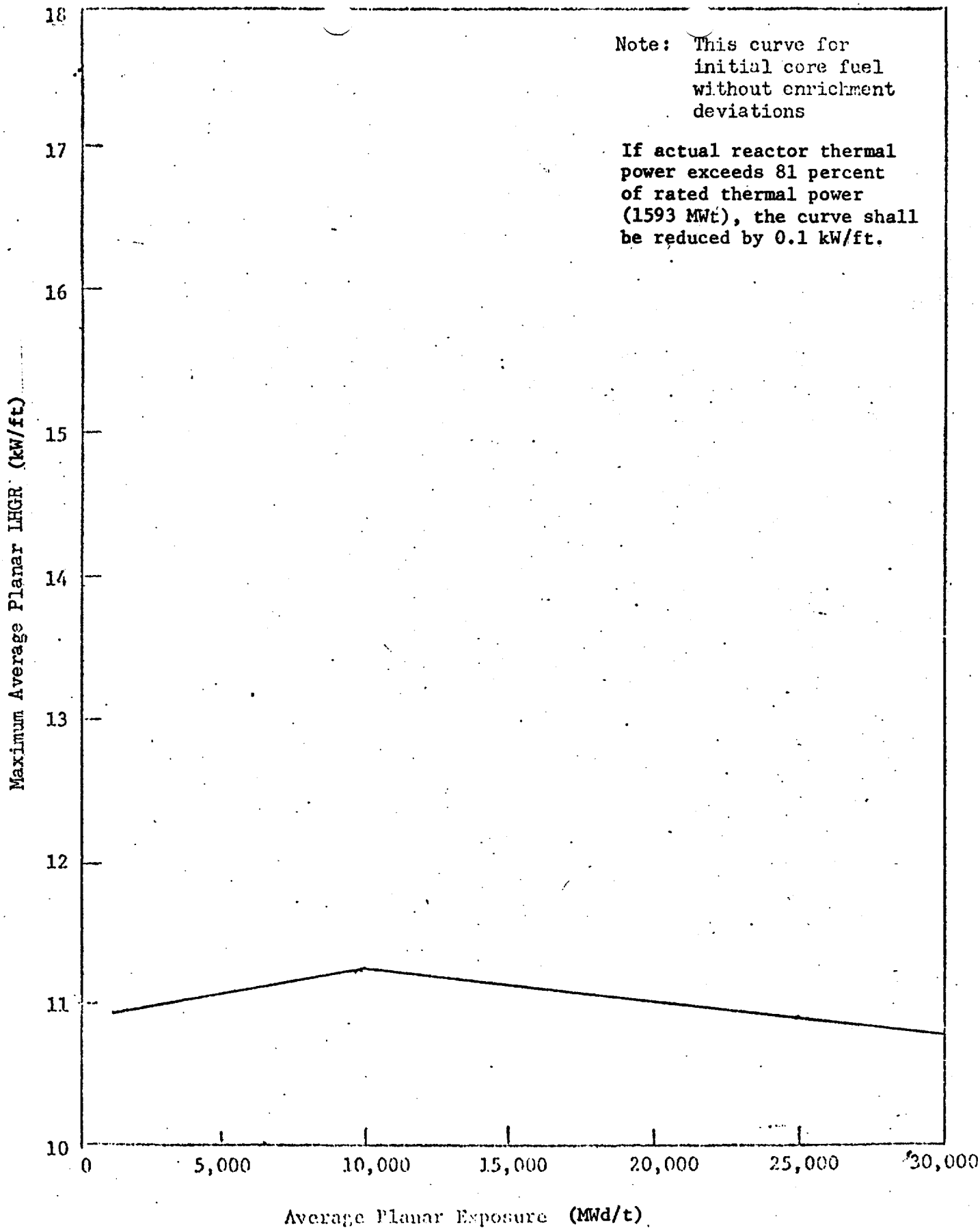

Figure 3.5.1 Maximum Allowable Planar LHGR

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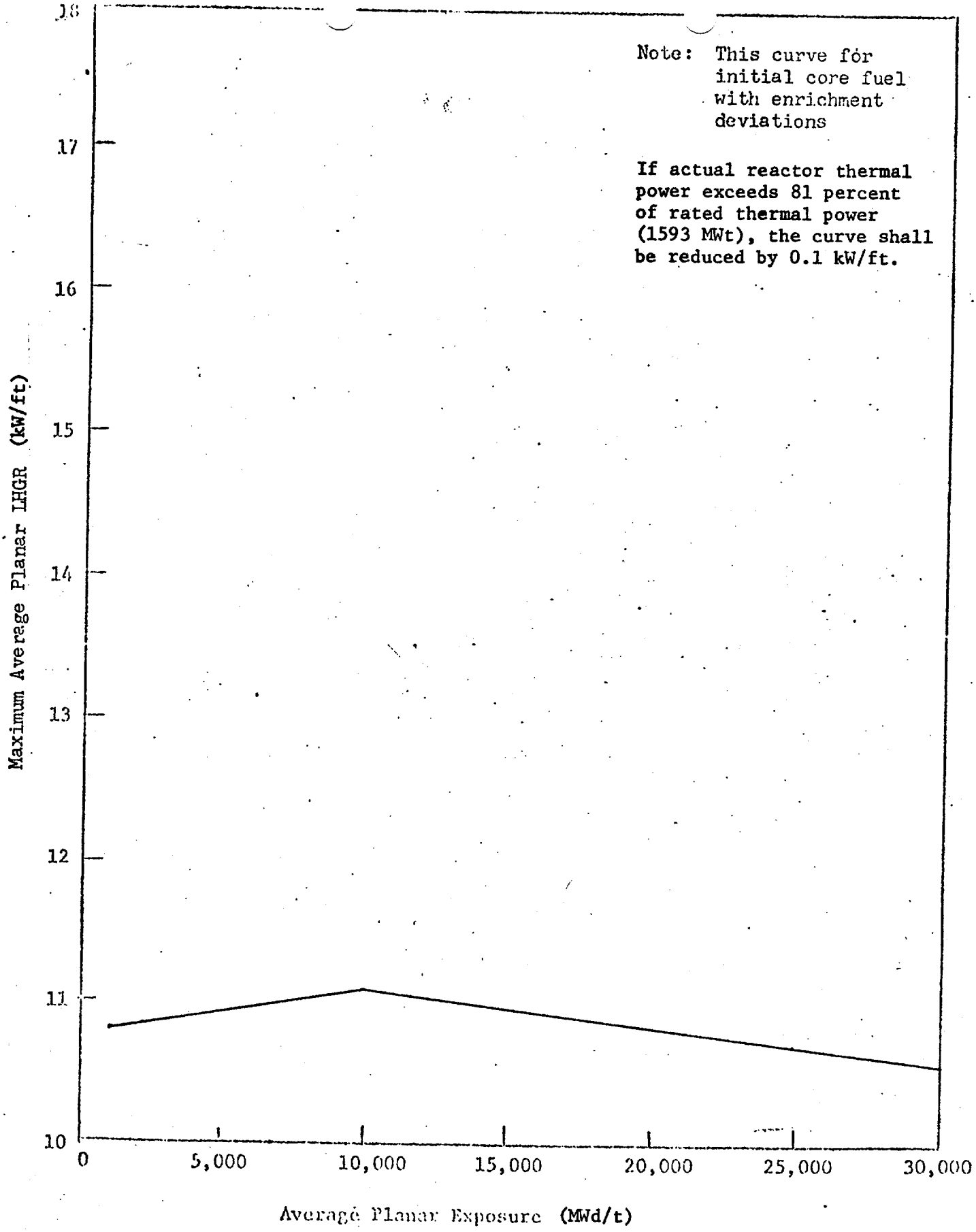

Figure 3.5.1A Maximum Allowable Planar LHGR

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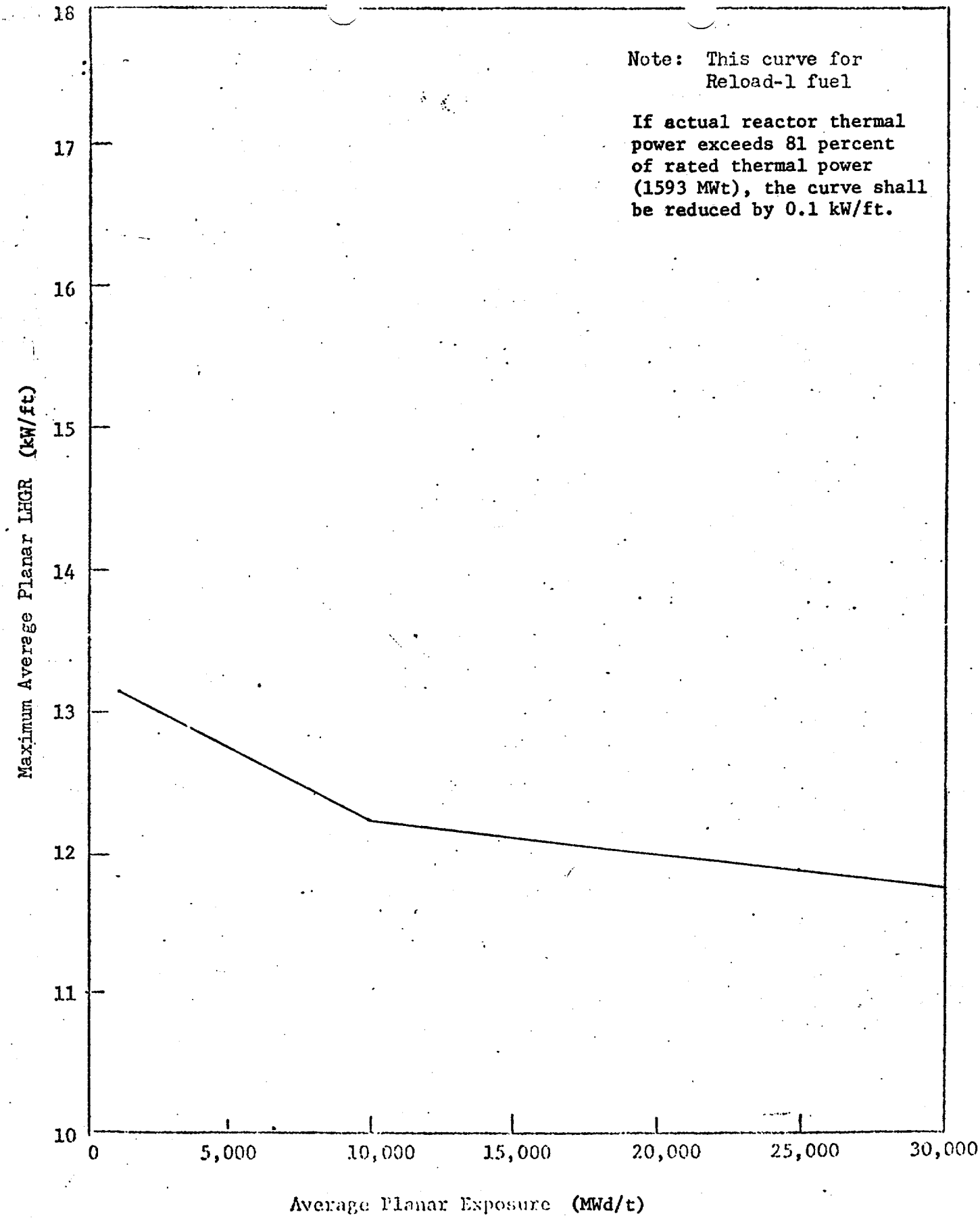

Figure 3.5.1.B Maximum Allowable Planar LHGR

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