Docket No. 50-331

6/30/75

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Iowa Electric Light and Power Company
ATTN: Mr. Duane Arnold, President
Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

Gentlemen:

Enclosed is a signed original of the "Order for Modification of License" issued by the Commission on June 30, 1975, for the Duane Arnold Energy Center. The Order revises, in its entirety, Appendix A to the Order for Modification of License dated December 27, 1975. However, all other provisions of that 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,

/S/

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

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

cc: See next page

DR 6-30

OFFICE	ORs	TR	OELD	RL	DD NRR	NRR
SURNAME	WPaulson:kmf GLear GL 6/30/75 KRGoller KRG	V. Stello	KKS	AGiambusso	EGCase	BCRusche BCR
DATE	6/30/75	6/30/75	6/30/75	6/30/75	6/30/75	6/30/75

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Iowa Electric Light & Power Company

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Mr. Dudley Henderson
Chairman, LinnCounty
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Environmental Protection Agency
Region VII Office
1735 Baltimore Avenue
Kansas City, Missouri 64108

Reference Service
Cedar Rapids Public Library
426 Third Avenue, S. E.
Cedar Rapids, Iowa 52401

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

June 30, 1975

Docket No. 50-331

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Security Building
P. O. Box 351
Cedar Rapids, Iowa 52406

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A copy of the Safety Evaluation is also enclosed.

Sincerely,

George Lear

George Lear, Chief
Operating Reactors Branch #3
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2. Safety Evaluation

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of IOWA ELECTRIC LIGHT AND POWER COMPANY, CENTRAL IOWA POWER COOPERATIVE, AND CORN BELT POWER COOPERATIVE (Duane Arnold Energy Center)	)))))))	Docket No. 50-331

ORDER FOR MODIFICATION OF LICENSE

I.

Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (licensees) are the holders of Facility Operating License No. DPR-49 which authorizes operation of the Duane Arnold Energy Center (the facility) at steady-state reactor core power levels not in excess of 1658 megawatts thermal (rated power). The facility is a boiling water reactor (BWR) located at the licensees' site near Palo in Linn County, Iowa.

II.

1. On May 21, 1975, the Nuclear Regulatory Commission (NRC) issued an Order for Modification of License[1/] restricting facility operation to core power levels not exceeding 50% of rated core power and core flow rates not exceeding 50% of design flow rate. As discussed in the May 21, 1975 Order, this action was taken as a result of indications of possible damage to fuel element channel boxes.

1/ See Order for Modification of License, In the Matter of Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (Duane Arnold Energy Center), Docket No. 50-331 dated May 21, 1975 (40 F.R. 23782, June 2, 1975).

The reduction in power and core flow were designed to reduce flow through core plate bypass holes sufficiently to reduce excessive vibration of the instrument thimbles in the bypass region. This, in turn, would reduce further channel box damage.

2. After discussion with the NRC staff on May 29, 1975, the licensees agreed to undertake a program of test, inspection and, if necessary, repair. The licensees agreed to operate the facility at full power for test purposes for a limited 72-hour period, to shutdown the facility immediately thereafter, to remove fuel elements from the core and to inspect the channel boxes for damage. Depending on the results of the inspection, the licensees agreed to make appropriate repairs, including plugging of the bypass flow holes and to submit safety analyses assessing the return to power operation with plugged bypass holes and any other changes made as a result of the inspection. The plant would resume power operation only after review of the safety analyses assessing operation with plugged bypass holes and authorization by the NRC.

3. Upon completion of the program of tests approved by the NRC staff's letter dated June 2, 1975, the reactor was shut down on June 6, 1975 and visual inspection of the channel boxes was performed. Inspection of the first four channel boxes showed unacceptable wear in the corners of the channel boxes adjacent to the instrument thimble. As a result of these observations, the licensees by letter of June 13, 1975 to the NRC staff, requested authorization to install core bypass flow plugs in the lower core plate as described in the enclosure to the licensees'

letter of June 6, 1975 to the NRC staff, and supplied analyses to demonstrate the adequacy of such plugs and the adequacy of the procedures for plug installation.

4. On June 18, 1975, the NRC issued an Order that, consistent with the understanding described in paragraph 3, authorized the installation of bypass hole plugs in the lower core plate. As discussed in the June 18, 1975 Order, the NRC staff concluded that the plugs will reduce the vibration of the instrument thimbles caused by flow through the bypass holes. The June 18 Order also added a condition to license DPR-49 that stated that the reactor shall not operate without authorization by the Office of Nuclear Reactor Regulation.

5. By letters dated June 10, 1975, June 16, 1975, and June 24, 1975, the licensees submitted analyses, including an emergency core cooling performance analysis, for reactor power operation with the plugs installed in the bypass holes. In its letter dated June 25, 1975, the licensees requested authorization to operate the reactor with plugs installed in bypass flow holes.

6. The NRC staff has reviewed the analyses submitted by the licensees on June 10, 16, and 24, 1975, to support operation with bypass flow plugs installed. As discussed in the NRC's Safety Evaluation, Duane Arnold Energy Center Operation with Plugged Bypass Flow Holes, dated June 30, 1975, 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[2/] 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 Duane Arnold 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. After ten months of successful service, the plugs in the Vermont Yankee reactor were removed at the time that the temporary curtains were removed.

7. Based on a review of the licensees' submittals of June 10, 16, and 24, 1975, and the prior related experience at the Pilgrim and Vermont Yankee reactors, the NRC staff concluded in its June 30, 1975 Safety Evaluation that operation of the Duane Arnold 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.

8. Copies of the following documents are available for public inspection in the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., 20555 and are being placed in the Commission's

2/ See Order for Modification of License, <u>In the Matter of Iowa Electric Light and Power Company</u>, (Duane Arnold Energy Center), Docket No. 50-331, dated December 27, 1974 (40 F.R. 1763 , January 9, 1975).

Local Public Document Room, Reference Service, Cedar Rapids Public Library, 426 Third Avenue, S. E., Cedar Rapids, Iowa: (1) the licensees' letters of June 6, 1975, June 10, 1975, June 13, 1975, June 16, 1975, June 24, 1975, and June 25, 1975; (2) the NRC letter of June 2, 1975 and the NRC staff Safety Evaluation of Duane Arnold Energy Center Operation with Plugged Bypass Flow Holes dated June 30, 1975, and the documents referenced therein.

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 dated June 30, 1975. All other provisions of the December 27, 1974 Order shall remain in full force and effect.
2. Operation of the Duane Arnold Energy Center 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
30th day of June, 1975.

Attachment:
DAEC Operating Restrictions

ATTACHMENT

APPENDIX A

DUANE ARNOLD ENERGY CENTER
OPERATING RESTRICTIONS

There are two limitations on the continued operation of the reactor for the remainder of Cycle 1. 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 1.34 as proposed by the licensee. The limiting value of MAPLHGR included with the proposed Technical Specifications submitted on August 9, 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 A-1 and A-2 for fuel types 1, 2, and 3. The limiting MAPLHGR for the four replacement fuel assemblies is 9.0 kw/ft.

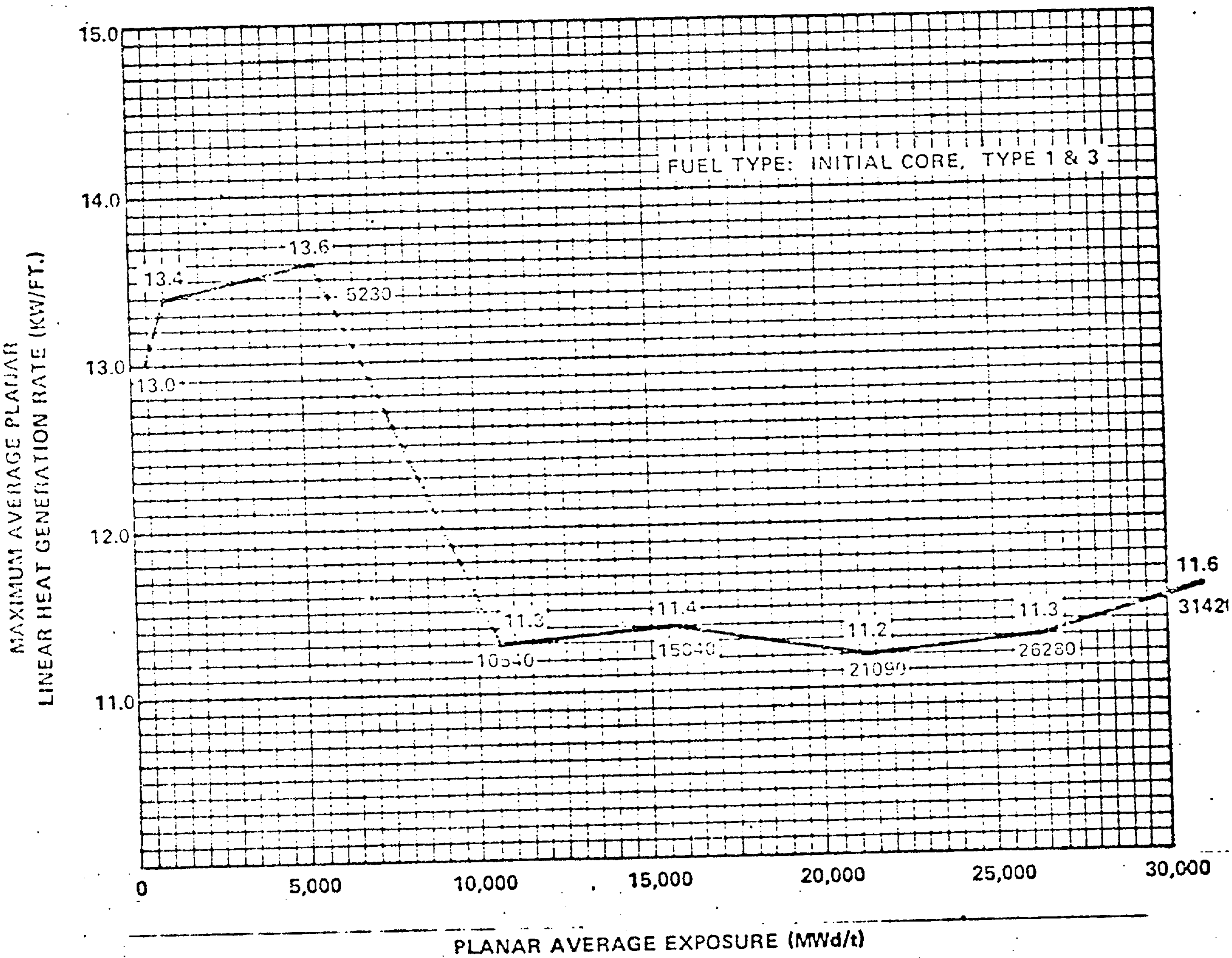

FIGURE A-1
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE
VERSUS PLANAR AVERAGE EXPOSURE

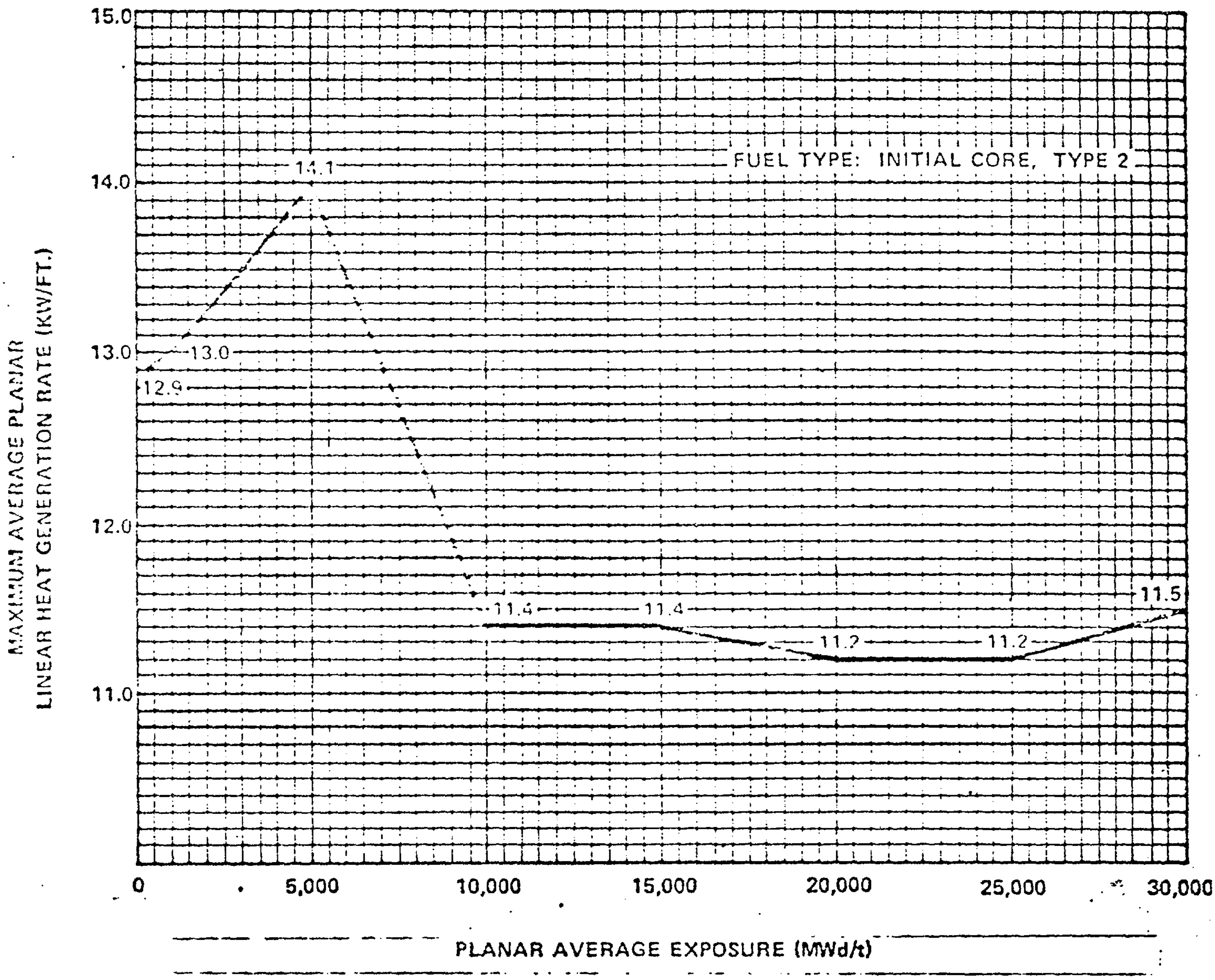

FIGURE A-2
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE VERSUS PLANAR AVERAGE EXPOSURE

Docket No. 50-331 June 30, 1975

SAFETY EVALUATION REPORT
DUANE ARNOLD ENERGY CENTER OPERATION
WITH PLUGGED BYPASS FLOW HOLES

1. Introduction

Iowa Electric Light and Power Company submitted References 1 through 4 to the NRC in support of its license amendment to continue operation of the Duane Arnold power plant for the remainder of cycle 1. The principal changes are the plugging of the bypass flow holes in the core support plate in order to reduce instrument tube-fuel channel interaction and the use of four replacement fuel assemblies.

2. Summary

Based on this review, the NRC staff has drawn the following conclusions regarding the proposed operation of the Duane Arnold Energy Center 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 1.34.

e. The MAPLHGR limits, based on previously approved restrictions in the AEC's order of December 27, 1974, are acceptable.

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

Operating restrictions for the remainder of cycle 1 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 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 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 4.1% at the core midplane to 5.0% at the core exit.

After the bypass flow holes are plugged, most of the fuel will be placed in its original core location. Four replacement 7x7 fuel bundles will be placed on the periphery and 36 bundles will be moved in the reactor to maintain quadrant symmetry. Some fuel shuffling is necessary to replace fuel damaged by a dropped fuel bundle. Rod patterns and withdrawal sequences will not be changed. 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 fuel storage margins are unaffected, and

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

We have reviewed the proposed core configuration and find it to be a minor change from the orginal 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 lbs on the body and latch and a maximum of 46 lbs (with the worst tolerance combination).

Removal of a plug can 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.

Plugs identical to those to be used in the Duane Arnold reactor were installed in both the Vermont Yankee and Pilgrim reactors. The plugs 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 on the latch in bending but will be significantly below the 500 lbs 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 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 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.

General Electric presented to the NRC staff a summary of channel inspections on BWR-2s and BWR-3s (5) . These older plants have instrument tubes similar to Duane Arnold, 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 Duane Arnold configuration with plugged bypass holes. Sixty-four channels (adjacent to instrument tubes and source tubes) were inspected during normal fuel outages in 5 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 meet this requirement.

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.

The mechanical design of the four replacement assemblies is essentially the same as the existing fuel and is acceptable to the staff.

5.0 Thermal-Hydraulic Design

The fuel cladding integrity safety limit for Duane Arnold has been changed to a minimum critical power ratio (MCPR) based on a thermal margin correlation, GEXL (6), which the staff previously has found acceptable (7). The fuel cladding integrity safety limit MCPR for the 7x7 cycle 1 fuel 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 6.3% for the TIP readings plus a 4.7% standard deviation due to voids in the bypass region.

Conservatism was applied to the axial power shape because the 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 1.34 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 per cent. 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 -37.4 dollars. This is more scram worth than the previously determined value of approximately -30 dollars and is due only to a recalculation of the Duane Arnold 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 Duane Arnold 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 analysis was initiated from 104 percent design power and the scram was initiated by the position switch on the turbine stop valves. A peak pressure of 1240 psig was calculated at the bottom of the vessel. The decrease in MCPR is 0.28 which is the limiting change in thermal margin. As a result, the steady-state MCPR must be equal or greater than 1.34 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 0.15.

The licensee also presented an overpressure analysis to show compliance with Section III of the ASME Boiler and Pressure Vessel Code. The limiting transient presented was closure of all main steamline isolation valves with an indirect scram. The analysis was performed at 104% power, no credit for relief valves, scram initiated by high neutron flux, and a failure of a single safety valve. A pressure of 1311 psig was calculated at the bottom of the vessel, which is below the ASME Code limit of 1375 psig.

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 107% 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 by-pass 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 fuel types 1, 2, and 3 (2). This allows operation of the Duane Arnold Energy Center at rated power; the Appendix K reanalysis will be submitted for staff review prior to July 9, 1975.

The calculation was performed using procedures described by General Electric in their December 13, 1974 letter from G. Gyorey to V. Stello, NRC. The licensee applied a MAPLHGR penalty or re-duction to their August 9, 1974 submittal for the longer delayed flooding time which occurs when the bypass holes are plugged. The August 9, 1974 submittal assumed a reflood time of 139 seconds, while the June 10th, 1975 analysis assumed a flooding time of 388 seconds.

The staff revisions to the GE-ECCS evaluation model (to account for delayed flooding due to steam updraft) delayed reflood from 139 seconds to 233 seconds. This delay in reflood time in addition to the other staff requirements reduced the MAPLHGR by approximately 12.3% (fuel type 1 and 3 at 5,280 MWD/T) and 9% (fuel type 2 at 5000 MWD/T). The June 10, 1975 calculation, with a reflood time of 388 seconds, in addition to other staff requirements of December 27, 1974 reduces the August 9, 1974 MAPLHGR by approximately 16.3% (fuel type 1 and 3 at 5,280 MWD/T) and 14% (fuel type at 5000 MWD/T).

The staff finds the MAPLHGR's for fuel types 1, 2, and 3 acceptable for interim operation until such time as the Appendix K submittal is reviewed by the staff.

The licensee presented the results of a sensitivity study (4) that showed the calculated MAPLHGR's for the four replacement fuel assemblies are as much as 12 percent lower than similarly calculated MAPLHGR's for fuel type 2. Using the acceptable MAPLHGR's for fuel type 2 (Fig. A-2) and a 12 percent reduction factor, the staff calculated an equivalent minimum MAPLHGR of 9.9 kw/ft for the four replacement fuel assemblies at a burnup of 20,000 MWd/t. As an added conservatism, the staff reduced this value by 0.9 kw/ft to derive an acceptable MAPLHGR of 9.0 kw/ft for the four replacement assemblies. It is not expected that these assemblies will be limiting because they are located at the periphery of the core where the assembly power is low. This MAPLHGR limit (9.0 kw/ft) is acceptable for interim operation until such time as the Appendix K submittal is reviewed by the staff.

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 LPRM 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 June 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 and accelerometers on the LPRM guide tubes.

Conclusion

Based on our evaluation of the safety analyses submitted by the licensee, we conclude that the Duane Arnold Energy Center 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.

References

1. Letter from R. Lowenstein (attorney for IELPC) to B. Rusche (NRC) dated June 6, 1975.

2. Letter from K. Shea (attorney for IELPC) to B. Rusche (NRC) dated June 10, 1975.

3. Letter from C. Sanford (IELPC) to B. Rusche (NRC) dated June 16, 1975.

4. Letter from J. Bouknight, Jr. (attorney for IELPC) to B. Rusche (NRC) dated June 24, 1975.

5. Meeting at NRC, April 24, 1975.

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

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

Conclusion

Based on our evaluation of the safety analyses submitted by the licensee, we conclude that the Duane Arnold Energy Center 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.

References

1. Letter from R. Lowenstein (attorney for IELPC) to B. Rusche (NRC) dated June 6, 1975.

2. Letter from K. Shea (attorney for IELPC) to B. Rusche (NRC) dated June 10, 1975.

3. Letter from C. Sanford (IELPC) to B. Rusche (NRC) dated June 16, 1975.

4. Letter from J. Bouknight, Jr. (attorney for IELPC) to B. Rusche (NRC) dated June 24, 1975.

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

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

Conclusion

Based on our evaluation of the safety analyses submitted by the licensee, we conclude that the Duane Arnold Energy Center 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.

References

1. Letter from R. Lowenstein (attorney for IELPC) to B. Rusche (NRC) dated June 6, 1975.
2. Letter from K. Shea (attorney for IELPC) to B. Rusche (NRC) dated June 10, 1975.
3. Letter from C. Sanford (IELPC) to B. Rusche (NRC) dated June 16, 1975.
4. Letter from J. Bouknight, Jr. (attorney for IELPC) to B. Rusche (NRC) dated June 24, 1975.
5. Meeting at NRC, April 24, 1975.
6. General Electric BWR Thermal Basis (GETAB): Data, Correlation and Design Application, NEDO-10958 (Nov. 1973).
7. "Review and Evaluation of GETAB for BWR's by Technical Review, Directorate of Licensing, U.S.AEC (Sept. 1974).

Conclusion

Based on our evaluation of the safety analyses submitted by the licensee, we conclude that the Duane Arnold Energy Center 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.

References

1. Letter from R. Lowenstein (attorney for IELPC) to B. Rusche (NRC) dated June 6, 1975.

2. Letter from K. Shea (attorney for IELPC) to B. Rusche (NRC) dated June 10, 1975.

3. Letter from C. Sanford (IELPC) to B. Rusche (NRC) dated June 16, 1975.

4. Letter from J. Bouknight, Jr. (attorney for IELPC) to B. Rusche (NRC) dated June 24, 1975.

5. Meeting at NRC, April 24, 1975.

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

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

APPENDIX A

DUANE ARNOLD ENERGY CENTER OPERATING RESTRICTIONS

There are two limitations on the continued operation of the reactor for the remainder of Cycle 1. 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 1.34 as proposed by the licensee. The limiting value of MAPLHGR included with the proposed Technical Specifications submitted on August 9, 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 A-1 and A-2 for fuel types 1, 2, and 3. The limiting MAPLHGR for the four replacement fuel assemblies is 9.0 kw/ft.

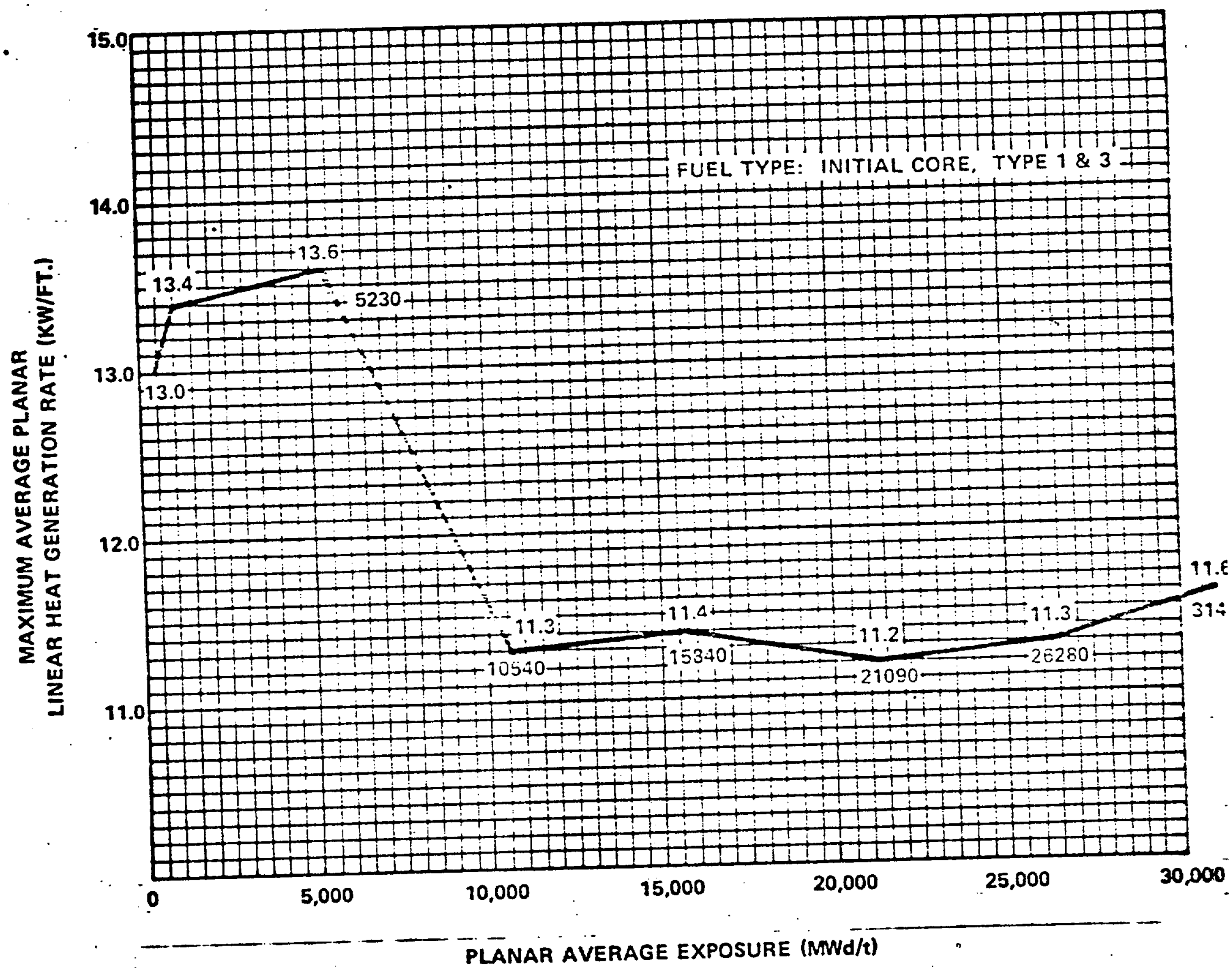

FIGURE A-1
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE
VERSUS PLANAR AVERAGE EXPOSURE

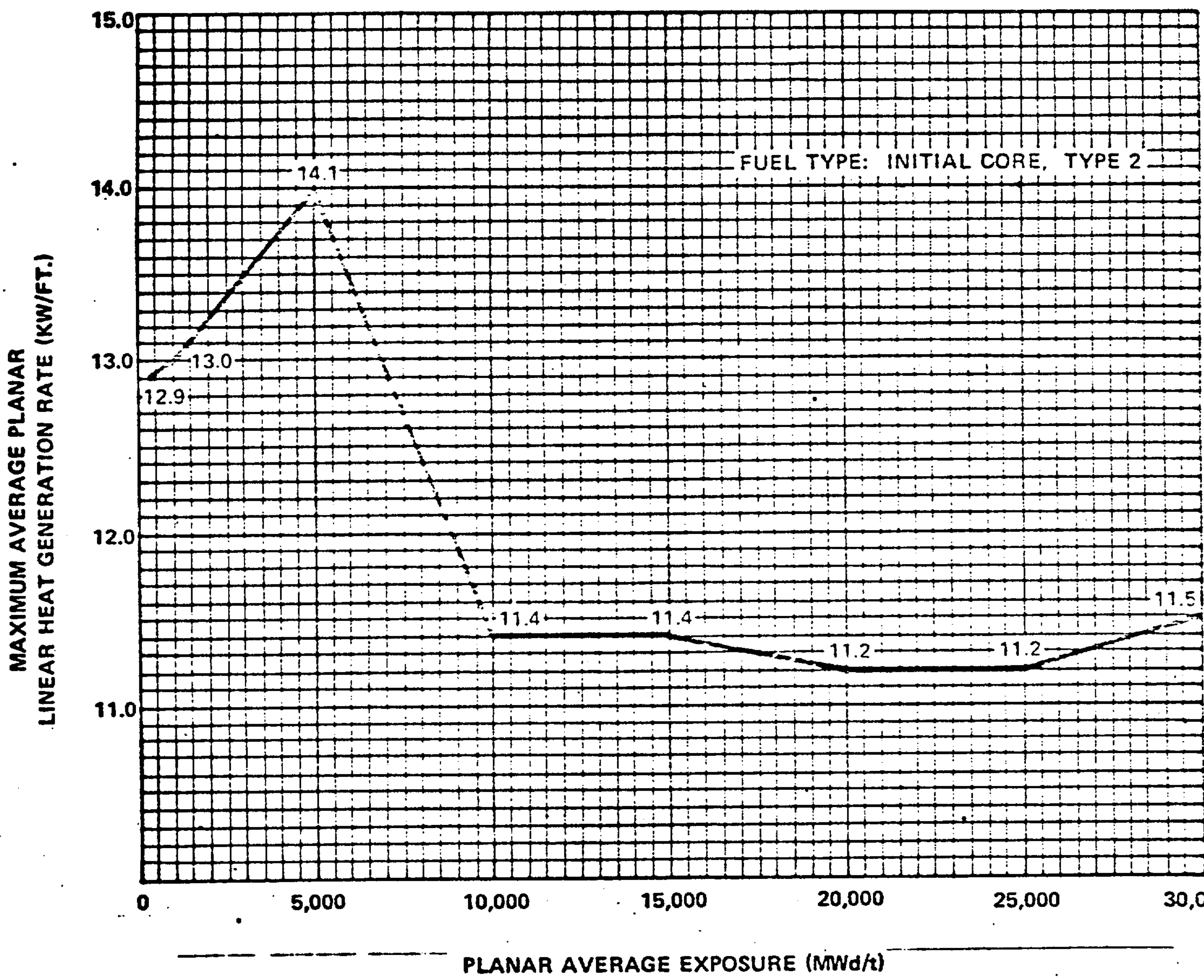

FIGURE A-2
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE VERSUS PLANAR AVERAGE EXPOSURE