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CORE PARAMETER STUDY

FOR

A 300-MW SODIUM GRAPHITE REACTOR

NAA-SR-MEMO 4486

By

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510 001

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INTRODUCTION

A primary objective in the design of a nuclear power reactor is achievement of minimum power generating costs consistent with reactor safety and maintenance requirements. Cost optimization studies are generally based upon a fixed technology with the realization that the resultant design may not be optimum when research and development work is completed.

The sodium graphite reactor of "current status" (NAA-SR Memo 4156) utilizes uranium-molybdenum alloy as fuel. It is expected that uranium carbide or other high performance fuel will prove more suitable after research and development work is completed. It should be clearly noted that the optimum design with U-Mo fuel would differ considerably from the optimum with UC.

As in other nuclear reactors, the fuel cycle costs are a significant part of the power generating costs. In an SGR core parameter study the important variations in power generating costs are those associated with the fuel cycle. With "thin" wall core vessels and inexpensive moderator and coolant, the variation in capital costs brought about by core modifications is not a major determinant of optimum core design. Certain other characteristics of the materials and components which make up an SGR ease the problem of core optimization. The sodium coolant is an excellent heat transfer medium. Current designs permit a large variation in heat flux with little concern for burnout heat flux or fuel element cladding temperature limitations. A variation in amount of coolant in the core does not significantly affect the moderating properties of the core.

With the fuel material fixed, (U-Mo alloy in this study), the major core parameter which is of significance in an SGR is the fuel-to-moderator volume ratio. A variation in this parameter can be achieved through a variation in lattice spacing or fuel rod diameter. Also of interest is the effect of core structural material and coolant on power generating costs.

The parameters investigated in this study are:

(1) Lattice spacing (or moderator element size)

(2) Fuel rod diameter (fixed length of 15 ft)

(3) Fuel cladding thickness

(4) Moderator canning thickness

(5) Zirconium versus stainless steel moderator canning

(6) Sodium flow area

The fuel with the greatest potential for SGR's, uranium monocarbide, is also investigated.

This list of variables differs from the list suggested by AEC. The AEC list is repeated below with comments.

(1) Moderator to Fuel Volume Ratio - This variable was considered in the survey both by variation in fuel element spacing (moderator) element size) with fixed fuel element geometry and by variation of fuel rod diameter in a fixed element spacing.

(2) Enrichment - In this study the enrichment is a dependent variable which changes with fuel element spacing, fuel rod diameter and burn up.

(3) Operating Critical Mass - The mass (U-235) is a dependent variable which varies with required enrichment and core inventory. These in turn depend upon rod size, element spacing, and burn up.

(4) Excess Reactivity - This item was varied in the study only for the reference core in which the effect of burnup in costs was considered. The excess reactivity required for higher burnup was obtained by increasing enrichment.

(5) Conversion Ratio - This is a dependent variable which changes as rod diameter, lattice spacing and moderator cladding thickness (or material) is varied.

(6) Maximum to Average Flux and Power Ratios - This quantity was not varied in the study. Improvement in the peak/average ratio is perhaps best affected by varying the fuel element spacing, which is impractical in the canned moderator SGR. The important change to be achieved by "flattening" in the SGR is a reduction in fuel inventory charges, which are not excessive in this reactor. Possible improvements brought about by flattening with a large number of control rods would be minor. Improvement made possible through variation in the slowing down power of the moderator, by using graphite blocks of graduated density or by inserting a better moderating material such as BeO near the surfaces of the core, could be quite effective in reducing inventory charges. This was not investigated because of the short time schedule involved.

The important reduction in power cost achieved through flattening of the flux would be obtained by permitting the reactor power to rise as the power of each element approaches that of the central fuel channel. However, an honest appraisal of the effects of power flattening must be made with the power level fixed.

(7) Power Density (MWT/ft^3 of core) - This quantity is a dependent variable in the SGR. It is determined by rod diameter, lattice spacing and permissible power per rod (fixed in the reference design by the central fuel temperature).

(8) Specific Power (MWT/MT of fuel) - The specific power (kw/kg-U) is almost independent of all variables in the SGR except the limit on fuel center temperature and fuel rod diameter. With the center temperature fixed the obtainable power rod is essentially independent of rod diameter. The number of rods per element and the fuel

enrichment have a minor effect on this variable because of the flux depression in the element. The specific power in the U-235 (kw/kg-25) varies inversely with the fuel enrichment.

(9) Hot Channel Factors - A constant hot channel factor was used in this survey. There is no problem of severe local flux peaking because of the long neutron diffusion length in an SGR. Variation in the fuel-to-moderator ratio has a negligible effect on the hot channel factor. The important part of this factor is the disadvantage factor in the fuel element. Additional contributing items are manufacturing tolerances, variations in U-235 content, flow variations. The major cost item which would be effected by a change in the hot channel factor is the fuel inventory.

The results of the parameter survey are presented in a series of curves. It is important to note that the results are all closely interdependent and that caution must be used in considering the effects of any single set of curves.

It should also be noted that the curves obtained from the study apply to the reference reactor (300 MWe canned moderator design with U-10 w/o Mo fuel). While the same general trends would be obtained from a study of any fixed design, the points in the variables at which minimum cost is obtained would probably differ to a considerable degree. The major cost reductions anticipated for sodium graphite reactors will come about as a result of improved fuel elements and simplified systems and components, which will result from the SGR research and development program.

A. Summary of Results

The SGR Status Report (NAA-SR-Memo 4156) and section B of this report describes a 300 MW SGR of current design. This design is a scale-up of the Hallam Nuclear Power Facility. The design was based on the heat transfer requirements with the rod diameter, number of rods per element and lattice spacing selected by judgment based on past experience. In spite of this hurried fixing of the design, required for the cost analysis in the Status Report and the SGR Development Program Report (NAA-SR-Memo 4199), the results of this physics parameter survey indicates that modifications in the core design could reduce the estimated power costs by less than 5 percent.

The power generating costs determined for this reference design are as follows:

Average Fuel Burnup	3,400 MWD/MTU	11,000 MWD/MTU	16,800 MWD/MTU
Capital Charges	5.60	5.60	5.60
Fuel Cycle Costs	7.43	4.01	3.55
Operation & Maintenance	.99	.99	.99
Total Power Generating Costs	14.02	10.60	10.14

Optimizing the fuel to moderator ratio can reduce power generating costs slightly (approximately 0.1 mill per kilowatt hour). This study indicates that the fuel to moderator ratio is not critical. Both lattice spacing and fuel rod diameter can be varied within a relatively broad range without significant changes in power generation costs. (See Section C.1 and C.2)

The influence of the structural material in the core (Section C.3, C.4, and C.5) is indicated by the 0.3 to 0.4 mill per kw hr increase in power generating costs with an increase in fuel cladding thickness from 0.010", 0.020" and the 0.3 mill per kw hr increase in power generating costs with an increase in moderator canning thickness from 0.016" to 0.024" stainless steel. Fuel cycle costs could be reduced 0.3 mill per kw hr if the 0.016" stainless steel canning could be replaced with 0.035" zirconium alloy canning. Capitalizing this saving in fuel costs results to a maximum permissible cost of zirconium cladding of \$60 per pound. It is probable that the zirconium could be obtained at a somewhat lower cost.

The sodium flow area in the fuel element channel can be varied over a relatively wide range (6 in^2 to 12 in^2) with no significant change in power generating costs (less than 0.1 mill per kw hr - Section C.6).

The advantages of a high performance fuel such as UC can not be adequately evaluated in study primarily limited to the reactor core. Full advantage of UC is taken in the design of the 255 MW and 330 MW ASGR (NAA-SR-Memo 4199, "SGR Development Program - Objectives and Estimated Costs).

With no changes in conditions external to the core and core structure the use of UC will reduce capital costs approximately \$550,000 and reduce fuel inventory charges approximately 0.3 mill per kw hr. (See Section C.7)

B. Description of Reference Design and Economic Ground Rules

B.1 Description of 300 MW SGR

The reactor chosen for the core parameter survey was the 300 MW SGR described in NAA-SR Memo 4156, "Status Report for Sodium Graphite Reactors." It is a scale-up of the Sodium Graphite Reactor now being constructed at Hallam, Nebraska.

The reactor core, shown in Figure 1, consists of slightly enriched, uranium-molybdenum fuel elements suspended in a closely packed array of canned hexagonal graphite moderator elements. The moderator elements shown in Figure 2, are scalloped at the corners so that each three adjoining elements for a circular channel between them which runs axially through the core. Concentric with this channel is the process tube, one end of which penetrates the lower grid plate, the other end terminates at the fuel element-hanger rod disconnect. The fuel elements are supported in these tubes. The main sodium flow is upward through the process tubes. The control rod thimbles are supported and sealed at the reactor loading face shield and extend downward to the bottom of the core.

Sodium inlet pipes enter the reactor cavity just above the top of the moderator cans and run downward to the lower part of the reactor vessel. The major portion of the sodium flows into the lower plenum below the grid plate and then flows upward through the process tubes to the sodium pool above the core. This flow carries the heat from the fuel elements. The remaining sodium (4 to 6%) flows upward in the gaps between the moderator cans and into the upper sodium pool. This sodium removes heat generated in the moderator elements. The alignment between moderator elements is maintained by spacers fastened to the upper ends of the cans. The cans are held in position at the top by a system of core clamps and are supported at the bottom on pedestals attached to the grid plate.

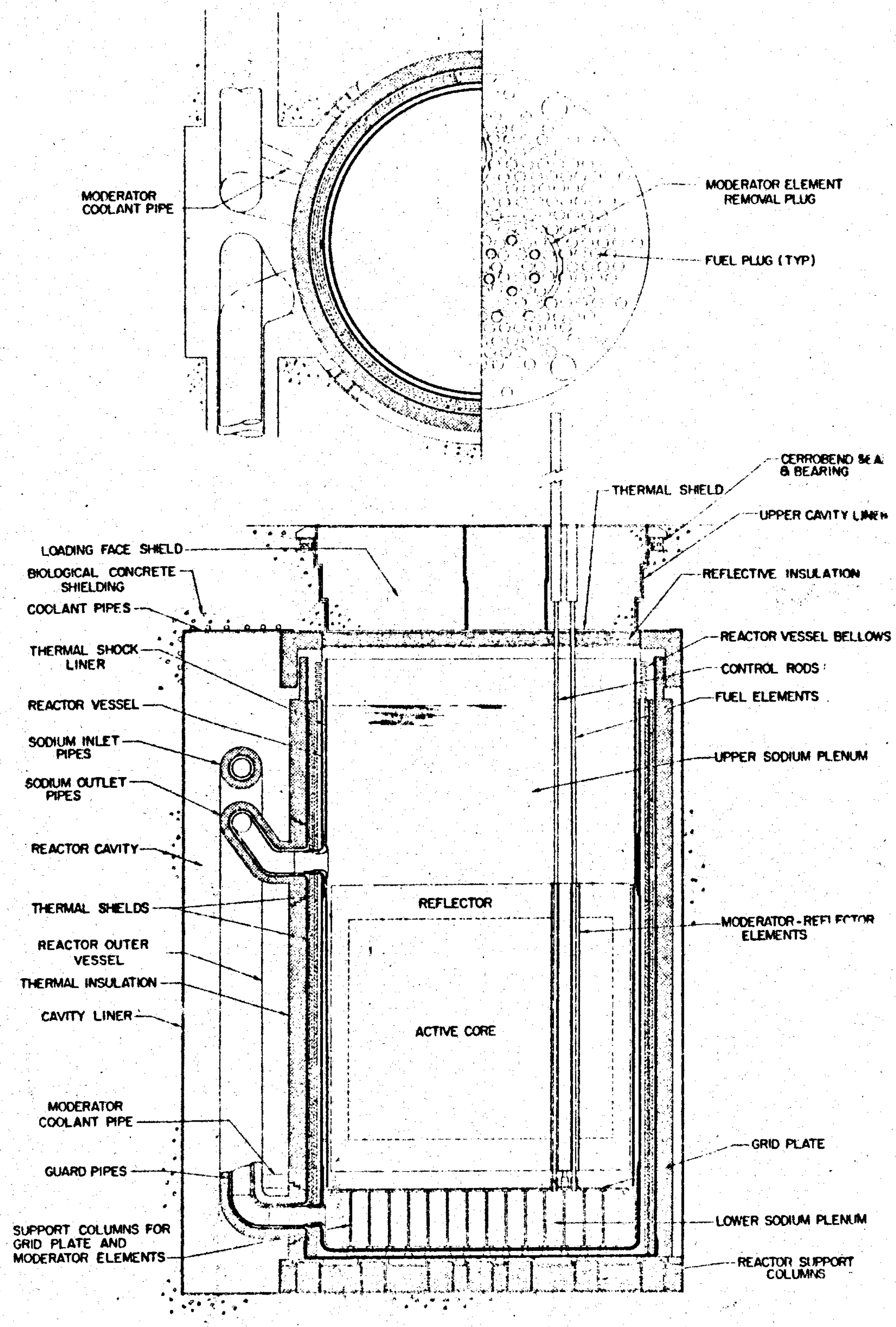


Figure 1. Reactor General Arrangement (300 Mw SGR)

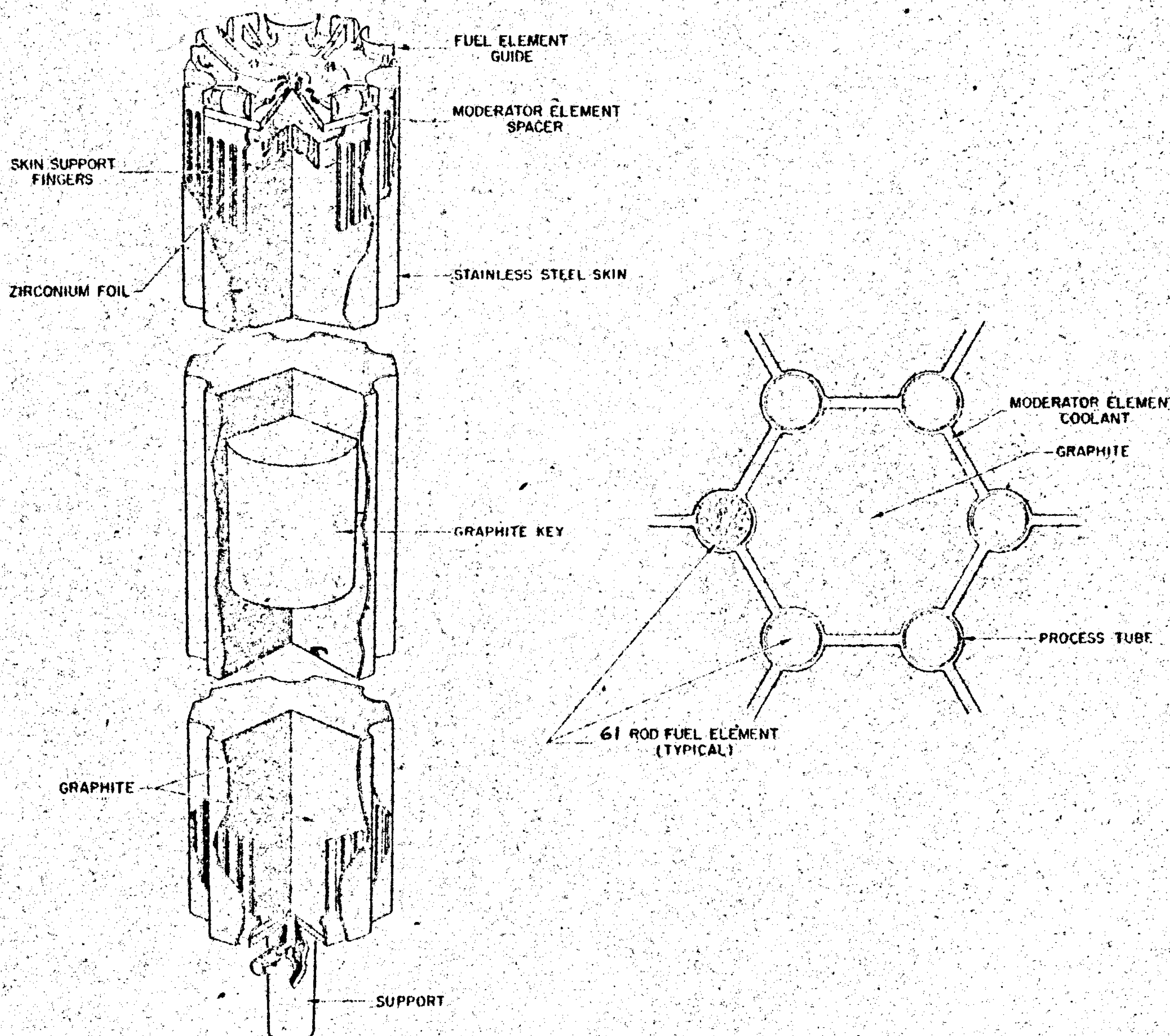


Figure 2. Moderator-Reflector Element

The moderator and reflector elements are graphite prisms hexagonal in cross section, canned in stainless steel. The elements are 16 inches across the flats by 19 feet long. The moderator can is Type 304 stainless steel 0.016 in. thick and has end closures or heads that are designed to compensate for differential thermal expansion between the graphite and the stainless steel. The top head is equipped with a spacer mechanism that maintains the proper gap between cans. This spacer is made of Type 405 stainless steel to minimize thermal expansion. The bottom head has a tubular support that fits over pedestals on the grid plate. A pump-out tube is built into the bottom head through which the can is evacuated prior to loading into the core. A mold-grade graphite with a density of 1.65 gms/cc is used.

The fuel elements are 61-rod clusters of Type 304 stainless steel tubing containing fuel slugs. The tubing is of 0.010 inch wall thickness and is sized for a 0.010 inch radial clearance around the 0.33 in. diameter fuel slug for a sodium bond. The fuel slugs are slightly enriched uranium-10 w/o molybdenum metal alloy. The active length of the fuel element is 15 ft. A helium volume is provided at the top of each fuel rod to provide for expansion of the sodium bond and to contain fission-product gases which might diffuse out of the fuel as it is irradiated.

In the reflector region, process channels formed by the corners of the graphite elements are not needed for fuel. These channels are filled with cylindrical canned graphite elements called reflector filler elements. These displace the sodium that would otherwise occupy this space and serve to increase the density and efficiency of the reflector.

The spare fuel channels in the active region of the core are filled with a similar device called a dummy element. This also serves to displace excess sodium and improve moderator density. Unlike the filler element which is attached to the grid plate, the dummy element is suspended in the process channel in a manner similar to that for the fuel element.

Table I

Significant Core Parameters for Reference Reactor

Reactor Thermal Power	884 MW
Electrical Output (net)	300 MW
Core Dimensions	
Height	15 ft.
Diameter	15.5 ft.
Moderator and Reflector Elements	
Hexagonal graphite logs - scalloped corners - stainless steel canned	
Length	19 ft.
"Across Flats"	~~16 inches~~
Graphite density	1.65 gms/cc
Stainless steel canning thickness	0.016 in.
Number of elements	~~174~~ [illegible]
Moderator Temperature (Avg)	450°C
Fuel Elements	
61 rod clusters - U-10 w/o Moly alloy fuel - Sodium Bond - Stainless Steel canned	
Fuel slug diameter	0.33 inches
Sodium Bond	0.010 in. (radial)
SS Cladding thickness	0.010 in.
Fuel Cross Sectional Area Rod Element	0.0839 $in.^2$
Element	5.2 in^2.
Fuel density	
U-10 w/o Moly alloy	17.1 gms/cc
Uranium density	15.4 gms/cc
Coolant	
Sodium Coolant	
Flow Area (each fuel channel)	9.5 in^2
Maximum sodium velocity	19.5 ft/sec
Inlet temperature	~~607°F~~ 675
Outlet Temperature	945°F
Core pressure drop	12.9 psi

Table I Continued

Weight of fuel per element	
U-10 w/o Moly alloy	263 kg
Uranium	237 kg
Number of fuel elements in core	214
Core loading	
U-10 w/o Moly alloy	56,200 kg
Uranium	50,600 kg

Control

B.2 Basis of Economic Evaluation

The standard factors as established for the Ten Year Civilian Power Reactor Development Program have been used for all the cost calculations.

Of primary interest are the bases for the fuel cycle cost studies; namely

(a) 80% Plant Factor

(b) 4% Annual Fuel Lease charge

(c) Uranium value based upon AEC Price Schedule

(d) Shipping cost - \$12.45/kg U

(e) U NO_3 to UF_6 Conversion - \$5.60/kg U

(f) $PuNO_3$ to Pu Metal - \$1.50/gm Pu

In conformance with a request from the AEC Division of Reactor Development the fuel cycle costs were computed, in all cases, for more than one average fuel burnup. The average fuel burnups chosen for our studies were 3400, 11,000 and 16,800 MWD/MTU. These average burnups correspond to 7,000, 20,000 and 30,000 MWD/MTU peak burnup. The calculational methods for fuel cycle costs are illustrated in Table II. The fuel cost calculations conform to the general form established for the Ten Year Civilian Power Reactor Development Program (S & L Method) but the presentation has been oriented to better illustrate the variation in fuel cycle costs with core parameters.

Table II

Fuel Cycle for 300 MW SGR (Reference Reactor)

	3400 MWD/MTU			11,000 MWD/MTU			16,800 MWD/MTU		
	$/kg U		mills/kw hr	$/kg U		mills/kw hr	$/kg U		mills/kw hr
A. Fuel Element Fabrication									
1. Fuel Processing and Fuel Element Assembly	$ 83.52			$ 86.98			$ 89.75		
2. Fuel Inventory Charges during fabrication (9 months)	11.60			15.60			18.70		
3. Shipment of New Fuel	3.00			3.00			3.00		
		$ 98.12	3.54		$105.58	1.18		$111.45	.81
B. Fuel Inventory Charges (at Reactor Site)									
1. Spares (15% of Core Load)	$ 1.55			$ 6.71			$ 12.35		
2. Core loading	9.48			36.30			63.20		
3. Post irradiation cooling period (4 months)	4.30			4.34			4.45		
		$ 15.33	.55		$ 47.35	.53		$ 80.00	.58
C. Net fuel Burnup									
1. Uranium Depletion									
Initial Uranium Value	$ 386.34			$ 519.71			$ 623.85		
Final Uranium Value	322.51			324.99			333.10		
	$ 64.33			$194.72			$290.75		
2. Net Plutonium Credit ($10.27/gm Pu) (12/gm Pu less 2% loss & Process charges)	19.30	45.03	1.63	46.80	147.92	1.65	61.60	229.15	1.67
D. Uranium Recovery									
1. Shipment of Irradiated Fuel	$ 12.45			$ 12.45			$ 12.45		
2. Inventory during shipment (1/3 month)	.36			.36			.37		
3. Separations	18.45			20.50			23.30		
4. Conversion to UF_6 ($5.60/kg x .99)	5.55			5.55			5.55		
5. Uranium losses (1.3%)	4.19			4.23			4.33		
6. Inventory during reprocessing	3.92			6.16*			5.73*		
		44.92	1.62		49.35	.55		51.73	.38
		$203.40	7.34		$ 350.20	3.91		$ 472.33	3.45
E. Fuel Fabrication Capital Charges			.09			.10			.10
Total Fuel Cycle Costs			7.43			4.01			3.55

* Include Inventory Charges during holdover of one fuel discharge - 2 fuel discharges processed together.

The capital investment required for the 300 MW SGR is summarized in Table III.

Table III

Summary of Capital Cost Estimate for 300 MW SGR

I	Land & Land Rights	$ 360,000
II	Structures and Improvements	7,250,000
III	Reactor Plant Equipment	9,400,000
IV	Heat Transfer System	22,850,000
V	Turbo Generator System	11,550,000
VI	Accessory Electrical Equipment	2,500,000
VII	Miscellaneous	500,000
VIII	Main Step-Up Transformer (Included VI)	
	Total Direct Construction Cost	$54,410,000
IX	Indirect Construction Costs	16,338,000
	Total Direct and Indirect Costs	$70,748,000
X	Contingency	7,075,000
XI	Escalation	
XII	Interest During Construction	6,304,000
	Total Capital Cost	$84,127,000

The following standard factors as established for the Ten Year Civilian Power Reactor Development Program have been used in the Capital Cost estimate.

(a) General and Administrative - 12.5% of Direct Cost

(b) Engineering Design and Inspection - 14.6% of Direct Cost plus G & A.

(c) Start-Up Costs - An allowance of $600,000 has been included for this item.

(d) Contingency - 10% of Direct and Indirect Costs

(e) Interest during construction - 8.1%.

(f) 14% Annual Capital Charges

(g) 80% Plant Factor

Based upon 14% annual capital charges, 80% plant factor and 300 MW net electrical output the capital charges for this plant are 5.60 mills/kw hr.

The operation and maintenance costs for this reactor have been estimated to be 0.99 mills per kw hr.

The power generating costs for the reference design (300 MW SGR) are indicated in Table IV.

Table IV

Power Generating Costs for 300 MW SGR (mills per kw hr)

	3400 MWD/MTU	11,000 MWD/MTU	16,000 MWD/MTU
Capital Charges	5.60	5.60	5.60
Fuel Cycle Costs	7.43	4.01	3.55
Operation and Maintenance	.99	.99	.99
Total Power Generation Cost	14.02	10.60	10.14

C. Core Parameter Study

C.1 Variation in Lattice Spacing

A variation in lattice spacing through a change in moderator element size varies the fuel to moderator ratio and the diameter of the reactor. Increasing the amount of moderator in the core tends to increase core reactivity three ways:

(a) Increase probability that fission neutrons will escape resonance capture in U-238 while slowing down.

(b) Increases eta or average number of neutrons released per neutron capture in U-235. (Spectrum is more nearly thermal).

(c) Decrease neutron leakage from core by increasing core diameter.

Partially or completely off-setting this increase in reactivity is the additional neutron capture in the moderator. Maximum core reactivity will be reached at some ratio of fuel to moderator; an increase or decrease in amount of moderator from this point will decrease reactivity. This fuel to moderator ratio does not necessarily give lowest power cost.

The optimum fuel to moderator ratio is determined by an economic study, such as this, in which all factors including conversion ratio and capital costs as well as fuel enrichment are considered.

For this study the size of the moderator elements in the 300 MW SGR was varied from 14 inches to 26 inches across "flats" on the hexagonal graphite logs. The variations in core reactivity are indicated in Figs. .

The following interesting conclusions can be drawn from this study:

1. The maximum core reactivity is achieved with the fuel to moderator ratio obtained with a moderator element size slightly greater than eighteen inches (See Figure 3).

2. Minimum fuel cycle costs are achieved with a fuel to moderator ratio obtained with a moderator element size of approximately eighteen inches. (See Figs. 6, 7 &8). The optimum fuel to moderator ratio, considering fuel cycle costs only, is therefore greater than the fuel to moderator ratio which gives maximum reactivity. This is primarily due to the increased resonance capture in U-238 and higher conversion ratios at the higher fuel to moderator ratios.

3. Minimum power generating costs are achieved with a fuel to moderator fatio slightly greater than the fuel to moderator ratio for minimum fuel cycle costs. (See Figs.10, 11,& 12). With an inexpensive moderator and with no pressurization required in the primary coolant system, the reactor core size can be varied without significantly affecting capital costs.

It is also obvious that the fuel to moderator ratio can be varied within rather broad limits without significantly affecting power generation costs.

FIG. 3

Variation in Core Reactivity with Lattice Spacing

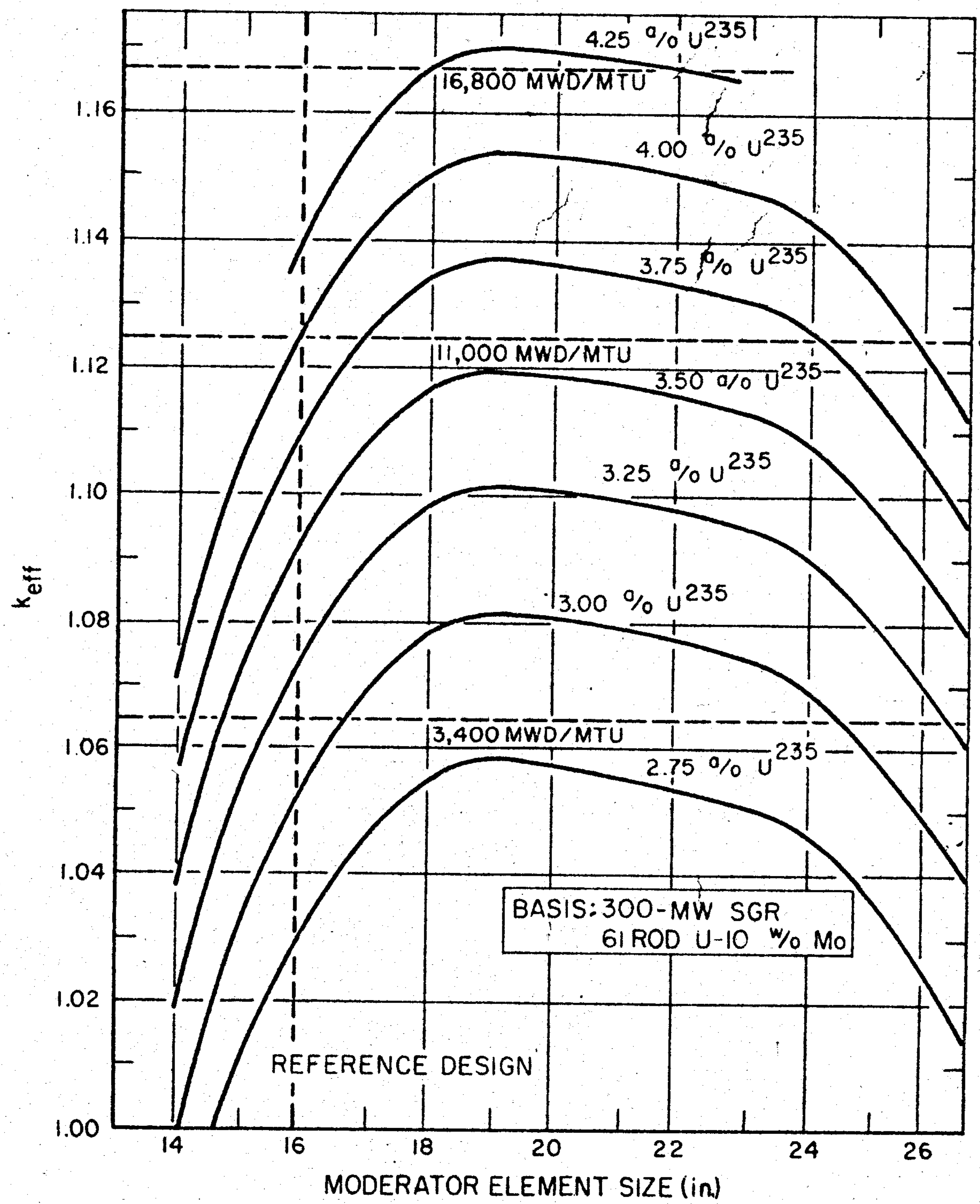


FIG. 4 - Variation in Uranium Enrichment With Lattice Spacing

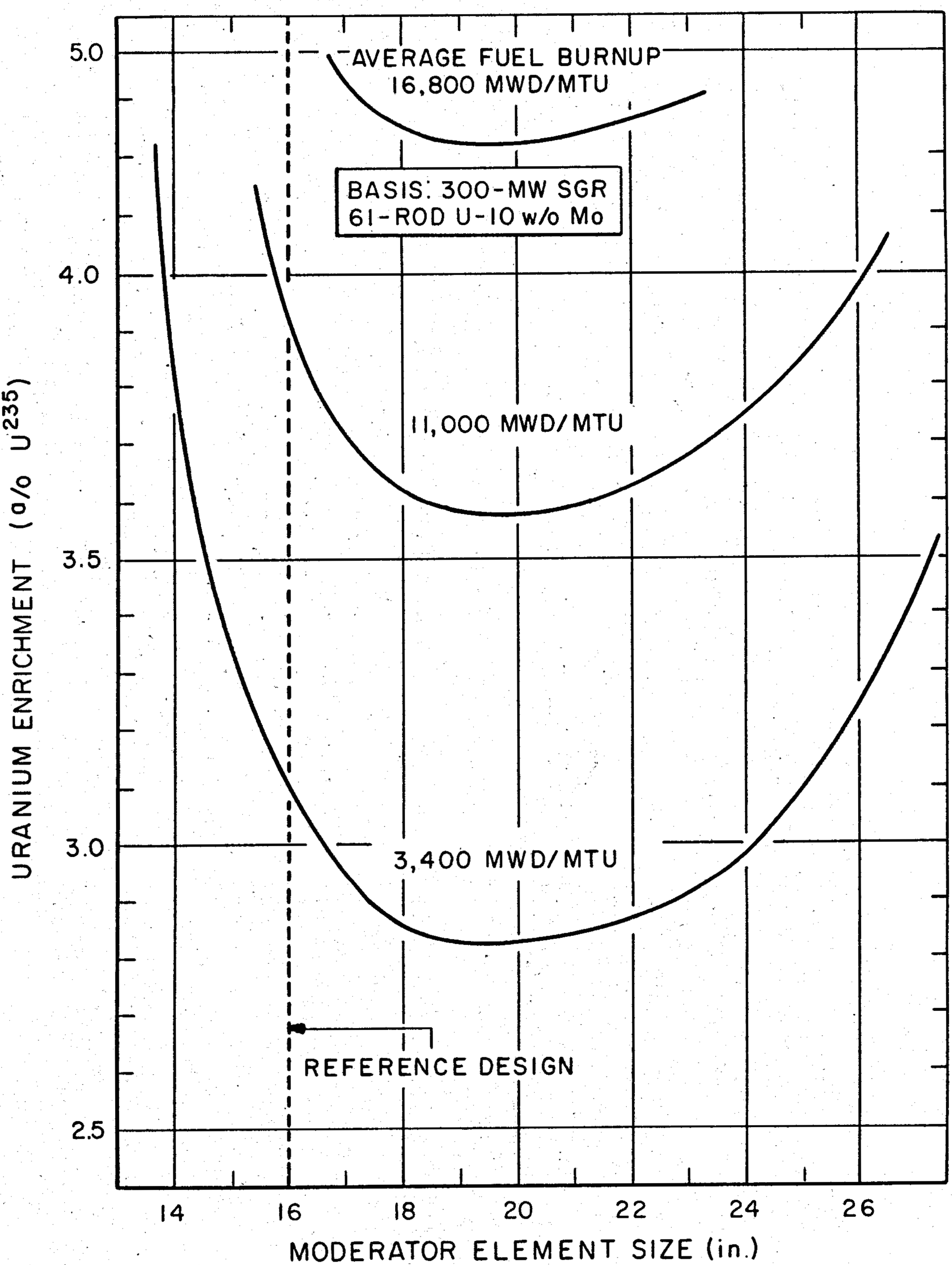


FIG. 5 - Variation in Fuel to Moderator Ratio with Lattice Spacing

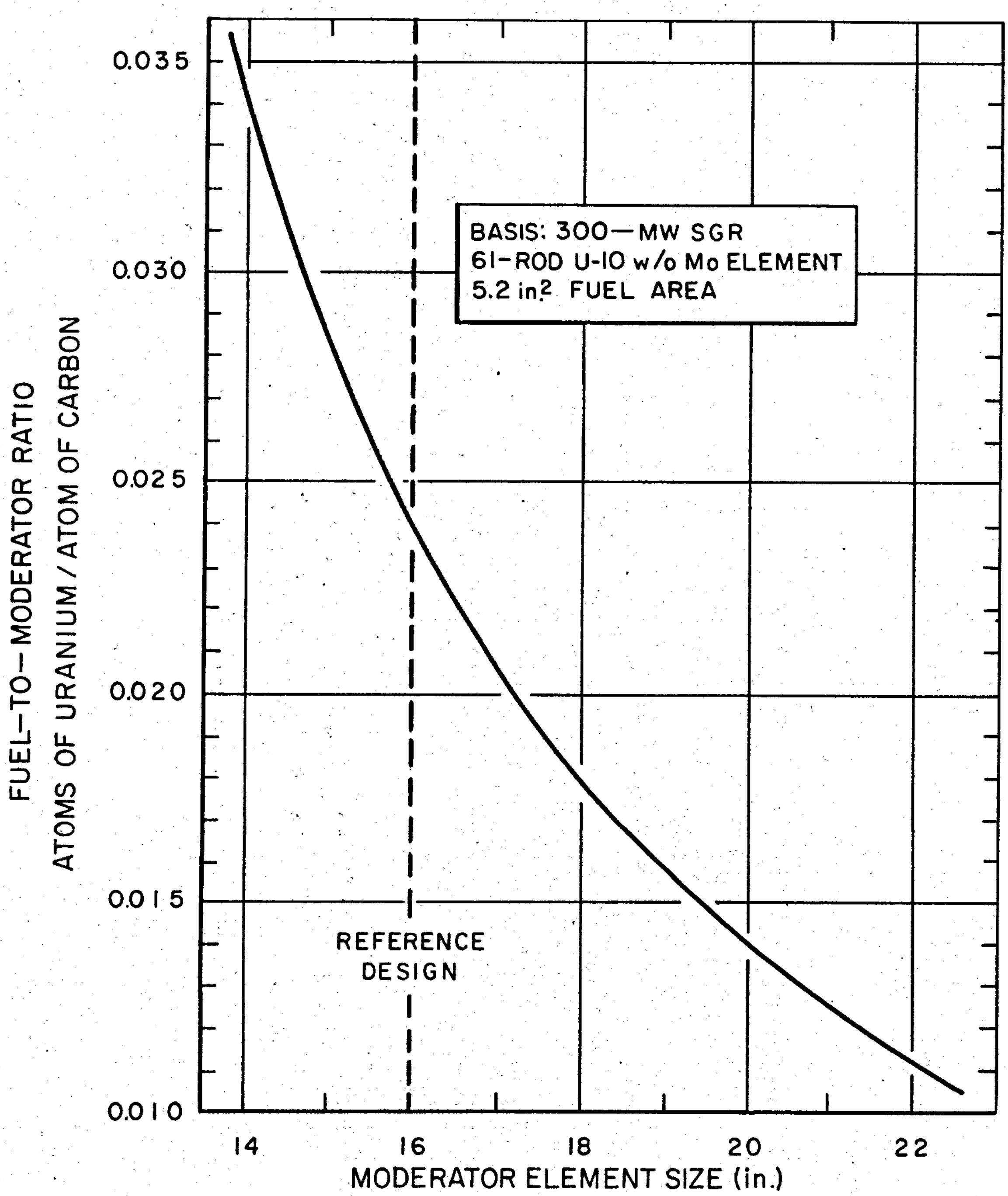


FIG. 6 - Variation in Fuel Cycle Costs with Lattice Spacing

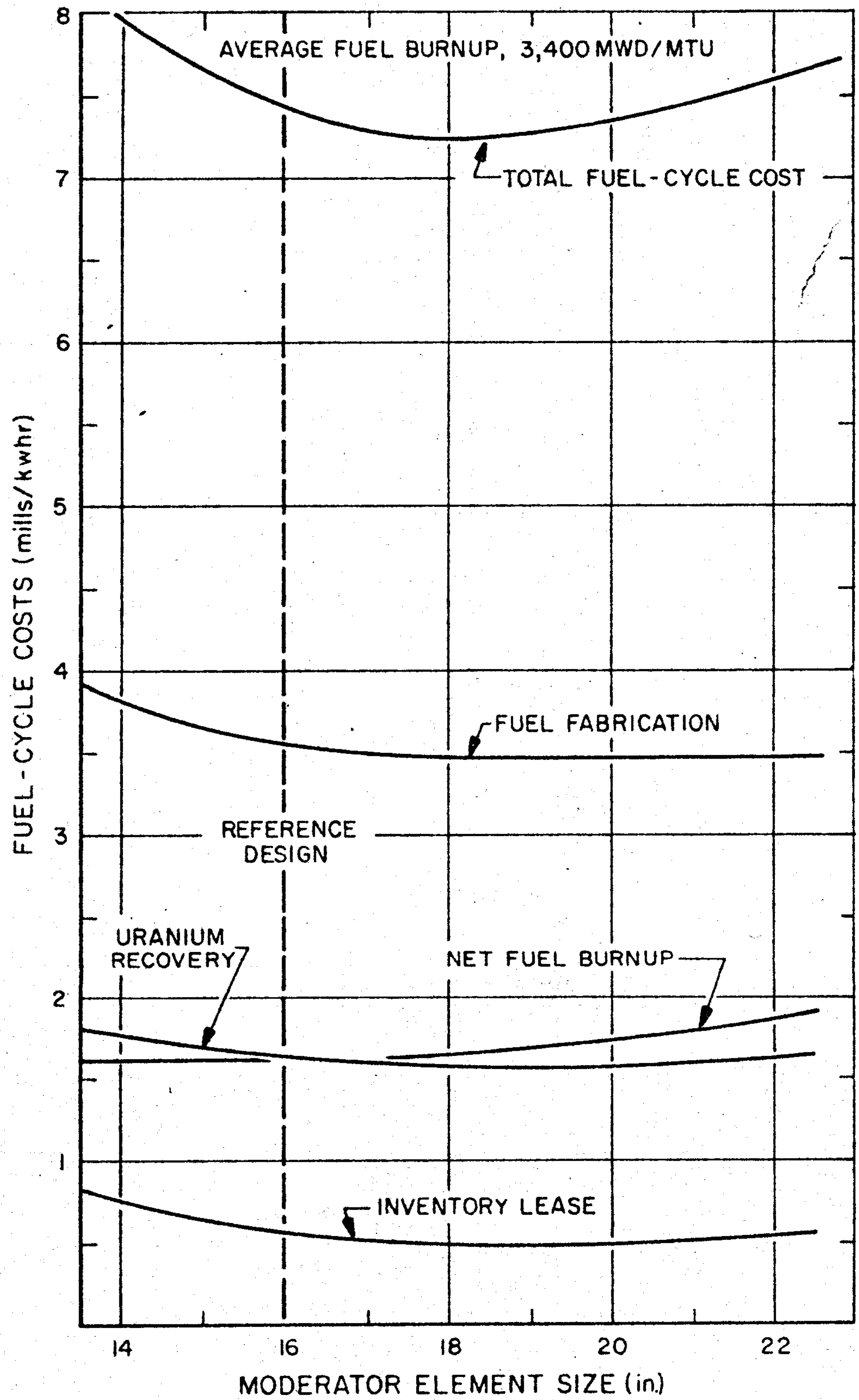


FIG. 7

Variation in Fuel Cycle Costs with Lattice Spacing
Average Fuel Burnup - 11,000 MWD/MTU

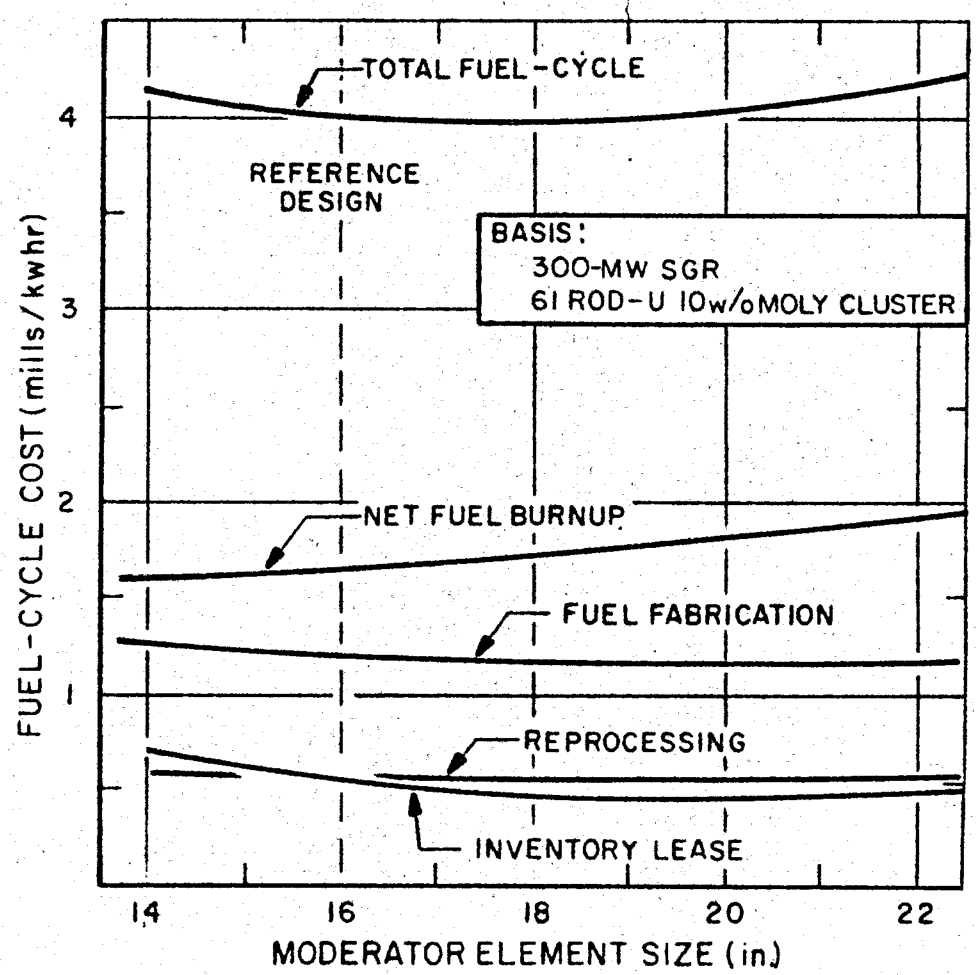
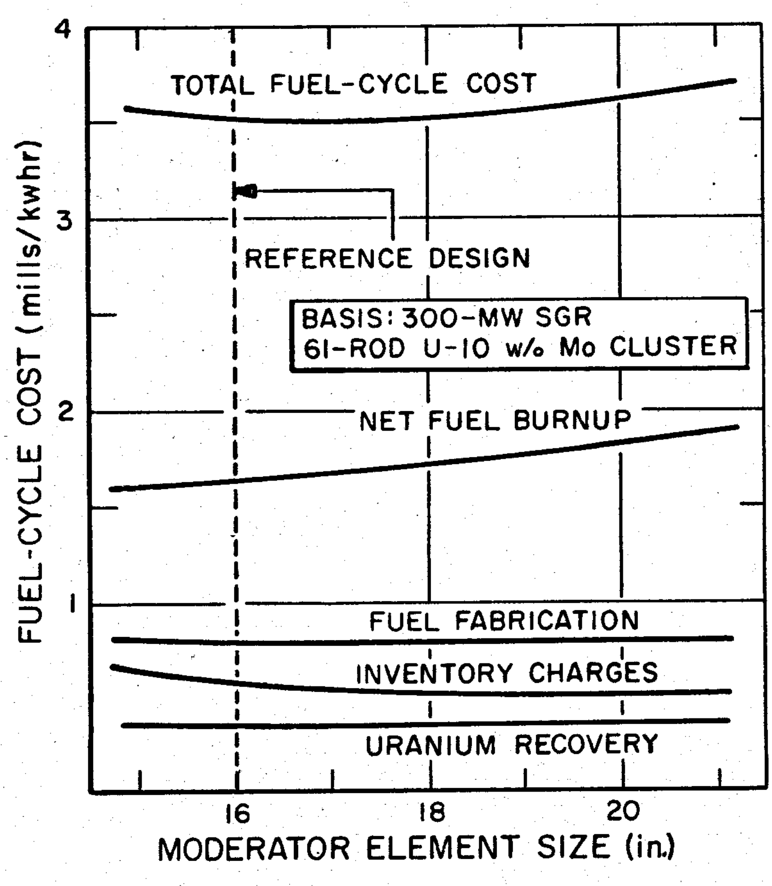


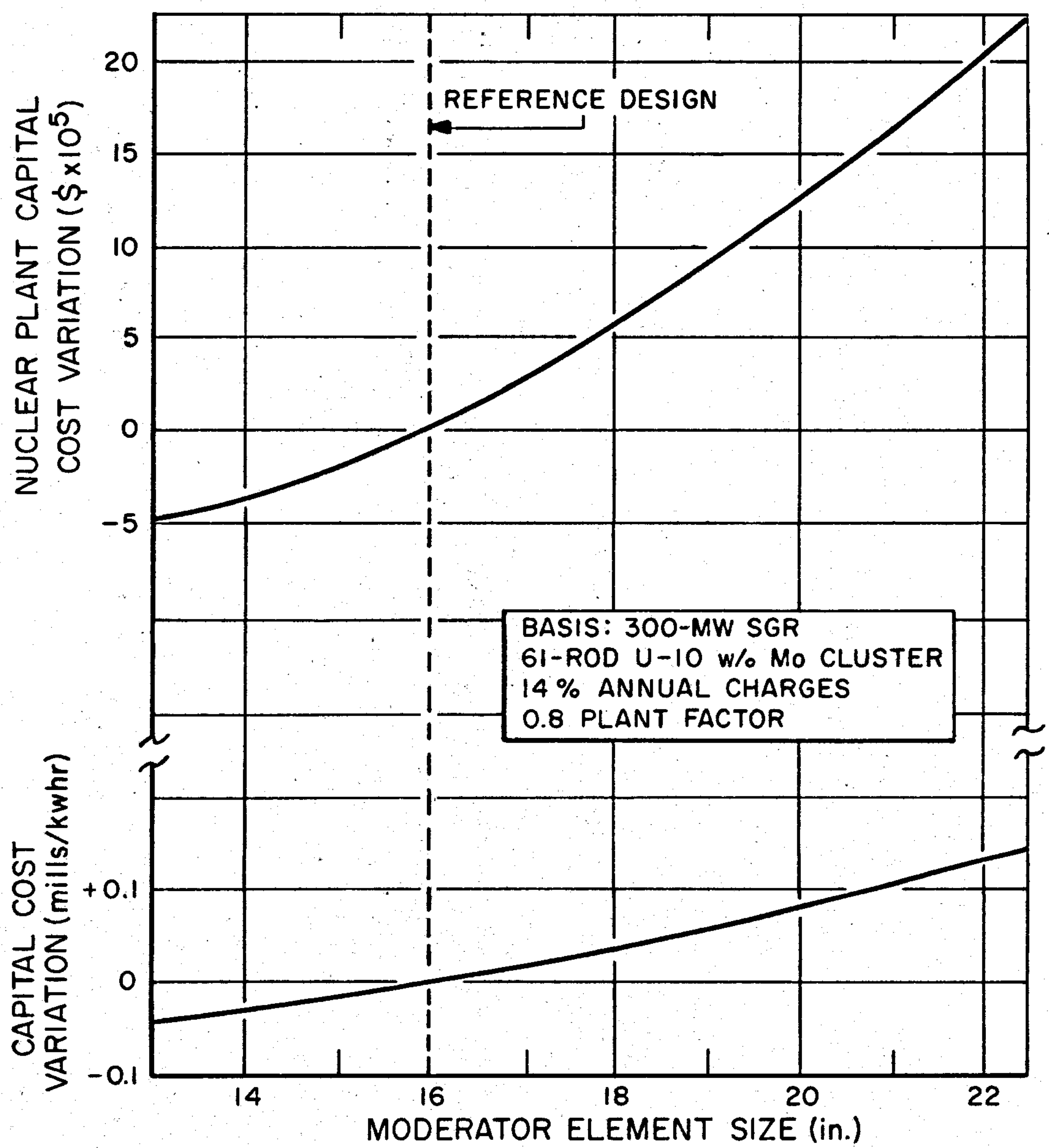
FIG. 8 - Variation in Fuel Cycle Costs with Lattice Spacing

Average Fuel Burnup - 16,800 MWD/MTU



Capital Cost Variation with Lattice Spacing

FIG. 9



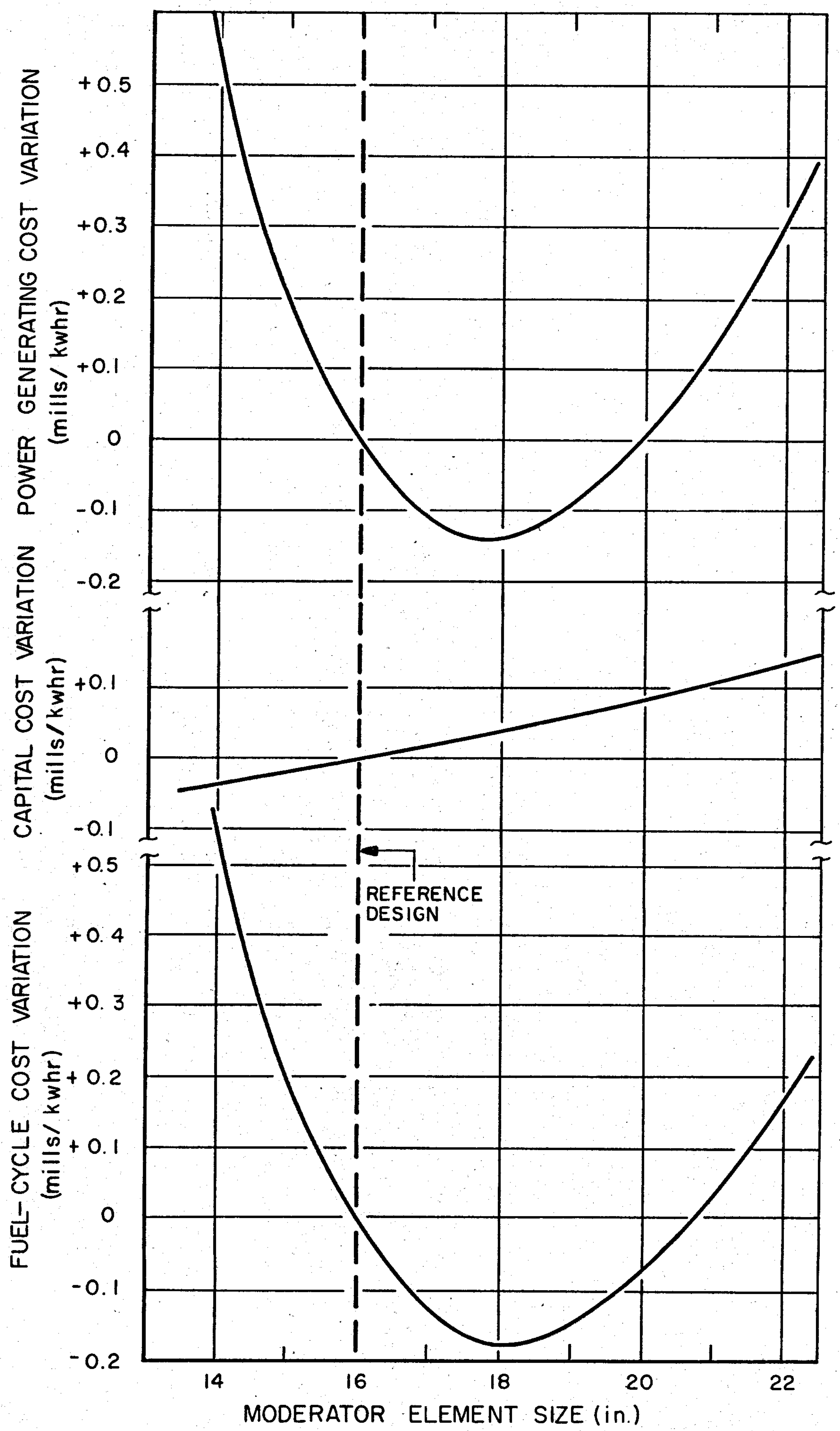


FIG. 10 - Variation in Power Generating Costs with Lattice Spacing
(Average Fuel Burnup - 3400 MWD/MTU)

FIG. 11 - Variation in Power Generating Costs with Lattice Spacing
(Average Fuel Burnup - 11,000 MWD/MTU)

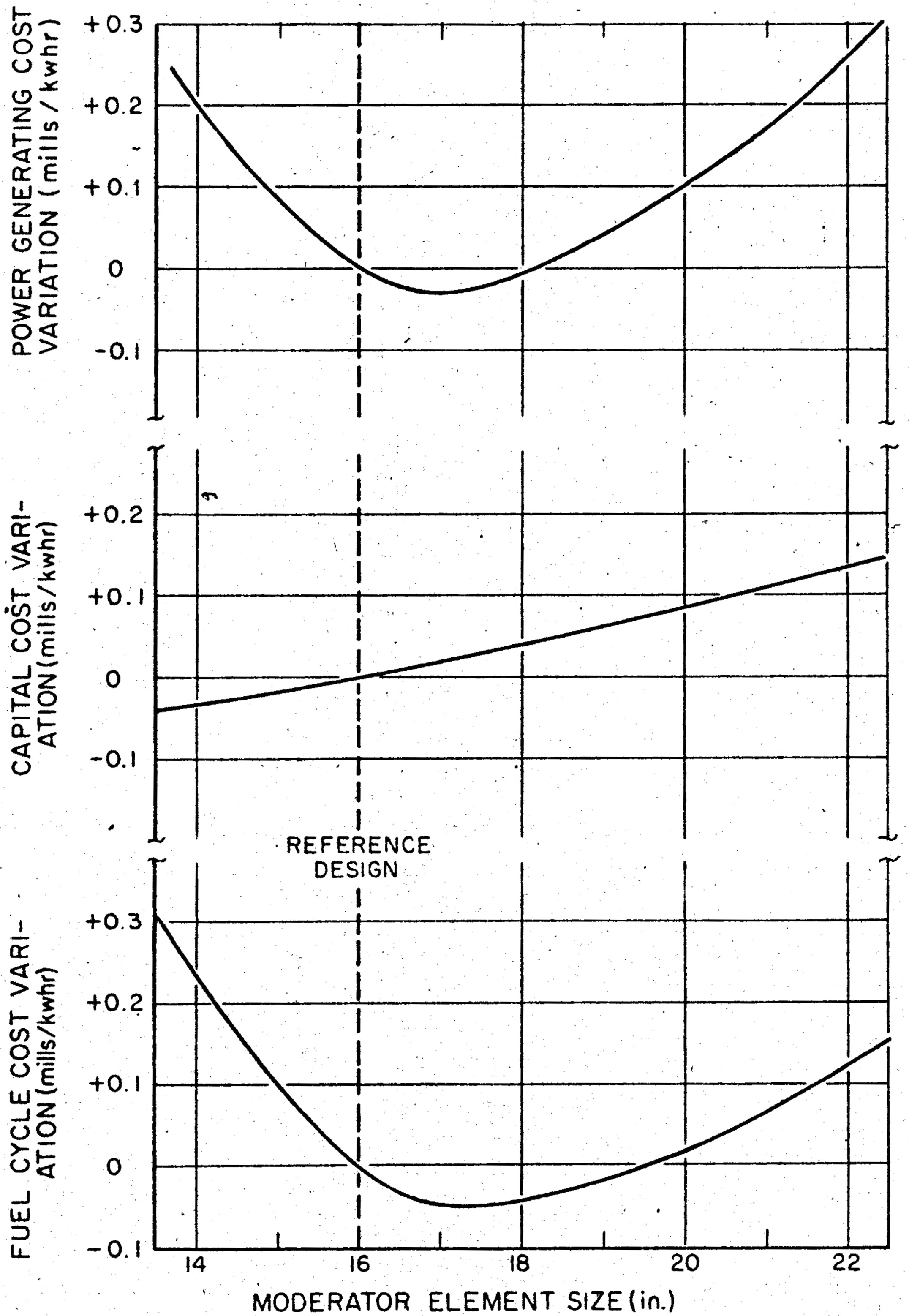
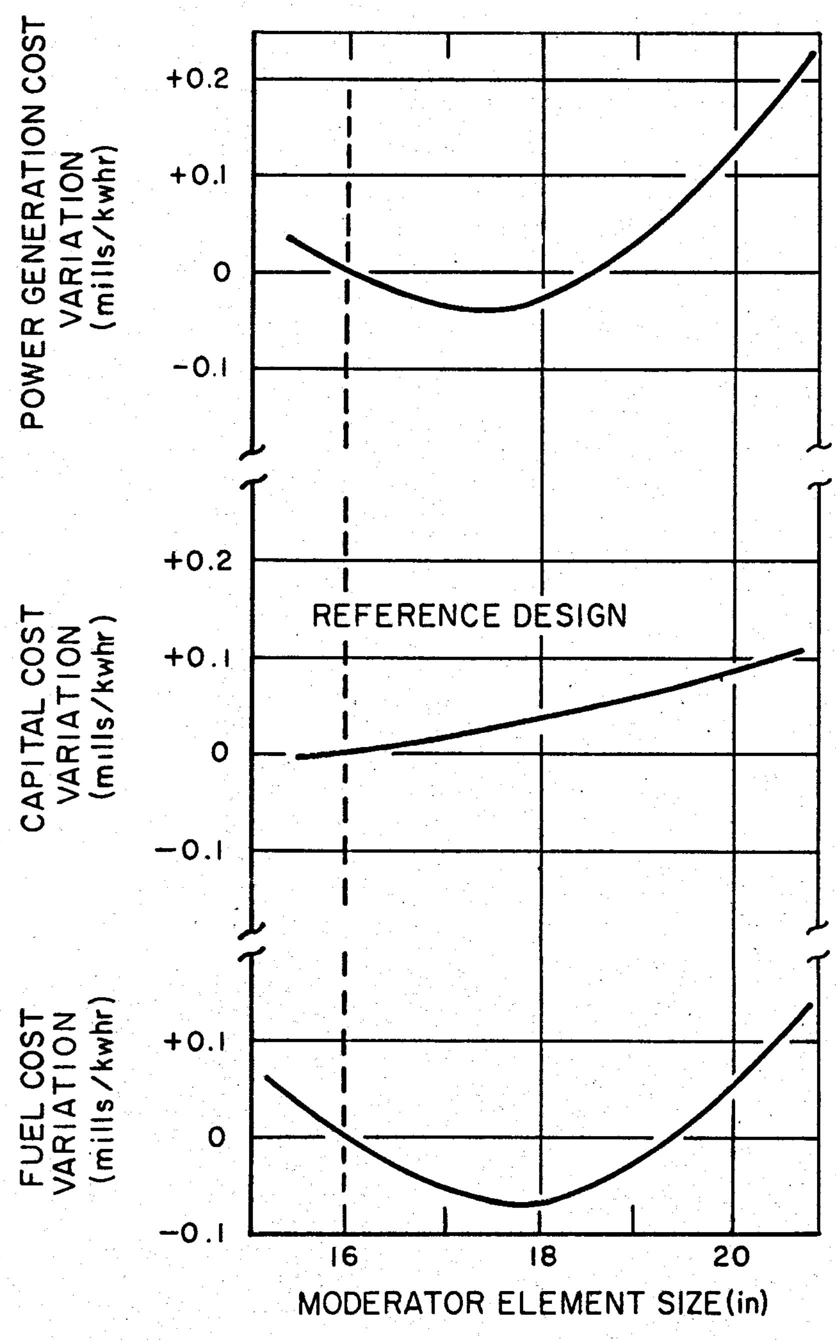


FIG. 12 - Variation in Power Generating Costs with Lattice Spacing

(Agerage Fuel Burnup - 16,800 MWD/MTU)



C.2 Variation in Fuel Rod Diameter

A variation in fuel rod diameter or fuel cross-sectional area affects many nuclear characteristics of the core.

An increase in fuel rod diameter with no change in moderator volume, tends to increase reactivity in two ways; (a) increase in fast fission of U-238 (b) increase in thermal utilization. In an undermoderated reactor such as the 300 MW SGR this increase in reactivity is offset by "hardening" of the neutron spectra. Fewer neutrons reach thermal energies because of increased opportunity for resonance capture in U-238 and increased fissioning of U-235 by intermediate energy neutrons (lower eta).

In this study the diameter of the fuel slugs was varied between 0.3 inch and 0.4 inch. No change was made in the sodium flow area. The size of scallops in the corners of the moderator elements was varied slightly in accordance with the variation in fuel element diameter.

The variation in core reactivity with a change in fuel rod diameter is indicated in Figure 13. The fuel enrichment required is indicated in Figure 14. The increased neutron capture in U-238 causes an increase in initial conversion ratio with an increase in fuel rod diameter.

Both fuel enrichment and fuel volume increase with an increase in fuel rod diameter resulting in an increase in the fuel inventory charges.

The cost to cast a fuel slug and assemble a fuel element is largely independent of fuel slug diameter. Therefore, for equivalent burnups, the fuel fabrication costs in mills per kw hr tend to decrease with increase in fuel slug diameter.

The variation in fuel burnup charges with fuel rod diameter is quite small. Figs. 16, 17 and 18 indicate the variation in fuel cycle costs with fuel rod diameter.

The change in fuel rod diameter requires no significant change in capital costs. The variation in power generating costs is therefore equal to the variation in fuel cycle costs.

The optimum fuel slug diameter (minimum power generation costs) will depend upon the average fuel burnup which can be obtained. Lower average fuel burnups favor larger slug diameters; for example, with an average fuel burnup of 3400 MWD/MTU the optimum slug diameter is approximately 0.36" but with an average fuel burnup of 16,800 MWD/MTU the optimum slug diameter is approximately 0.30".

FIG. 13 - Variation in Core Reactivity with Fuel Slug Diameter

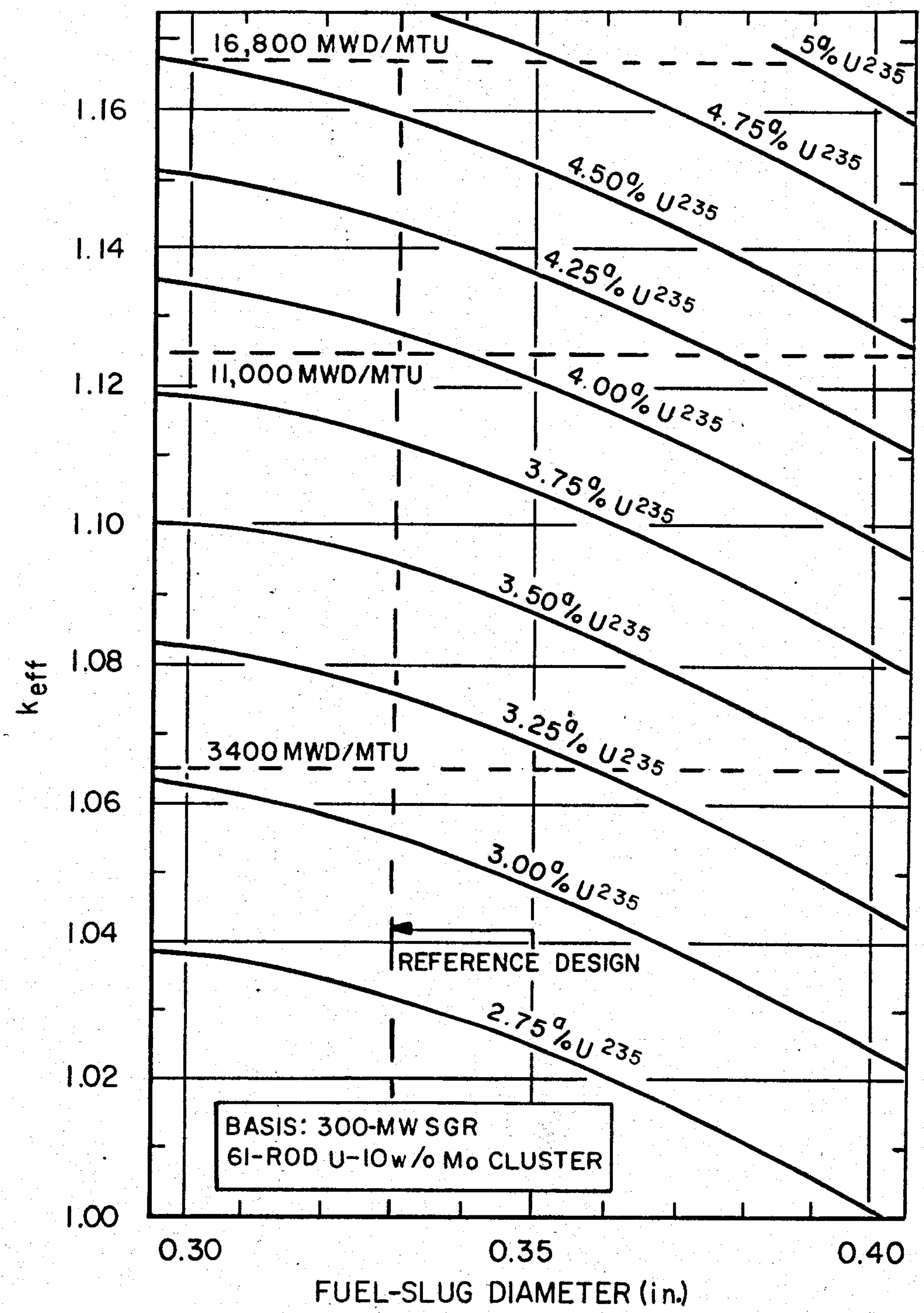


FIG. 14 - Variation in Uranium Enrichment with Fuel Slug Diameter

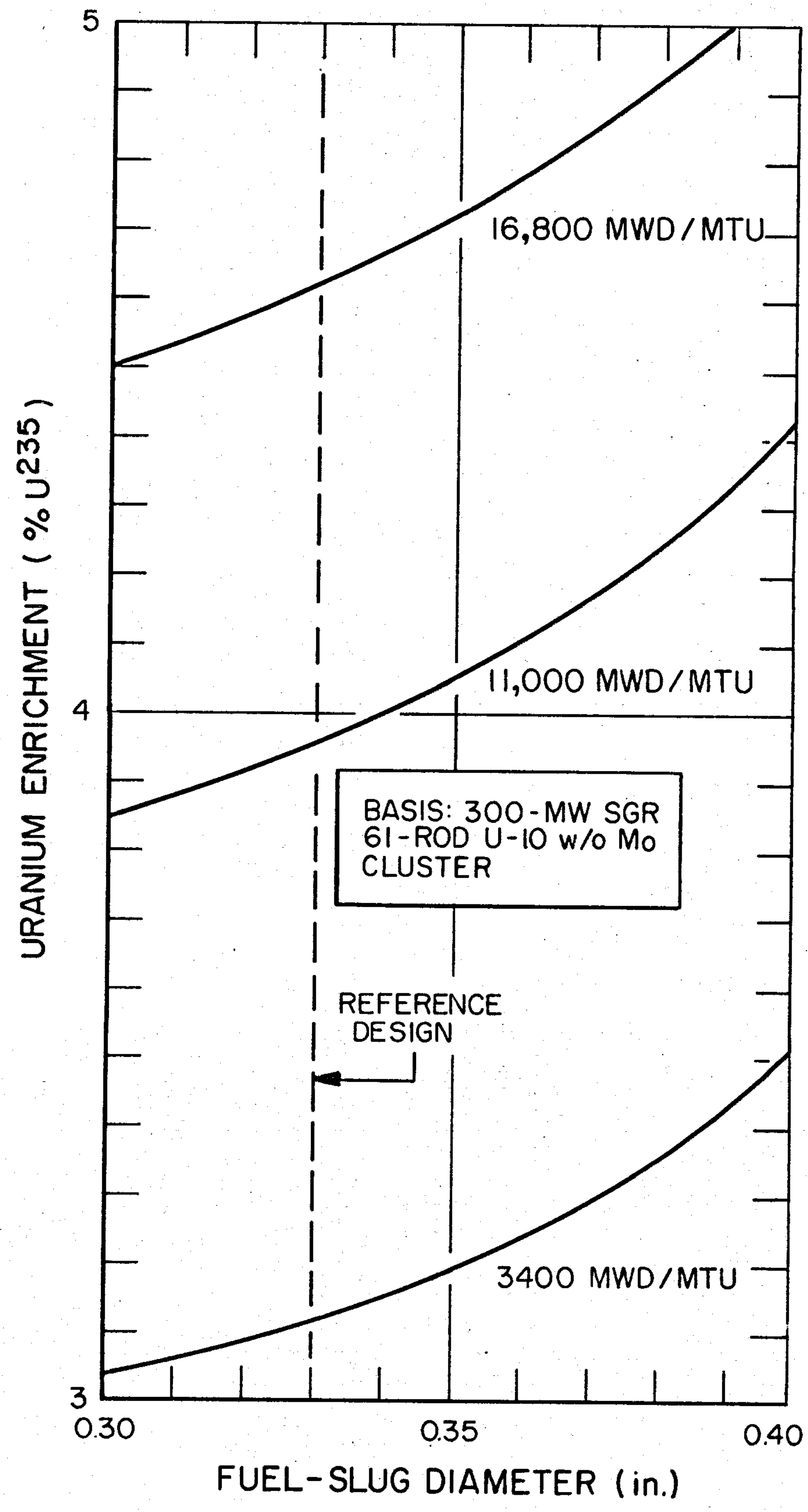
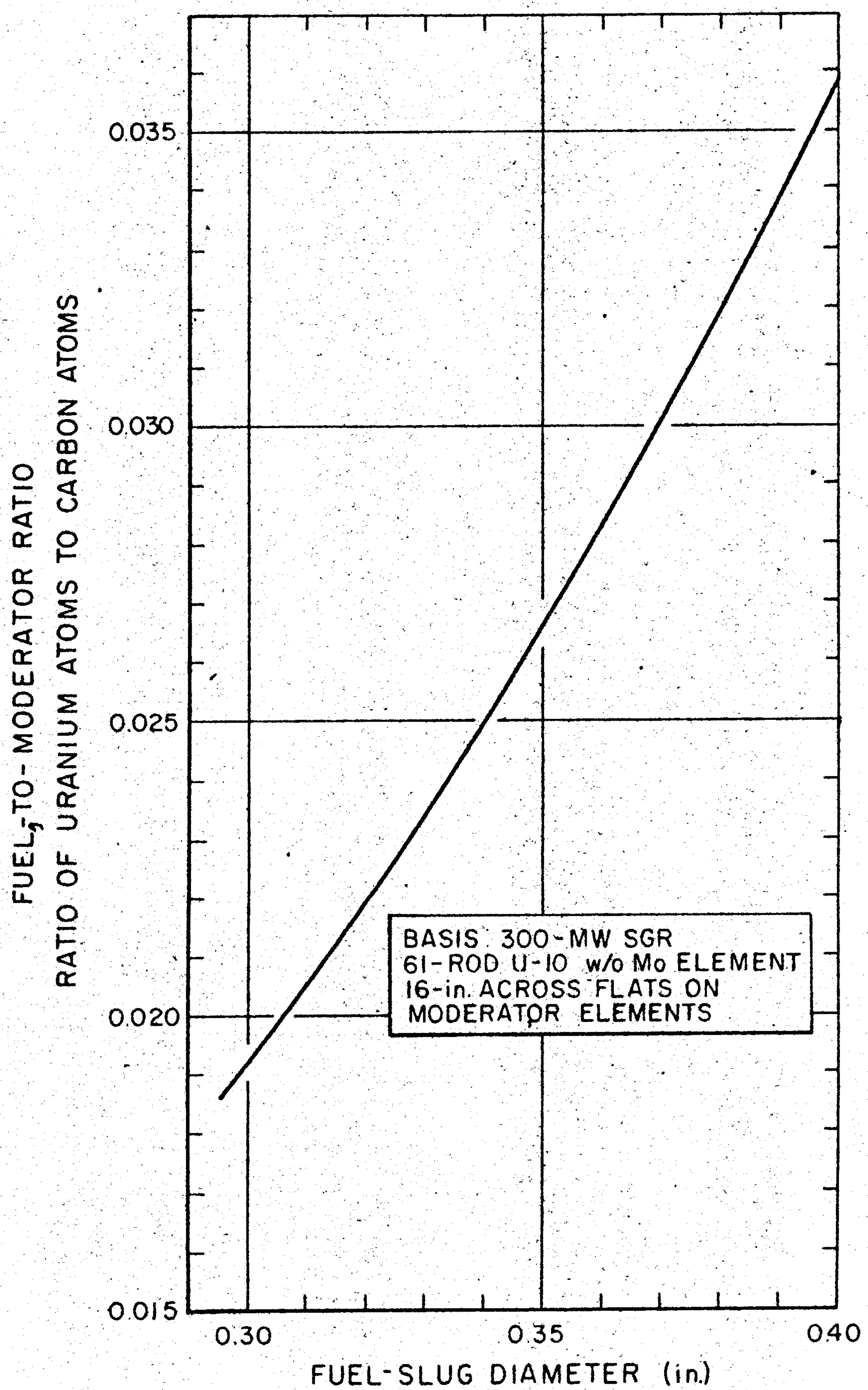


FIG. 15 - Variation in Fuel to Moderator Ratio with Fuel Slug Diameter



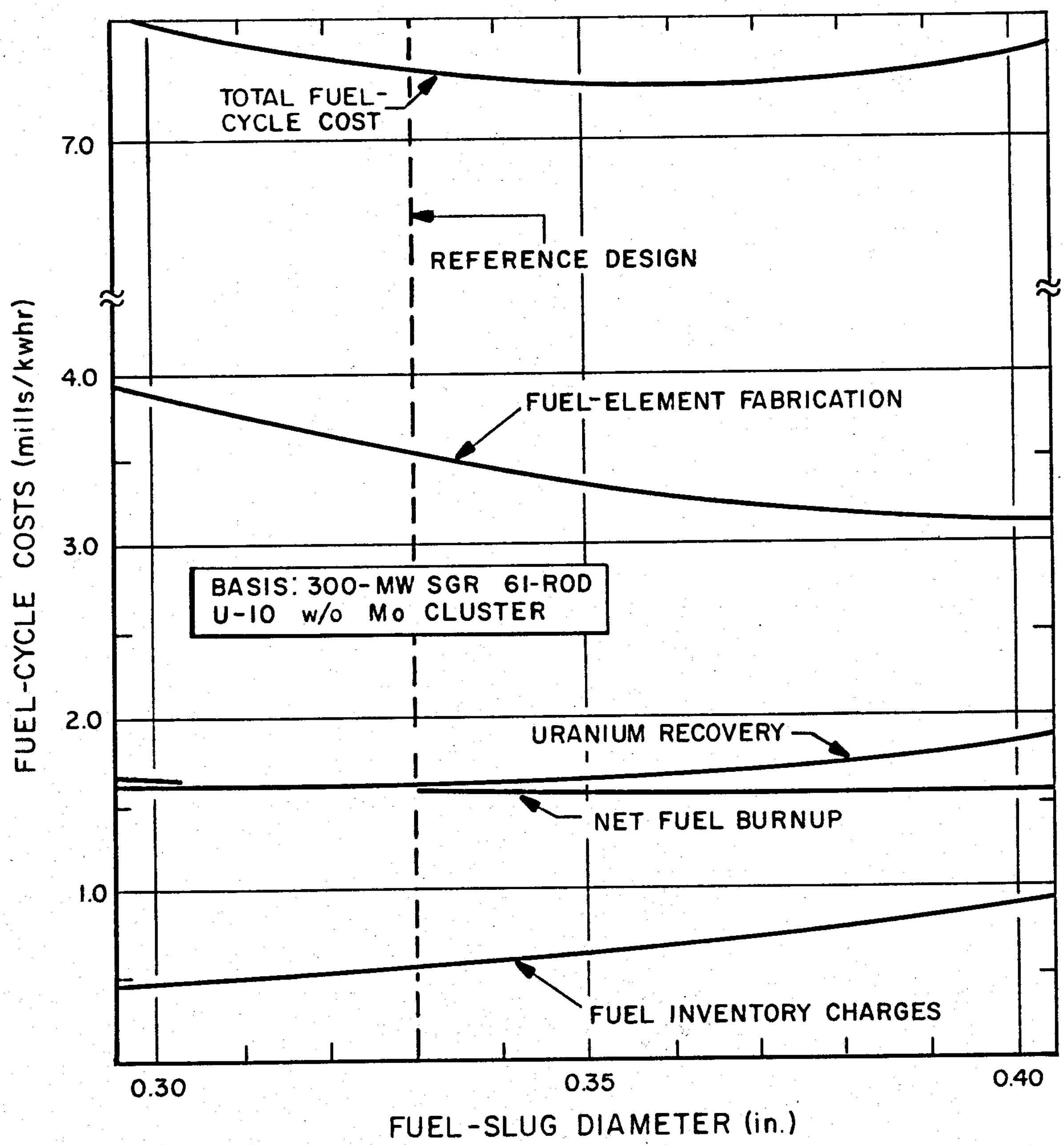


FIG. 16 - Variation in Fuel Cycle Costs with Fuel Slug Diameter

(Average Fuel Burnup - 3400 MWD/MTU)

FIG. 17 - Variation in Fuel Cycle Costs with Fuel Slug Diameter

(Average Fuel Burnup - 11,000 MWD/MTU)

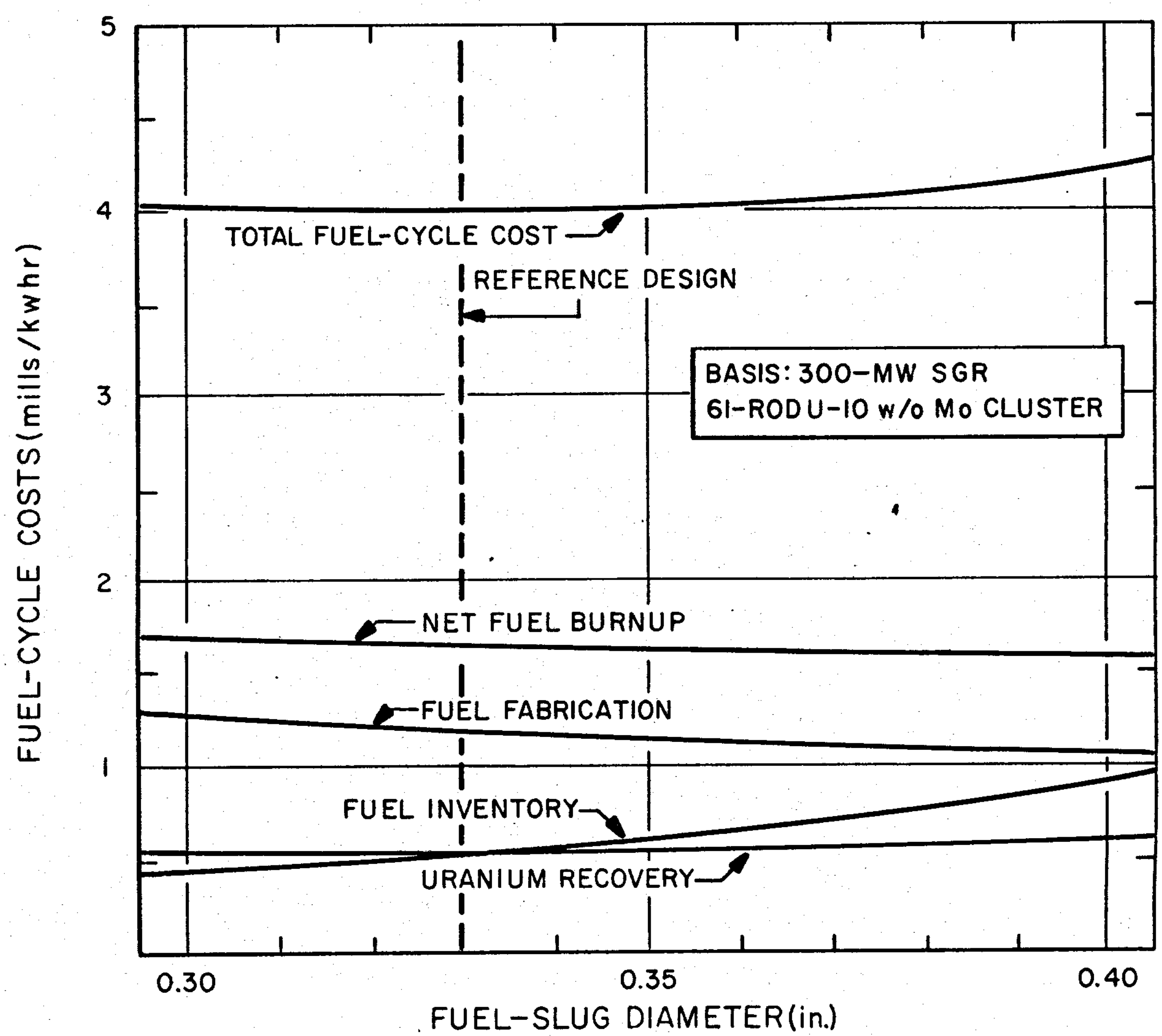
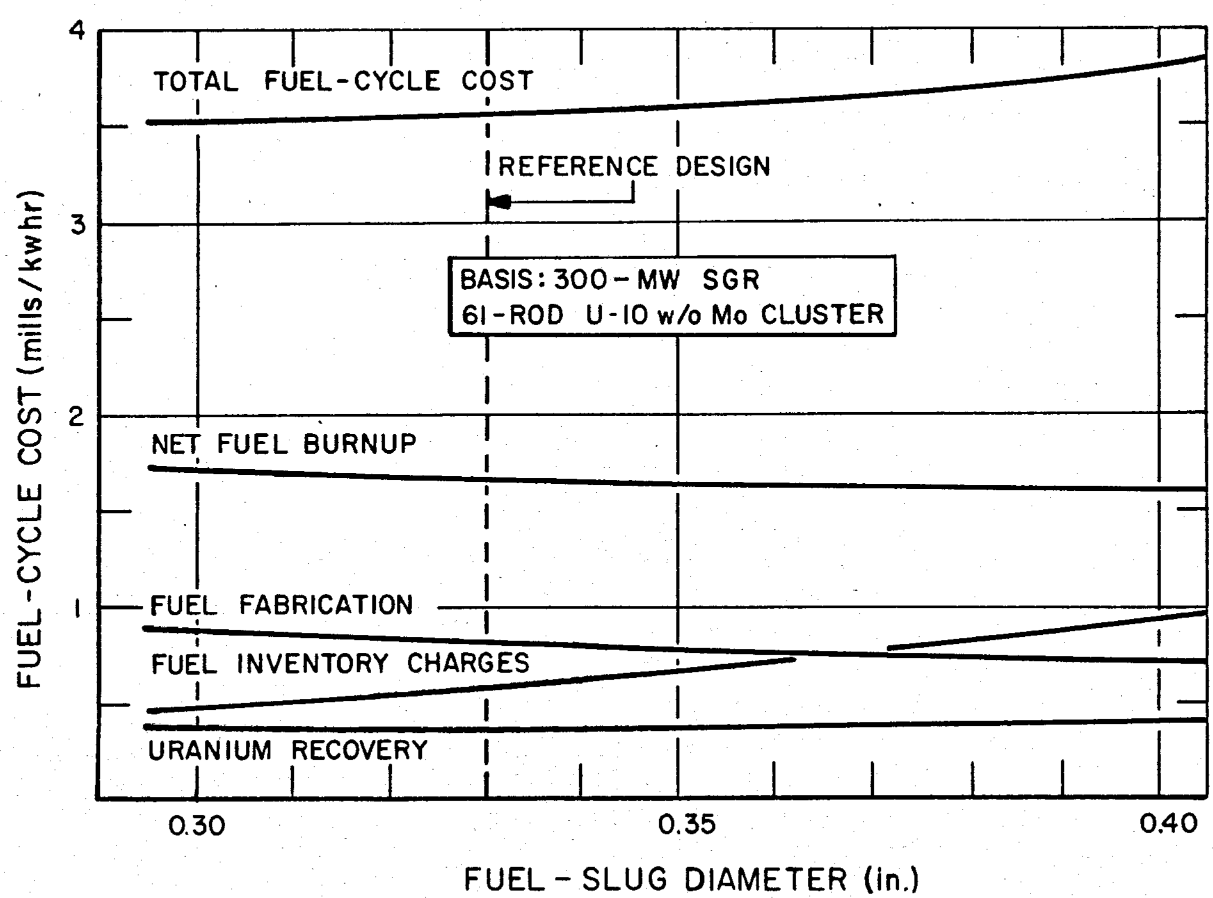


FIG. 18 - Variation in Fuel Cycle Costs with Fuel Slug Diameter

(Average Fuel Burnup - 16,800 MWD/MTU)



C.3 Variation in Fuel Cladding Thickness

In the reference design established for this reactor core parameter study the fuel cladding is stainless steel tubing with an 0.35" inside diameter and 0.010" thick. An increase in this cladding thickness increases the amount of nuclear "poison" in the core and therefore decreases core reactivity. The variation in core reactivity with fuel cladding thickness is indicated in Figure 19.

The increase in fuel cycle costs with increase in fuel cladding thickness is primarily due to the increase in fuel enrichment. The variation in fuel cycle costs with fuel cladding thickness is indicated in Figs. 21, 22, and 23 for average fuel burnup of 3,400 MWD/MTU, 11,000 MWD/MTU, and 16,800 MWD/MTU.

If there were no corresponding increase in fuel fabrication cost, a decrease in fuel cladding thickness to 0.005" from the 0.010" in the reference design would decrease fuel cycle costs slightly (~ 0.1 mill/kw hr).

FIG. 19 - Variation in Core Reactivity with Fuel Cladding Thickness

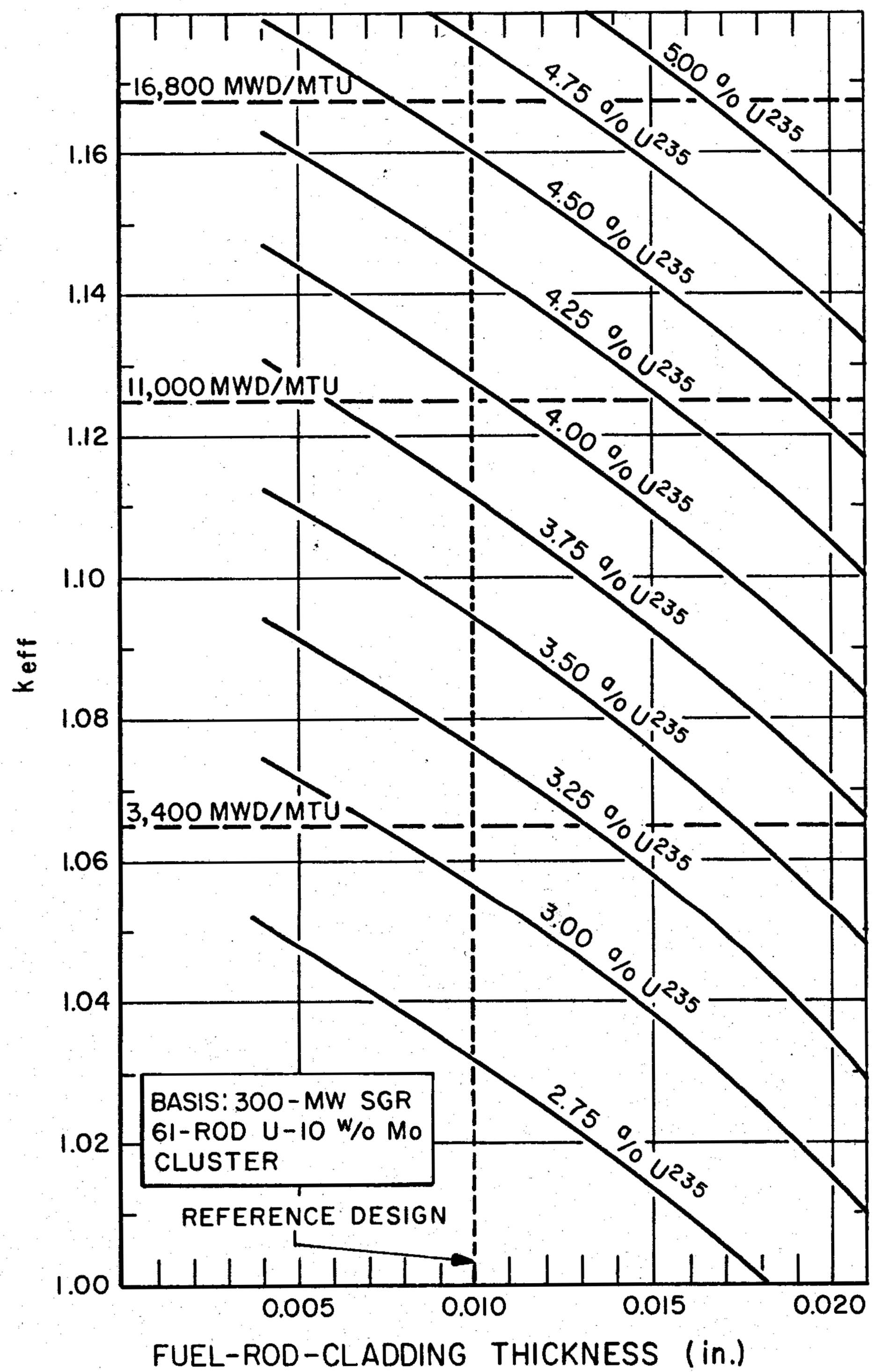


FIG. 20 - Variation in Fuel Enrichment with Fuel Cladding Thickness

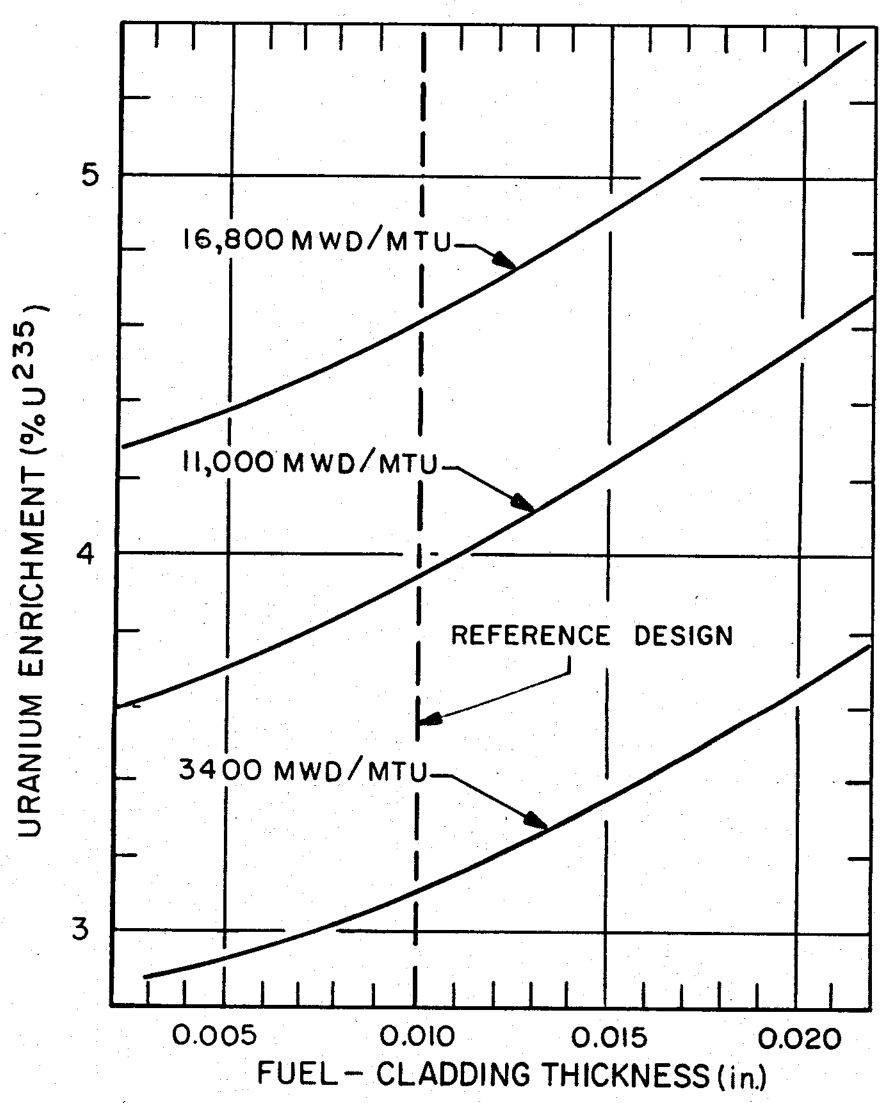


FIG. 21 - Variation in Fuel Cycle Cost with Fuel Cladding Thickness
(Average Fuel Burnup - 3400 MWD/MTU)

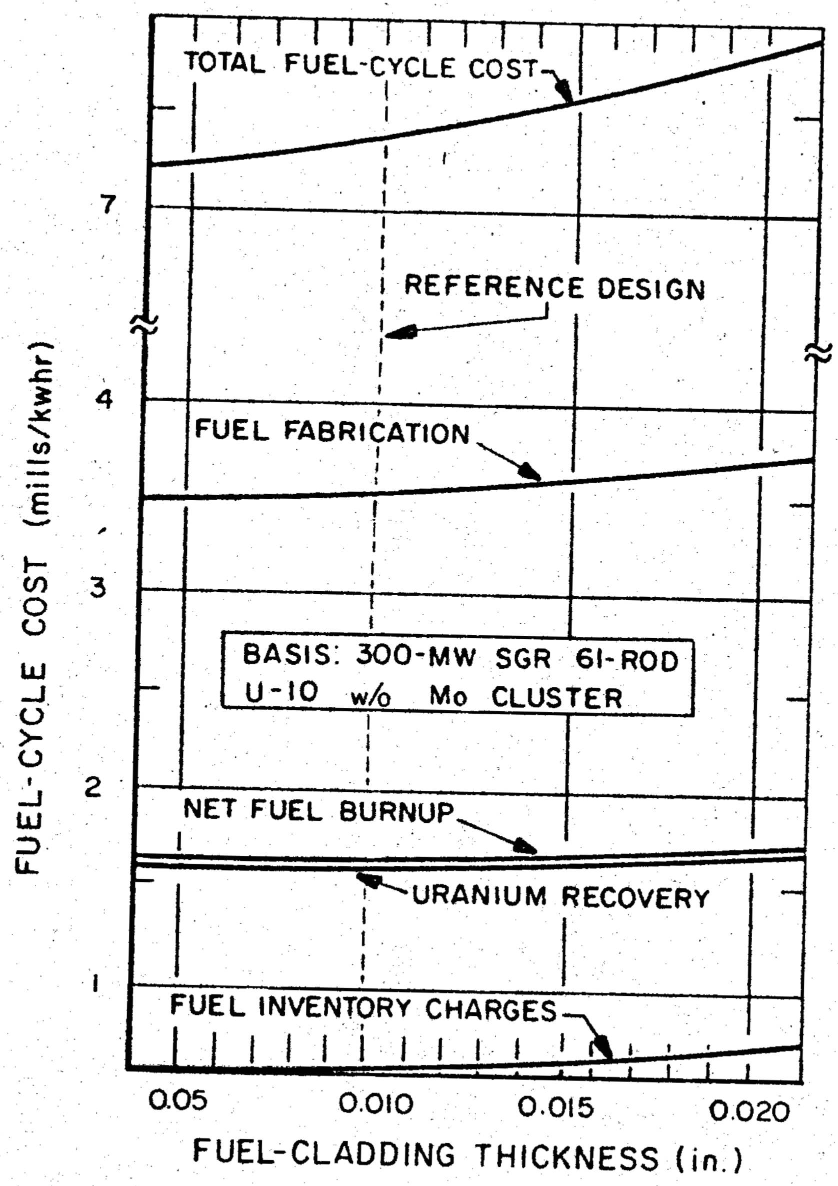


FIG. 21 - Variation in Fuel Cycle Cost with Fuel Cladding Thickness
(Average Fuel Burnup - 3400 MWD/MTU)

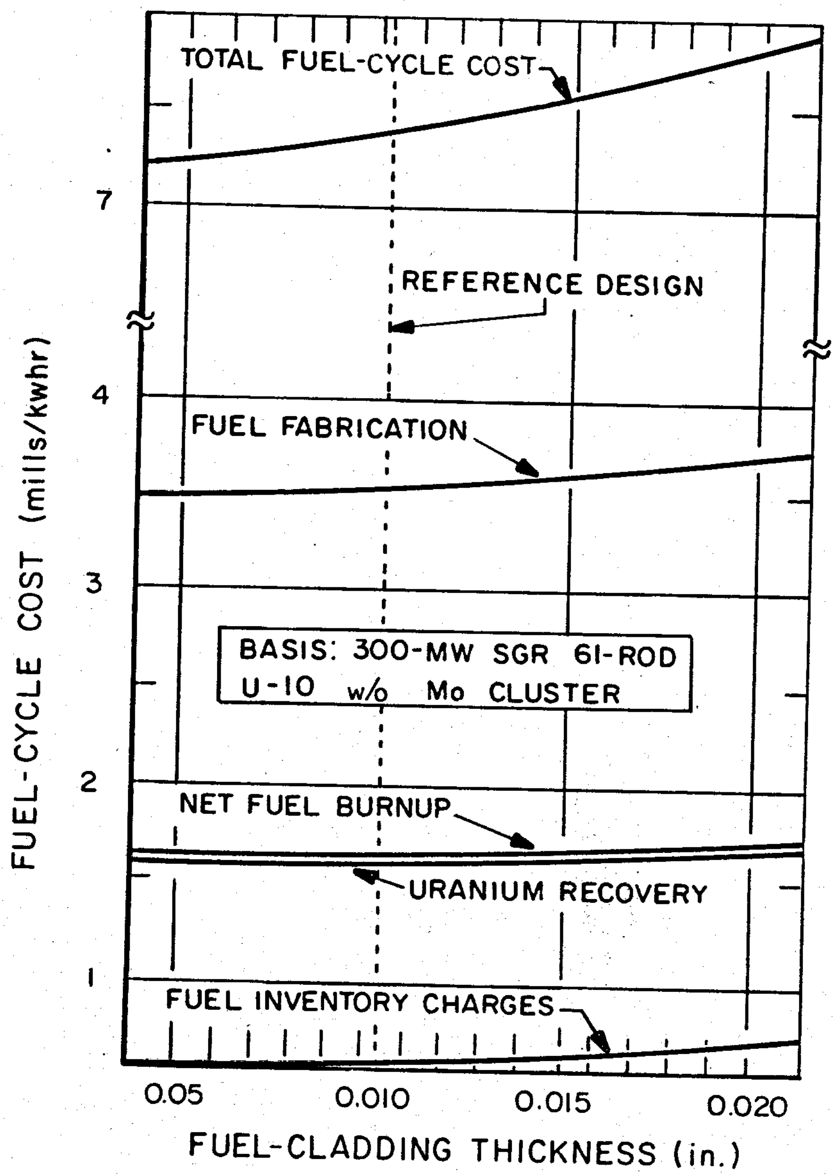


FIG. 22 - Variation in Fuel Cycle Cost with Fuel Cladding Thickness

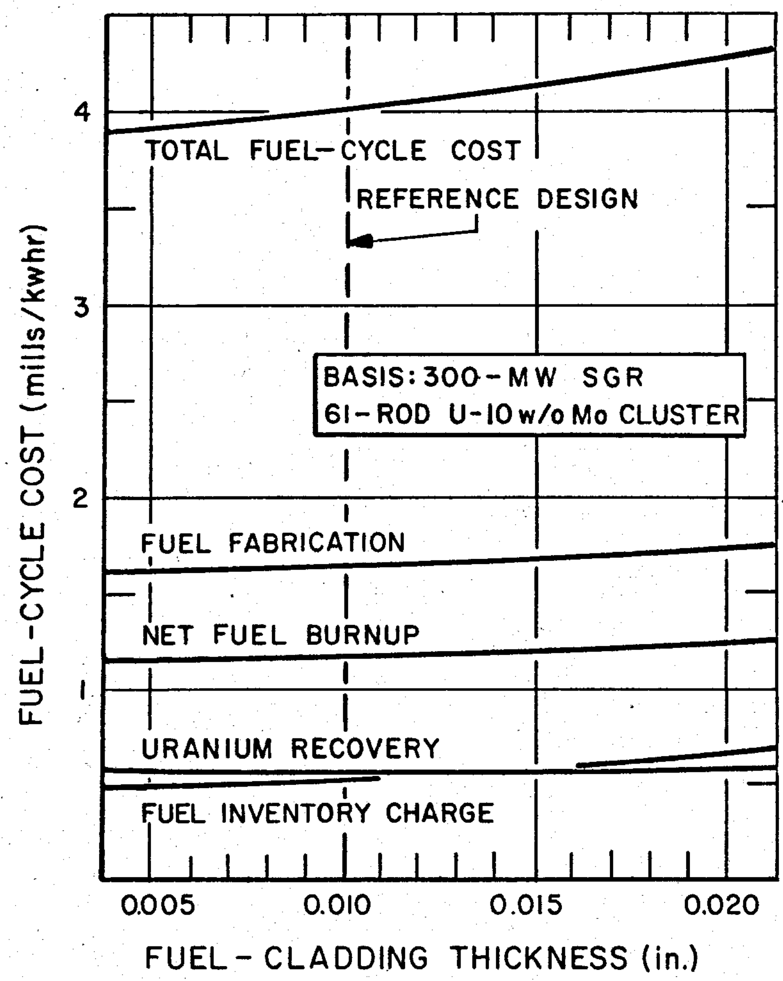
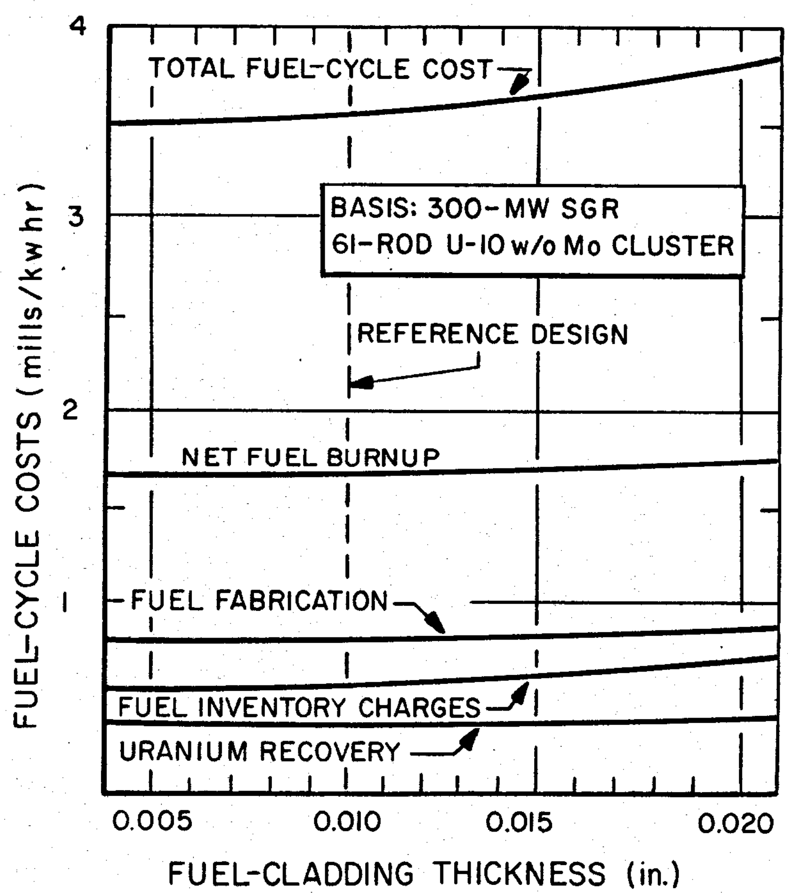


FIG. 23 - Variation in Fuel Cycle Cost with Fuel Cladding Thickness

(Average Fuel Burnup - 16,800 MWD/MTU)



C.4 Variation in Moderator Can Thickness and Material

A disadvantage of the Sodium Graphite Reactor of current design is the relatively large amount of stainless steel structural material introduced into the core by the moderator element canning. Zirconium and zirconium alloys have frequently been considered for this application. Reactor designs are now underway to eliminate a large portion of this structural materials.

To indicate the significance of the moderator can thickness on power generation costs a study was made varying the stainless steel moderator can thickness between 0.016" and 0.024". The effect of moderator can thickness on core reactivity is indicated in Figure 24.

The effect on fuel cycle costs of the enrichment variation with moderator can thickness is indicated in Figs. 26, 27 and 28. This variation in moderator can thickness is expected to make a negligible variation in capital costs of the plant.

FIG: 24 - Variation in Core Reactivity with Moderator Canning Thickness

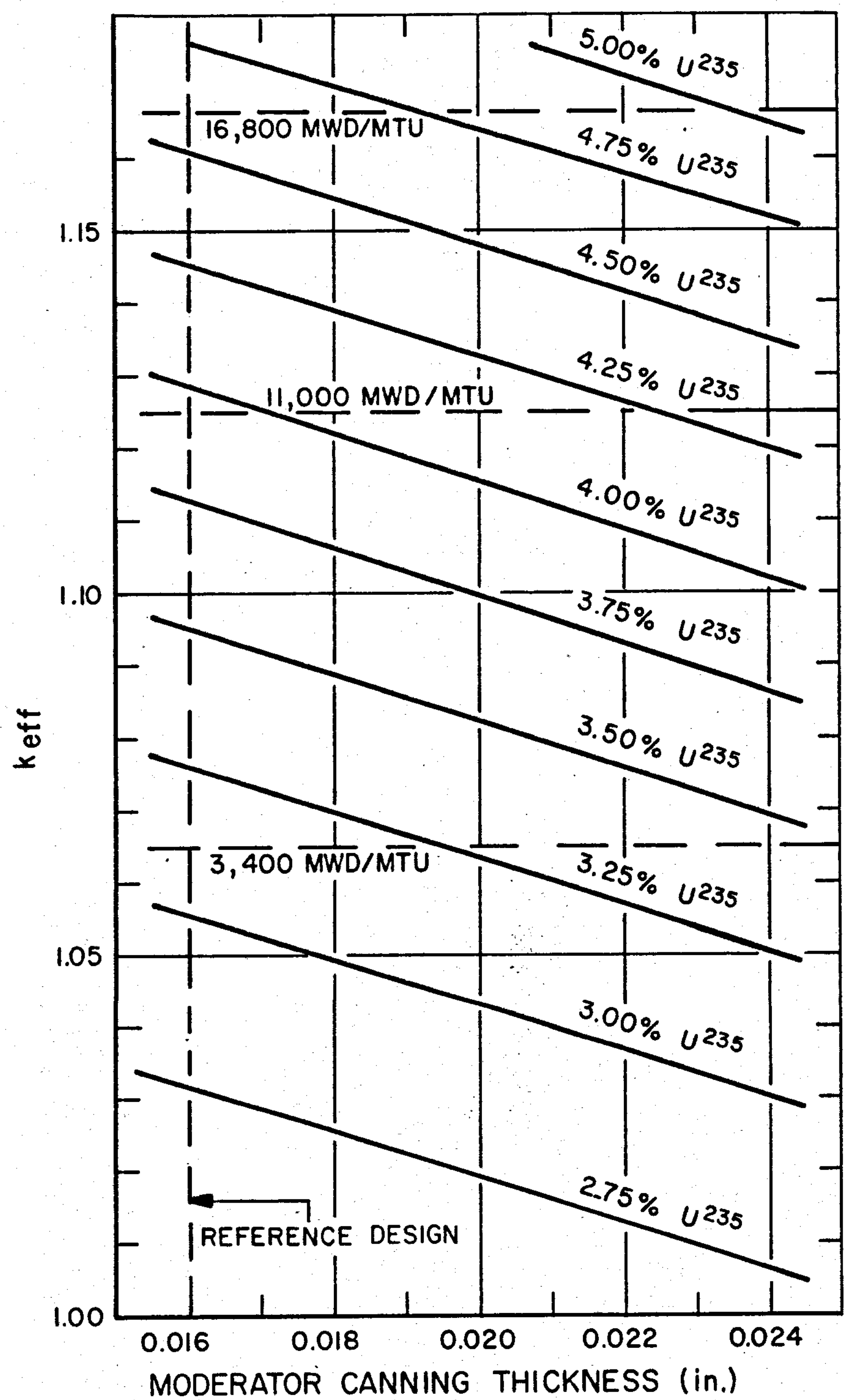


FIG. 25 - Variation in Fuel Enrichment with Moderator Canning Thickness

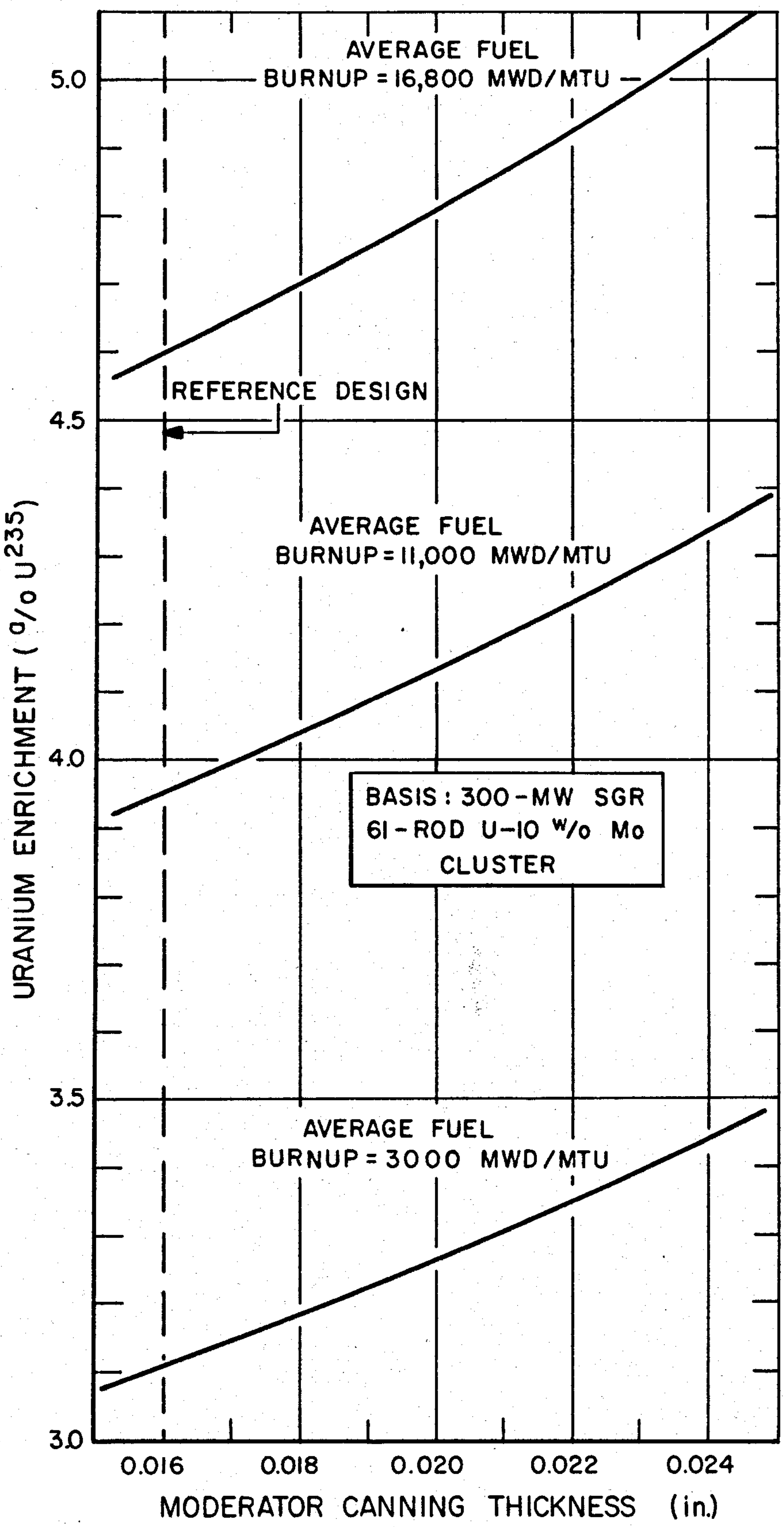


FIG. 26 - Variation in Fuel Cycle Cost with Moderator Canning Thickness

(Average Fuel Burnup - 3400 MWD/MTU)

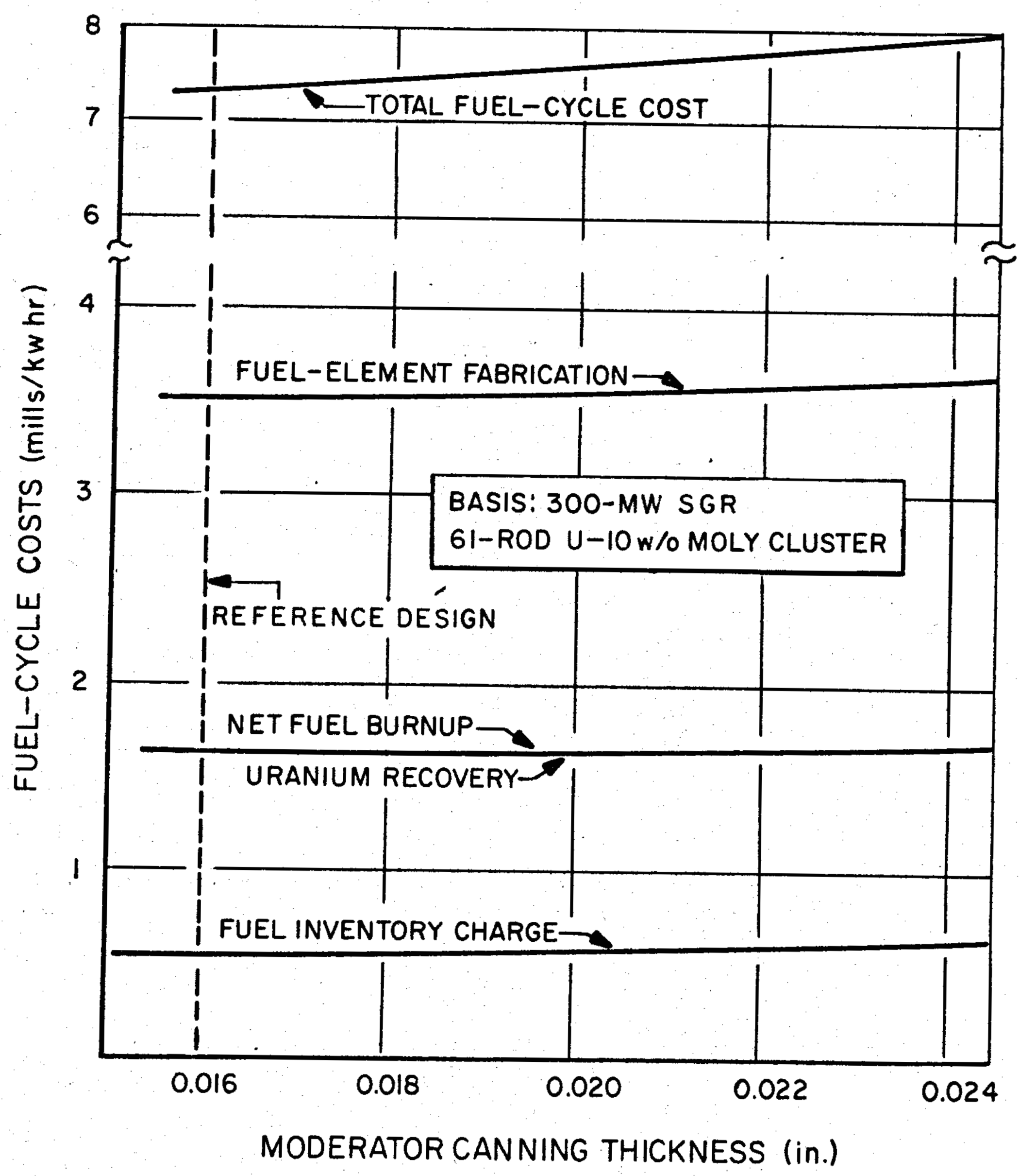


FIG. 27 - Variation in Fuel Cycle Costs with Moderator Canning Thickness
(Average Fuel Burnup - 11,000 MWD/MTU)

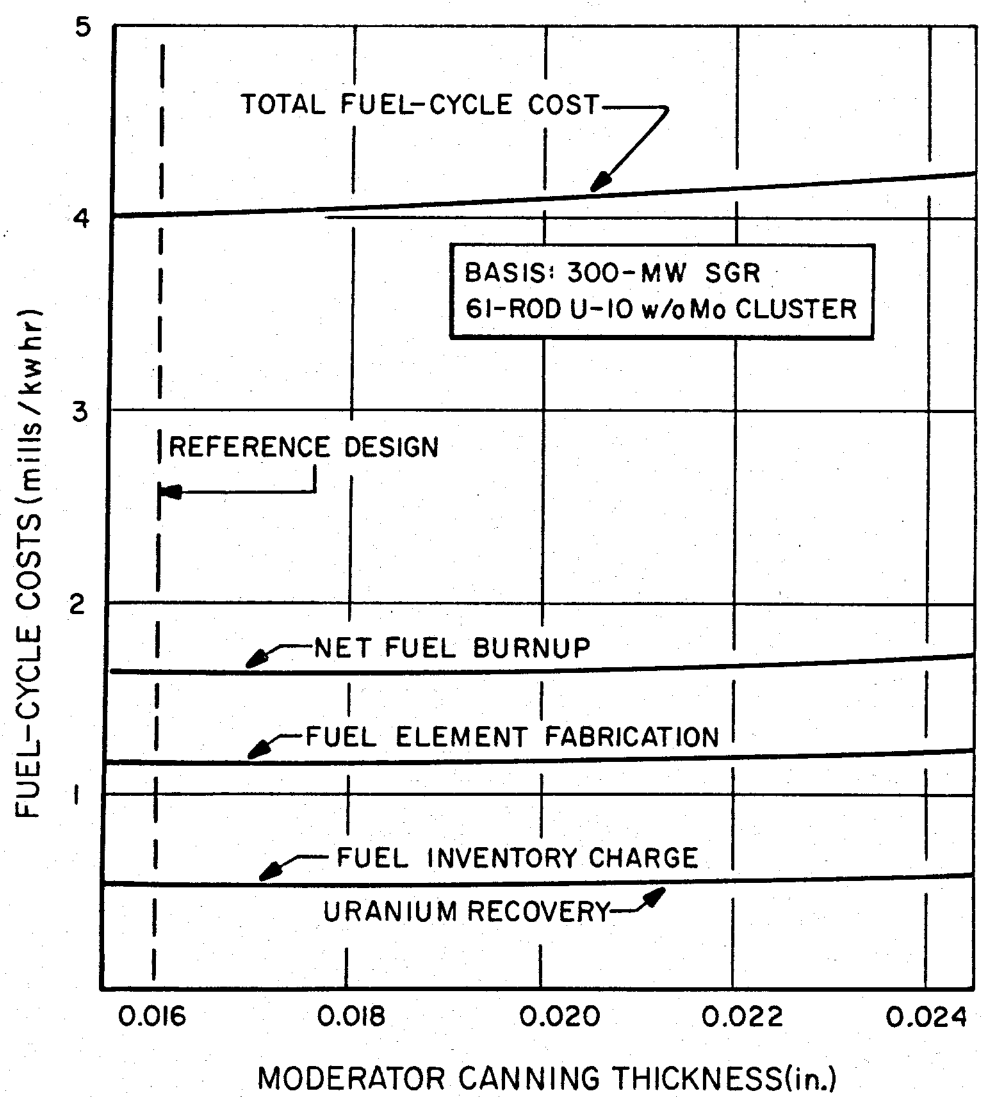
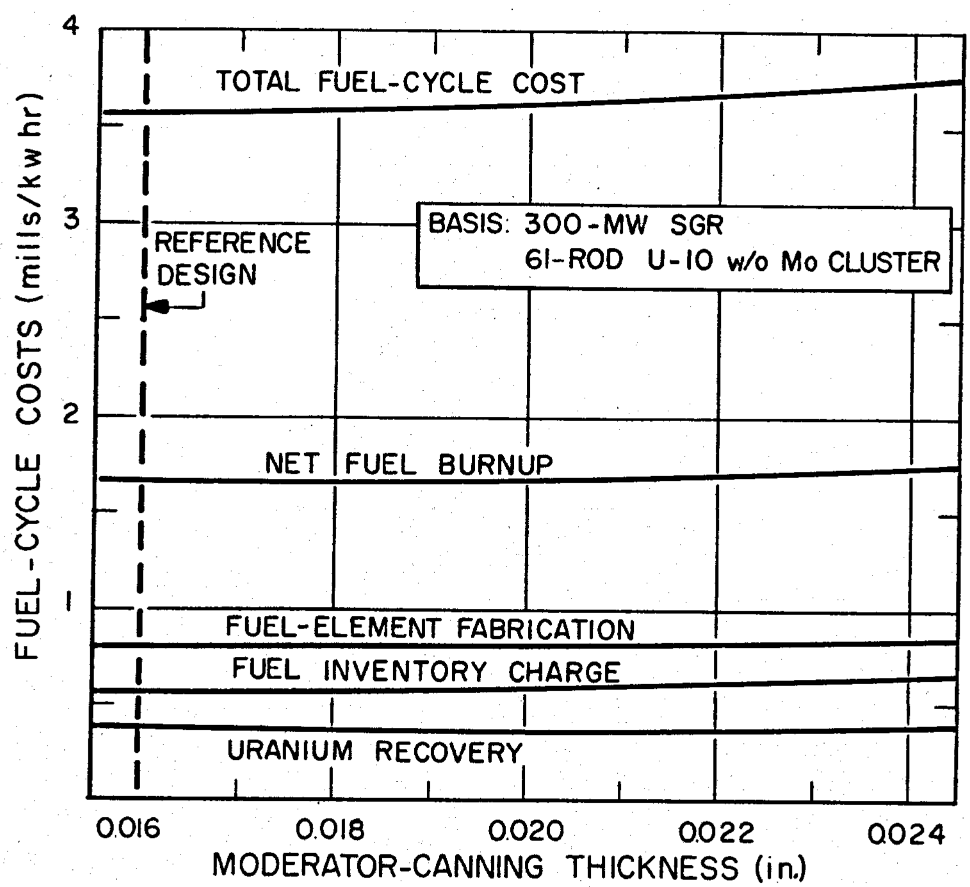


FIG. 28 - Variation in Fuel Cycle Costs with Moderator Canning Thickness

(Average Fuel Burnup - 16,800 MWD/MTU)



C.5 Comparison of Zirconium and Stainless Steel Moderator Canning

The use of structural materials with low neutron cross sections have been investigated for possible use as moderator element canning material in SGR's. Zirconium has been successfully used in the SRE. Zirconium alloys are being investigated for use at the higher coolant temperatures proposed for the Advanced SGR.

Figure 29 indicates the reduction in fuel enrichment possible with the use of zirconium moderator canning. This lower fuel enrichment reduces fuel inventory charges and the net fuel burnup charges. This difference in total fuel cycle costs is indicated in Figure 30.

The net reduction in fuel cycle costs is approximately 0.3 mills per kw hr at all burnups. This fuel cycle cost reduction if capitalized at 80% plant factor and 14% annual capital charges justifies an increase in capital investment of $4,500,000 or an increase of $2,900,000 in direct construction cost. Approximately 47,400 lbs of zirconium would be required in the moderator and reflector elements in the reference design. The "break even" point on the zirconium versus stainless steel cans is approximately $60 per pound of zirconium canning.

The cost of zirconium moderator cans has not been estimated but it is expected to be less than the "break even" point of $60 per pound. At the present time alloys of zirconium are under development which may have the strength require at SGR and advanced SGR coolant temperatures.

FIG. 29 - Variation in Required Fuel Enrichment With Average Burnup of Fuel Removal for Stainless Steel and Zirconium Moderator Element Canning

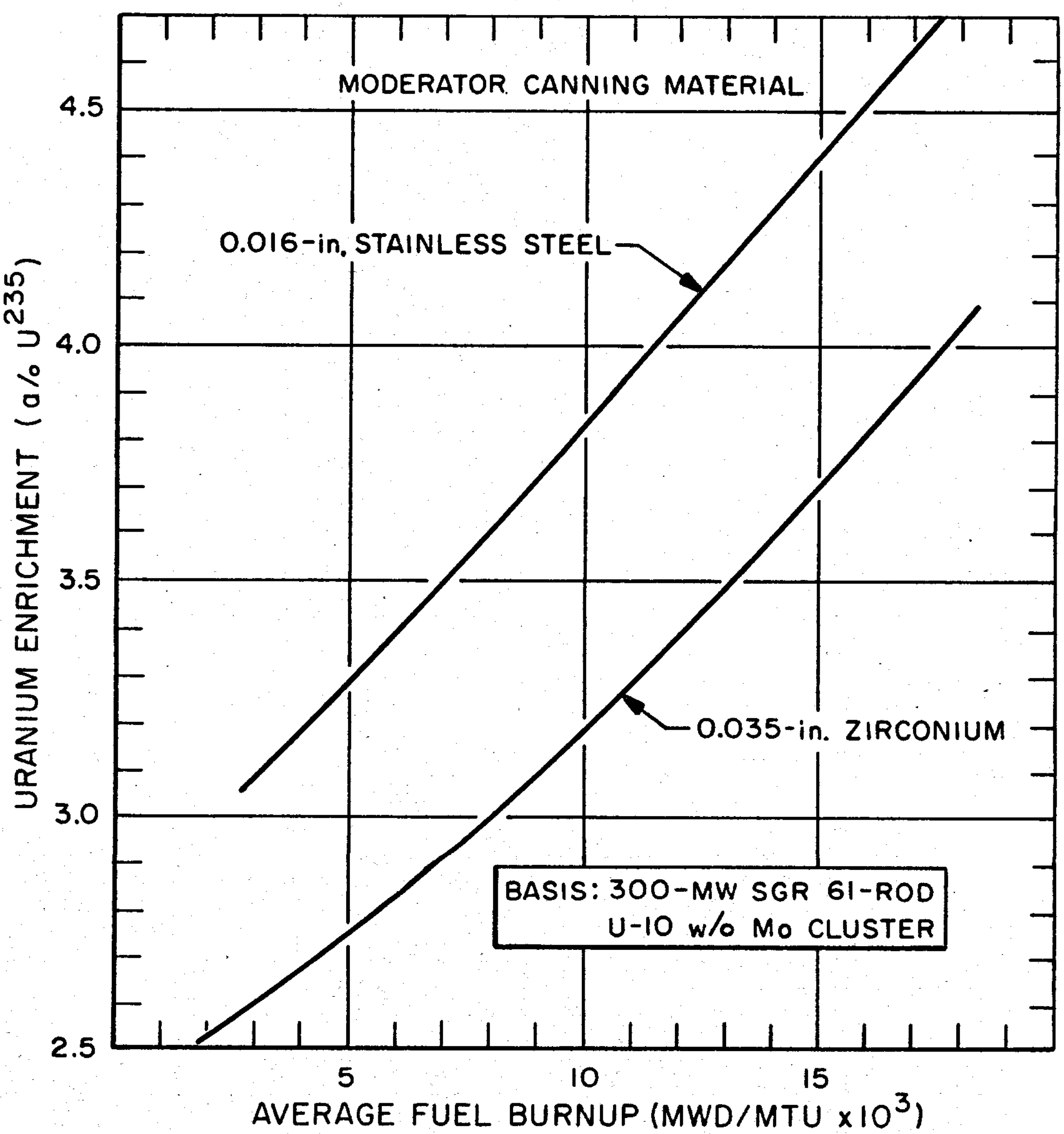
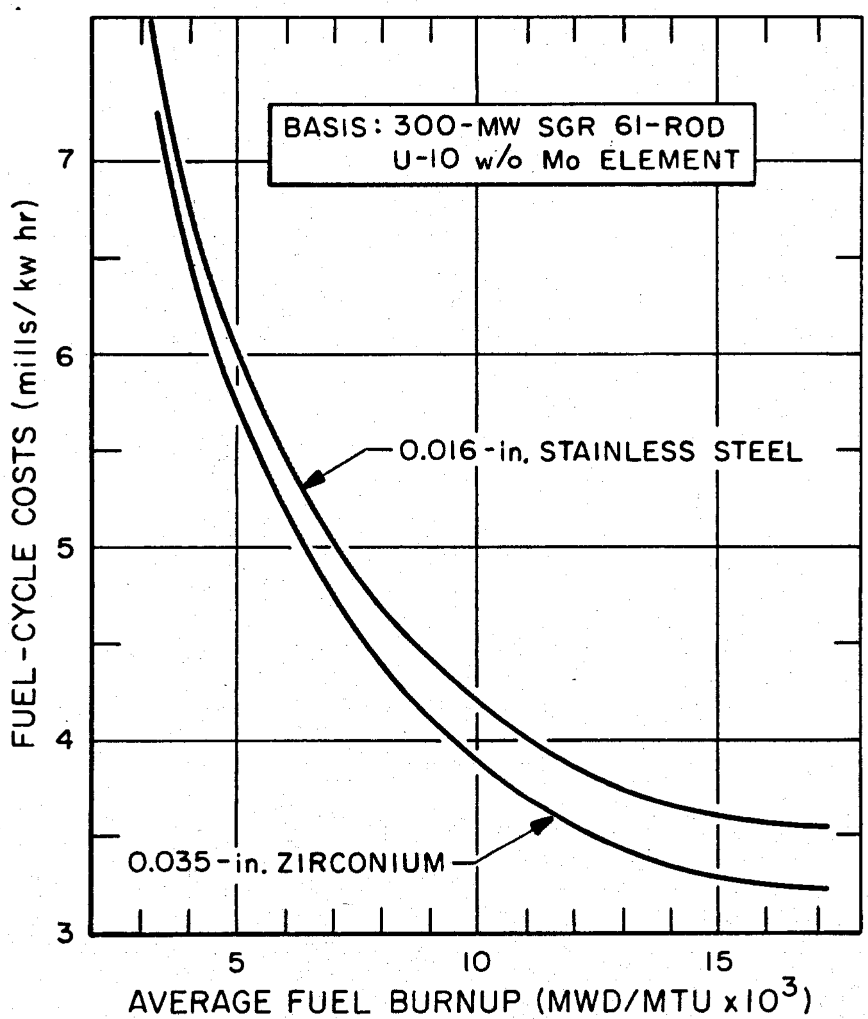


FIG. 30 - Variation in Fuel Cycle Costs with Average Fuel Burnup - Stainless Steel and Zirconium Moderator Canning



C.6 Variation in Sodium Flow Area

In the reference design (300 MW SGR) the maximum sodium velocity is 19.5 feet per second and the pressure drop across the core is 12.9 psi. This sodium velocity and pressure drop present no engineering problems. The four primary pumps are each rated 725 horsepower. An increase in sodium flow area per channel decreases the sodium velocity, pressure drop and pumping power. This decrease in pumping power increases the net plant output and net plant efficiency.

The increase in sodium flow area also introduces additional neutron "poison" into the reactor core. An increase in fuel enrichment is required to maintain core reactivity.

The variation in core reactivity with sodium flow area is indicated in Figure 31. The sodium flow area over the range studied (6 in^2 to 12 in^2) has relatively little effect on core reactivity.

The variation in fuel cycle costs with sodium flow area is indicated in Figures 33, 34, and 35. The net variation is less than 0.10 mills per kw hr, which is probably within the accuracy of this calculation.

A decrease in sodium flow area to 6 in^2. would require a 1250 hp increase in primary pumping capacity. Since the primary pumping power requirement is approximately 1% of the total power output of the turbo-generator this variation in pumping power has a negligible effect on net plant efficiency. The net effect on power costs would be less than 0.1 mill per kilowatt hour.

The capital cost of the additional pump capacity required with the smaller flow areas would also make a minor increase in power generating costs (less than 0.1 mills/kw hr).

For the 300 MW SGR it appears that the sodium flow area can be determined by considerations other than variation in power generating cost if kept within the limits of 6 and 12 in^2. The variation, in each of the various factors of power generating costs affected is small. In addition, any increase in power generating costs caused by increased pumping power is partially or completely compensated for by a decrease in fuel cycle cost.

FIG. 31 - Variation in Core Reactivity with Sodium Flow Area

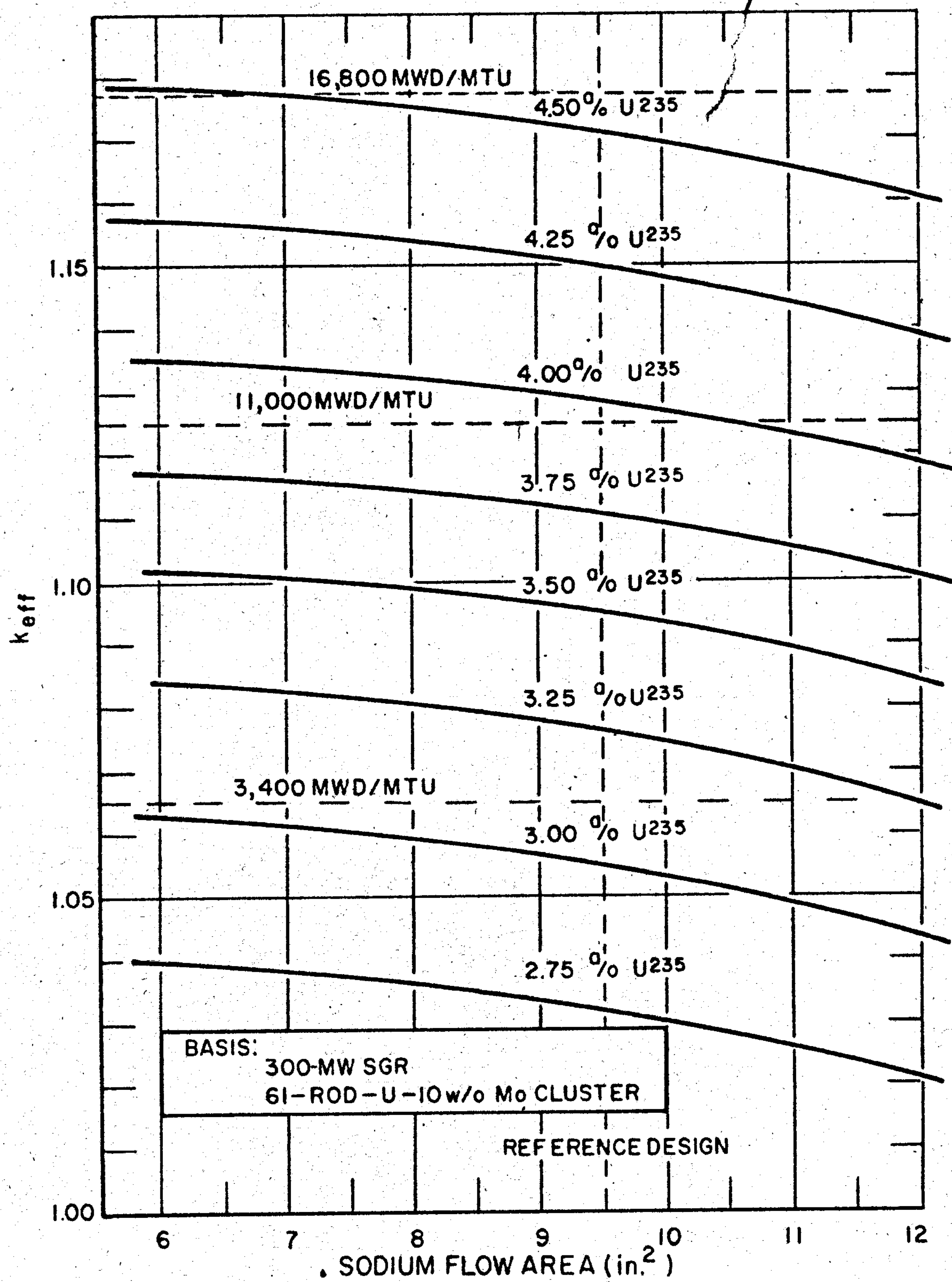


FIG. 32 - Variation in Fuel Enrichment With Sodium Flow Area

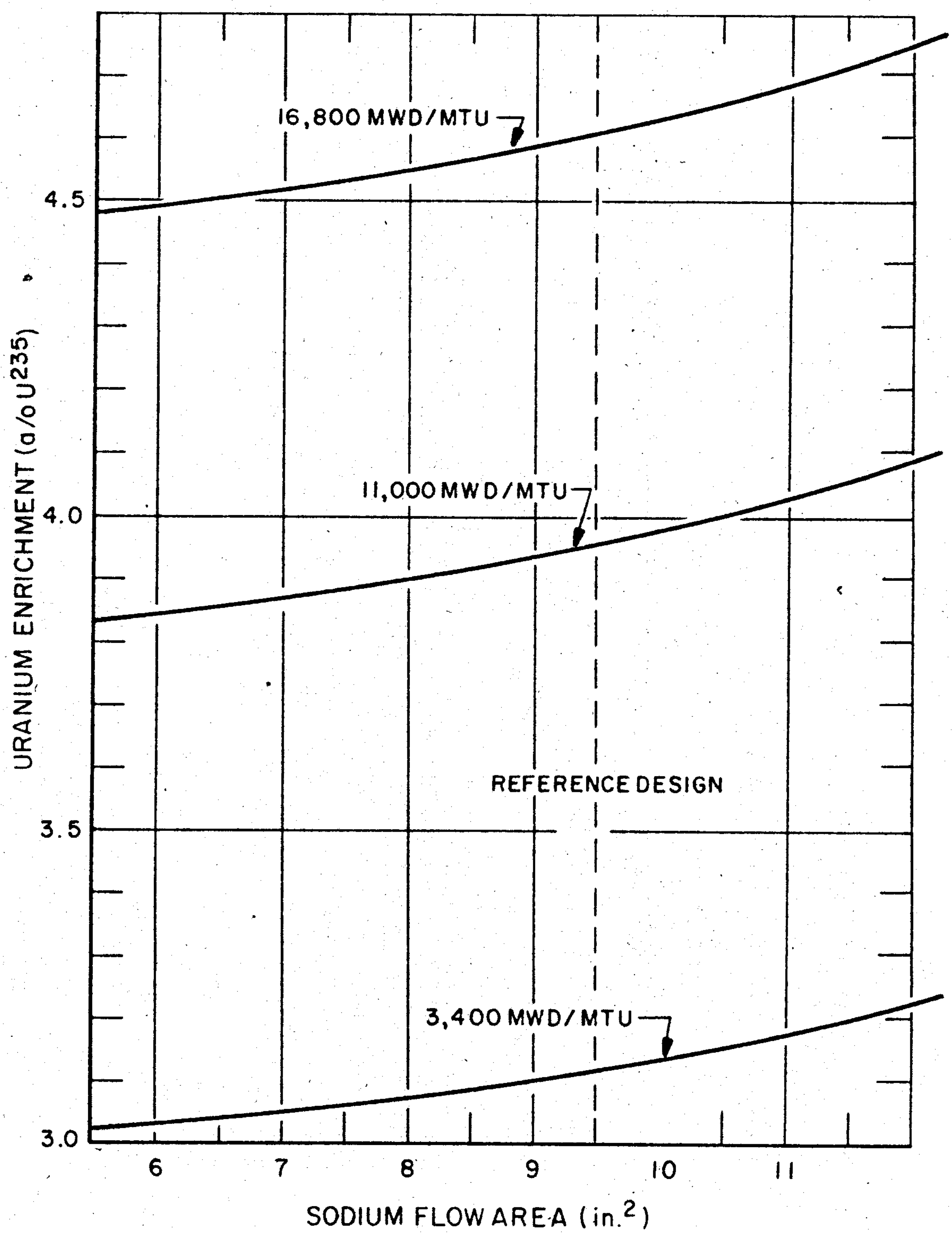


FIG. 33 - Variation in Fuel Cycle Costs with Sodium Flow Area

(Average Fuel Burnup 3400 MWD/MTU)

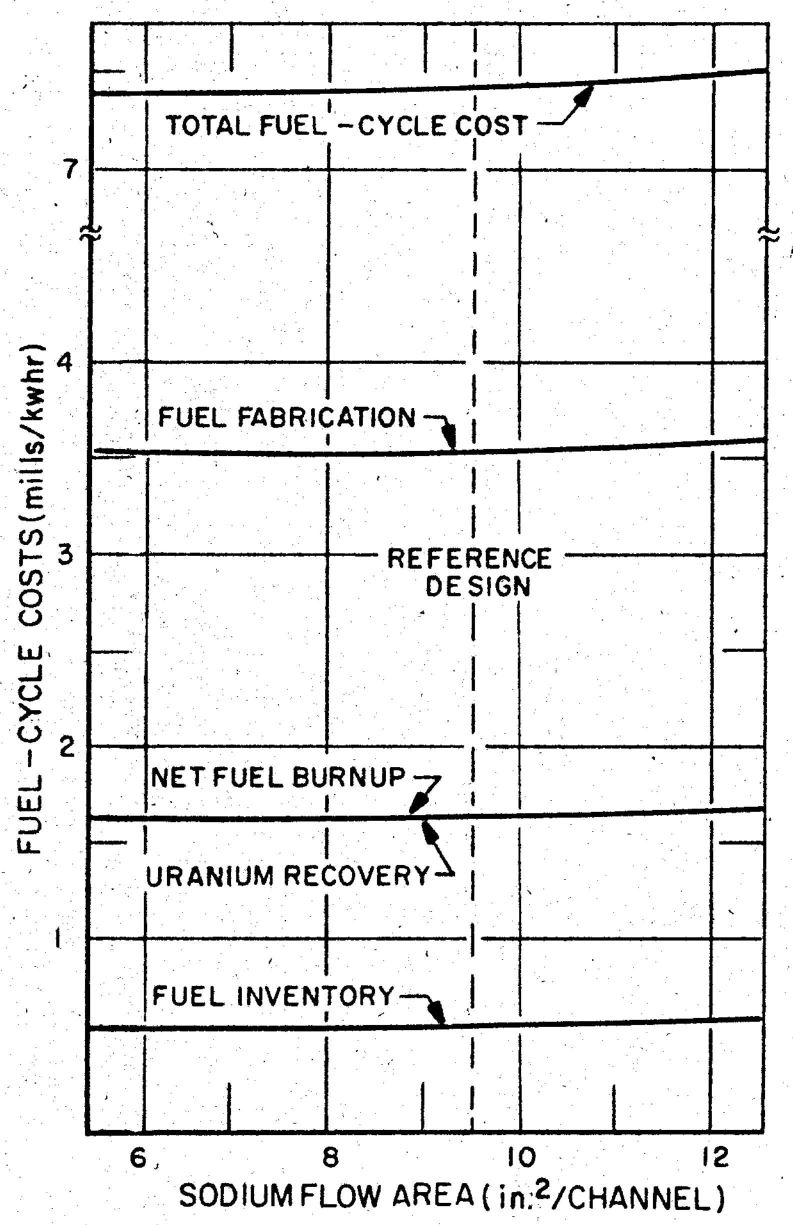


FIG. 34 - Variation in Fuel Cycle Costs with Sodium Flow Area

(Average Fuel Burnup - 11,000 MWD/MTU)

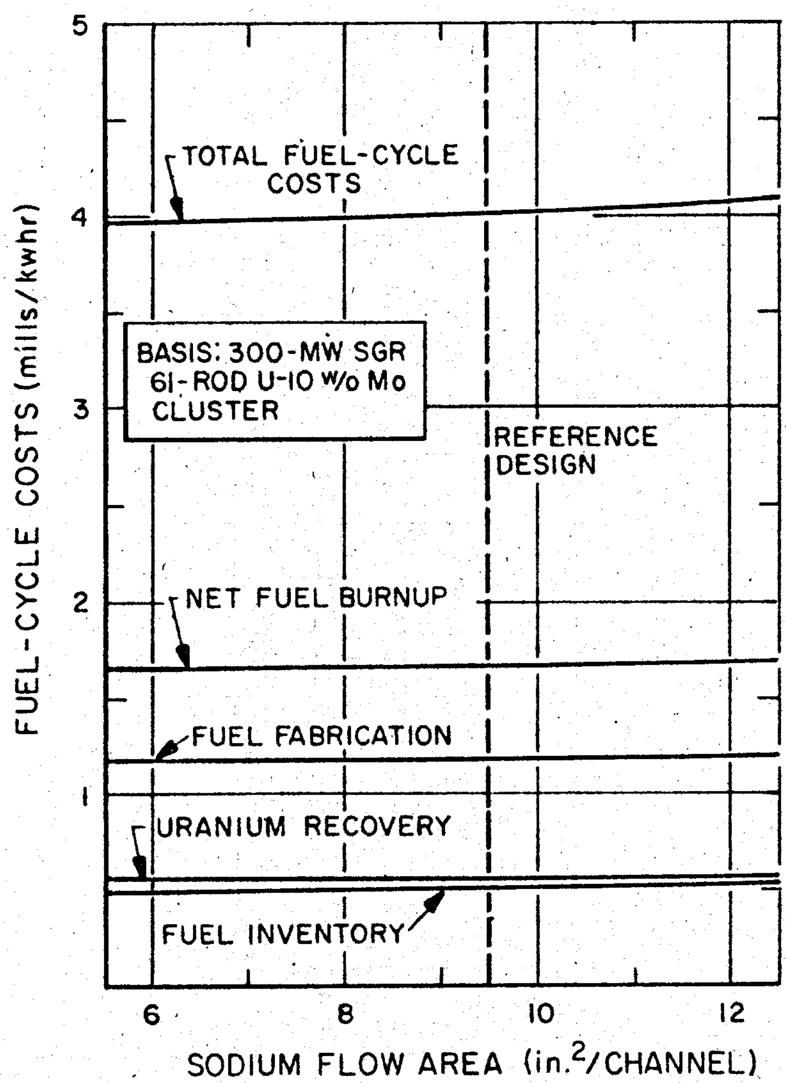
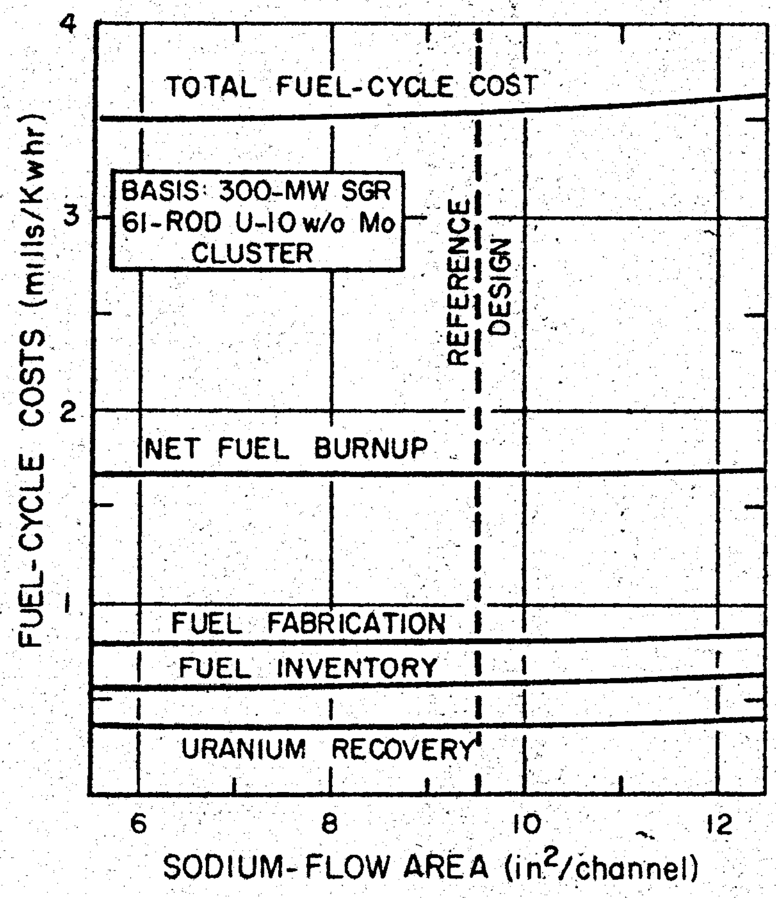


FIG. 35 - Variation in Fuel Cycle Costs with Sodium Flow Area

(Average Fuel Burnup - 16,800 MWD/MTU)



C.7 Comparison of UC and U-10 w/o Molybdenum Alloy

In the continuing effort to make further improvements in sodium graphite reactors an essential item is the development of a high temperature, high thermal conductivity fuel, which can be carried to high burnup.

Uranium monocarbide (UC) with a melting point of 4300°F, a thermal conductivity of approximately 14 Btu/hr-ft-°F and a uranium density of approximately 13 gms/cc, appears to be an excellent fuel for high temperature sodium cooled reactors. High burnup limitations, essential to any fuel, can be proven only after extensive irradiation tests.

Uranium monocarbide promises to reduce power generating costs in several ways. Two of the cost reductions which can be achieved with UC are as follows:

1. High power density in the fuel - fuel inventory charges for the reactor core loading are a significant part of power generating costs (See Table II). This cost would be even more important if the uranium were sold to the utility instead of leased at the relatively low rate of 4% per annum.

2. Higher thermal efficiency - Although some increase in sodium temperature is possible with present technology, significant increase in sodium temperature (to 1200°F) requires a fuel, such as UC, which can operate reliably at higher temperatures. Higher sodium temperatures will permit adoption of modern, efficient steam cycles in nuclear plants.

In this study sodium temperature and all conditions external to the reactor were not changed. A higher power density in the fuel was achieved by reducing the number of rods (from 61 to 19), reducing fuel cross sectional area (from 5.2 in^2 to 3.7 in^2 per element) and reducing core height (from 15 ft to 12 ft).

A brief tabulation of significant core parameters is indicated in Table V.

Since sodium temperatures were not varied in this study, the improvement in thermal efficiency was not considered. The composite improvement in fuel cycle costs possible with UC is indicated in NAA-SR Memo 4199, "SGR Development Program - Objectives and Estimated Costs."

A comparison of fuel enrichment between UC and U-10 w/o Mo is indicated in Figure 36. The enrichment required for UC is lower for all fuel burnups. This lower enrichment and the lower fuel inventories makes a significant reduction in fuel inventory charges. Figure 37 indicates fuel cycle costs for UC when used in a "canned moderator" SGR.

In addition to the fuel cycle cost reduction, the smaller core size reduces capital costs of the reactor core and reactor structure approximately $550,000.

Table V

300 MW SGR

Comparative Values of Significant Core Parameters

UC Core and Reference Reactor

		Revised UC Core	Reference Reactor
Reactor Thermal Power		884 MW	
Electrical Output		300 MW	
Core Dimensions			
Height	ft	12	15
Diameter	ft	13.7	15.5
Moderator Elements			
Length	ft	16	19
Across Flats	in	14	16
Fuel Elements			
Fuel Material		UC	U-10 w/o Mo
No of Rods		19	61
Fuel Slug diameter -in.		0.50	0.33
Fuel Density	gm/cc	13.6	17.1
U Density	gm/cc	13	15.4
Fuel Cross Sectional Area			
Rod	in^2	0.196	0.0839
Element	in^2	3.73	5.2
Fuel Weight per element			
Total	kg	117	263
Uranium	kg	111	237
No. of Fuel Elements in Core		220	214
Core Loading			
Total Fuel	kg	25,800	56,200
Uranium	kg	24,500	50,600

FIG. 36 - Variation in Required Fuel Enrichments with Average Burnup of Fuel Removed for U-10 w/o Mo and UC Fuel

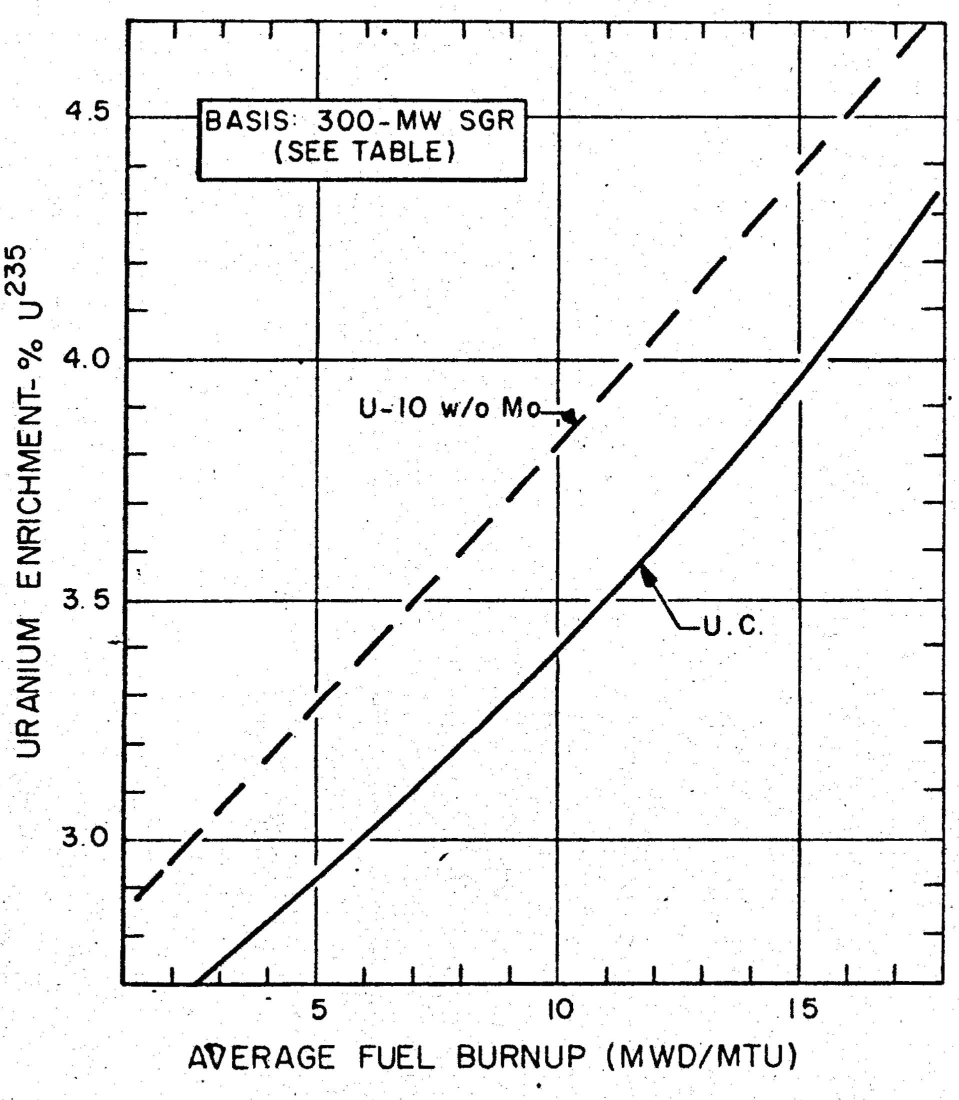


FIG. 37 - Variation in Fuel Cycle Costs with Average Fuel Burnup for a Uranium Carbide Fueled Sodium Graphite Reactor