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Lewis Research Center
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SUMMARY

As a first step in a comparison of the relative performance of nuclear reactors and of other thermal sources for space power systems, criticality calculations were made for small, cylindrical, lithium (Li)-cooled reactors which use tungsten (W) as the structural material.

The nuclear fuels considered were plutonium-239 (Pu^{239}), uranium-233 (U^{233}), and uranium-235 (U^{235}). These fuels were assumed to be in the form of pure metals, nitrides, and oxides. The radial neutron reflectors considered were beryllium oxide (BeO), W, molybdenum (Mo), and carbon (C).

The calculations revealed that the smallest cores are those fueled with metallic U^{233} and Pu^{239}. These cores can be as small as 11 centimeters in diameter and 30. 5 centimeters long; however, at desired operating temperatures these cores would contain molten fuel.

The smallest solid fuel core contains uranium-233 nitride ($U^{233}N$) in a core 13. 5 centimeters in diameter and 30. 5 centimeters long. The reactor diameters increase about 2. 5 centimeters when the oxide fuel is used in place of the nitride. Cores fueled with U^{235} are considerably larger than those fueled with U^{233}.

A $U^{233}N$ core of the same length (30. 5 cm) and fuel loading, but radially reflected by W instead of BeO, is 16. 5 centimeters in diameter. This diameter can be decreased to about 15 centimeters by increasing the overall core volume fraction of $U^{233}N$ from 45 to 47. 5 percent. Calculations show that the radial power distribution of such a core can be made uniform to within ±10 percent without a significant reactivity loss. An examination of the heat-transfer limits indicates that this core should be capable of transmitting more than 1 megawatt of thermal power to a Li coolant.

INTRODUCTION

Reference 1 discusses some of the areas of technology important to the development of fast-spectrum, lithium (Li)-metal-cooled, nuclear reactors capable of producing 2.5 megawatts of thermal power for 20 000 hours while enclosed in a full (4π) man-rated shield. Limiting the fuel burnup is a major consideration in choosing a reactor size. However, the use of replaceable reactors of shorter core life could relax the requirements on fuel burnup and thereby permit the use of a smaller reactor in which the thermal power level would be determined by reactor size. A separate, fully man-rated shield, which is not part of the replaceable reactor or its primary loop, could accommodate reactor replacement in, for example, a manned orbiting laboratory.

Reactors smaller than those discussed in reference 1 also could be used for unmanned electric propulsion missions where only shadow shielding is required. A reactor length-to-diameter ratio greater than 1 would be desirable in order to reduce shadow shield requirements. For this application, the large fuel burnup probably would necessitate venting of the fuel elements to release the fission product gases that would be generated during extremely long core life. The advantage of nuclear reactors over other thermal sources in space power systems is high power-density capability. As a first step in a comparison of the potential performance of nuclear reactors with the performance of other sources for space power systems, a preliminary study of criticality-limited, Li-cooled reactors is presented.

An argument against the early consideration of very small reactors is, "What happens when we need much more power?" The answer to this is that the future electrical power requirements in space are estimated to be small when compared to the capabilities of nuclear reactor systems. Also, reactor cores are reasonably scalable as long as the power density and operating temperatures are not increased.

Reactor Models

From the standpoint of criticality the smallest reactor cores are metallic-fuel spheres with thick neutron reflectors. While the addition of a moderator, such as water or polyethelene, can decrease the critical mass, moderation will increase the critical size of the sphere, as shown in figure 1, which was taken from reference 2. From a heat-transfer point of view, however, spheres of metallic fuel are impractical; therefore, Li-cooled, cylindrical reactors are considered herein.

Two reactor models were assumed to compare the following nuclear fuels: metallic plutonium-239 (Pu^{239}), the metal, nitride, and oxide of uranium-233 (U^{233}, $U^{233}N$, and $U^{233}O_2$), and the nitride and oxide of uranium-235 (U^{235} and $U^{235}O_2$). The nitrides and oxides were homogeneously mixed with tungsten (W) in the fuel material. The pure metal fuels, which would be molten at desired operating temperatures, were contained in thick-walled (0.4-cm) W cylinders or pins with outside diameters of approximately 2.5 centimeters. Furnace tests have shown that molten U can be contained by cylinders of single-crystal tungsten. However, in order for such reactors to have practical core lifetimes, the W mass transport problem in a temperature gradient would have to be solved. In order to achieve the smallest diameter core, the minimum Li coolant fraction for a core using a pin-type element (i.e., pins touching) was used. This coolant fraction was 10.3 percent when account was taken of the extra flow area at the periphery of the core. This is too small a coolant fraction for a high power density in a pin-type core, but it does give the smallest possible core for low power levels. For comparison purposes, this same coolant fraction was also used to cool the cermets which were assumed for the other reactor model. In this model the nitride- and oxide-cermet fuel material was solid with a triangular array of approximately 0.5-centimeter-diameter coolant holes at a 1.4-centimeter spacing. Each of the holes was clad with 0.038 centimeter of W. The core average volume fraction of U in the fueled W cermets was allowed to vary from 0.45 to 0.55. By setting the fuel fraction limit at 0.55 some contingency for fuel zoning was allowed. However, because of limiting-strength and thermal-conductivity considerations, the maximum fuel fraction in the cermet in the core should not be greater than 0.60. All cores were assumed to be 30.48 centimeters long, which made the length-to-diameter ratio of the core larger than 1 for the smaller cores. For shadow shielding situations this ratio may be desirable.

Method of Analysis

Calculations of published critical experiments containing the same fuels and materials that are to be considered here allow a check of the validity of calculational techniques and of the adequacy of available neutron cross sections. For this report, neutronic calculations were performed on spherical critical experiments which are reported in references 2, 3, and 4.

For these calculations the discrete-angle, multigroup, neutron transport program TDSN (ref. 5) was used. The P_1 approximation for neutron scattering and S_4 discrete angle segmentation was used.

Multigroup cross sections used in the analysis were obtained from the GAM II program (ref. 6), which averages cross sections over the fission and slowing down spectrum. Separate GAM II calculations were performed for the homogenized core and reflector materials by using 13 neutron energy groups. The lower energy of the lowest group used was 0.662×10^{-9} joule (0.414 eV); in the W-reflected, fast reactors, the number of neutrons with energy less than 0.662×10^{-9} joule (0.414 eV) is negligible.

The neutron thermalization program GATHER II (ref. 7) was used for averaging cross sections for five thermal groups when a thermalizing reflector of beryllium (Be) was used. The reflector flux spectrum was used to average the microscopic cross sections for the core since the only thermal neutrons in the system are those which slow down in the reflector before returning to the core.

The critical experiments which were calculated and the results of the calculations are listed in table I. The density of each of the fuel materials used, as well as the ratio of the atom density of each material relative to UO_2, is given in table II. A comparison of the values of k_{eff} (neutron multiplication factor) which are listed in table I with an experimental value of 1.00 reveals that the calculated multiplication factors for the W-reflected spheres are about 2 percent too high and that those for the Be-reflected core are about 3 to 4 percent too high. The calculational error (6.5 percent) for the beryllium oxide (BeO) reflected core is the greatest.

For this report the differences between the calculated values of k_{eff} and the experimental value of 1.00 were assumed to comprise a constant bias; therefore, 0.065 was subtracted from the values of k_{eff} of the BeO-reflected reactors, and 0.020 was subtracted from the values of k_{eff} of the W-reflected reactors.

A further investigation of the discrepancy with a BeO reflector showed that by increasing the number of groups to 35 and the order of the calculation to S_8 the discrepancy for a P_1 calculation was reduced to 3.8 percent. However, this is still too large a discrepancy for reactor design purposes. Several possible causes for this discrepancy will have to be checked and isolated, possibly by experimental means. Nevertheless, the present biased calculations should be adequate for the purpose of this report.

The next step in the current investigation was to calculate the multiplication factors and power distributions for cylindrical reactors made up of the same materials as had been used for the spherical reactors. The cylindrical-geometry option of the TDSN program had been satisfactorily checked (ref. 8) by calculating Lewis ZPR-II critical experiments. Two-dimensional calculations would have required more computer time than could be justified for the survey calculations of this report; so a procedure was devised whereby only a few two-dimensional calculations were required.

The following is the procedure which was devised. A one-dimensional, 13-group, P_1 calculation for the W-reflected reactors was compared to a one-dimensional, 6-group, transport-corrected, P_0 calculation. The energy group structure is shown in table III. The 6-group, P_0 value of k_{eff} was only 0.62 percent greater than the 13-group, P_1 value. This 6-group P_0 model was then used to perform a synthesis of iterative, one-dimensional calculations. The radial leakage was first calculated by using a geometric axial buckling with an assumed reflector saving. Next, the axial leakage was calculated by using the calculated radial leakage. Then, a new radial leakage was calculated by using the calculated axial leakage. The final value of k_{eff} from this synthesis was only 0.81 percent greater than that from a two-dimensional, cylindrical-geometry, 6-group P_0 calculation. Similar calculations were made for the Be-reflected cores, and similar checks were obtained. The results of this series of calculations indicated that none of these model differences would significantly affect the relative results of this report, and that practically all of the calculations could be made with a one-dimensional radial model. In this model the axial bucklings were computed for the core and the reflector as $\left\{\pi/[H + 8.5 + 2\delta(E)]\right\}^2$, where H is the core height in centimeters, and $\delta(E)$ is the energy-dependent extrapolation distance, $0.71\,\lambda_{tr}$, where λ_{tr} (cm) is the neutron mean free path for the core and the reflector regions, respectively. An axial-reflector saving of 8.5 centimeters, as determined by comparing one- and two-dimensional cases, is assumed for all the one-dimensional cylindrical calculations. Axial reflectors were assumed to be 10.16 centimeters thick and composed of 85 percent W and 15 percent Li^7.

A one-dimensional, radial model of one of the cores was then used to investigate flattening the radial power distribution by increasing the fuel concentration in the regions of low power density and decreasing it in the regions of high power density. A maximum of four different fuel concentrations (i.e., zones) was used. The average fuel loading for the core was kept constant so that the cost (in reactivity) of the fuel zoning could be determined.

A life study, by means of the Vulcan program (ref. 9) and TDSN calculations of a radial model, was then performed on this zoned reactor to obtain an estimate of the reactivity required for 20 000 hours of operation and of the concomitant shift in power distribution.

CALCULATIONS AND RESULTS

Fuels

Calculations were made to determine the multiplication factor k_{eff} as a function of the diameter of 30.5-centimeter-long cylindrical reactors made up of different fuels and fuel materials. Results of these calculations for reactors which were assumed to be cooled with isotopic Li^7, which took up 10.3 percent of the core volume, are shown in figure 2. A 10.16-centimeter-thick radial reflector made of 95 percent BeO and 5 percent Li^7 was used in these calculations.

From a criticality standpoint, U^{233} and Pu^{239} are the best fuels for these small, fast-spectrum reactors, even with the reduced densities of the molten state. If the assumption is made that 6 percent excess reactivity ($\Delta k/k$) is a reasonable amount for burnup, for additional temperature defect above the melting point, and for all other reactivity losses, then it is seen that for the volume fractions considered 30.5-centimeter-long cores as small as 11 centimeters in diameter are possible.

The next best fuel material is $U^{233}N$ cermet. However, if $Pu^{239}N$ had been calculated, it probably would have proved to be better than $U^{233}N$ from the standpoint of reactivity. For these fuel volume fractions (10.3 percent Li^7 coolant, 86.5 percent fuel material, and 3.2 percent W cladding), 30.5-centimeter-long cores as small as 13.5 centimeters in diameter are possible.

The calculations indicated that the $U^{233}O_2$ cores would be about 3.5 centimeters greater in diameter than the $U^{233}N$ cores and that the use of U^{235} would require an even greater increase in diameter of both the oxide and nitride cores. With U^{235}, the nitride core would have to be about 27 centimeters in diameter, and the oxide core would have to be 38 centimeters in diameter. This 12-centimeter increase in diameter for the U^{235} cores over the U^{233} cores would result in a considerable increase in shield weight for a small reactor, even if the reactor were shadow shielded.

Reflectors

From a reactivity standpoint, BeO would make the best radial neutron reflector, but there are several reasons for considering some other refractory material:

(1) The use of a reflector material such as W, which can operate at the desired temperature and with coolants in the core, would allow locating the radial reflector inside the pressure vessel. The reflector then would be closely coupled to the core. And required cooling could be accomplished by the primary reactor coolant rather than by a lower temperature coolant loop.

(2) The use of a neutron reflecting material that significantly attenuates the core gamma flux as well as the neutron leakage flux would allow the shielding to be initiated as close to the core as possible.

(3) Heavy metal reflectors would not cause a large increase in the power density at the interface between the core and the reflector.

For these reasons calculations were made for W-reflected reactors as well as for BeO-reflected reactors. A representative comparison (fig. 3) shows that the core diameter must increase about 3.0 centimeters to keep the same neutron multiplication k_{eff} when W instead of BeO is used for the reflector. For a 10-centimeter-thick, BeO reflector, some thickness of neutron absorber would always have to be put between the core and the BeO to reduce the high power density at the edge of the core. A calculated example of this high power density at the edge of the core is shown in figure 4. Removal of this power-density peaking always results in a loss in reactivity; thus, the core diameters of actual BeO-reflected cores would be somewhat larger than shown in figures 2 and 3.

Some calculations were also made to check the relative reactivity of molybdenum (Mo) and carbon (C) as neutron-reflector materials. These calculations were made for two core sizes. The results are compared with those for a W reflector in table IV.

Core Power Flattening

Figure 3 shows that a W-reflected, 15.2-centimeter-diameter, $U^{233}N$ fueled core has a multiplication factor of about 1.08 when the fuel material volume fraction is 47.5 percent of the core. This corresponds to a fuel material composition of 55 volume percent $U^{233}N$ and 45 volume percent W. This core was selected for calculations to determine how uniform the radial power distribution could be made by allowing the fuel concentration to vary in different parts of the core. In general, the more zones that are allowed and the greater the maximum fuel concentration, the better the power flattening that can be obtained. But in order to limit the scope of this study, a maximum of four different fuel concentrations was allowed, and, from a metallurgical standpoint, the maximum $U^{233}N$ allowed in the fuel material was 60 percent. In these calculations the average fuel loading over the whole core was kept constant so that the cost (in reactivity) of the fuel zoning could be determined. The results of the last of a series of iterative calculations, along with the radial power distribution of the unzoned core, are shown in figure 5. By using four fuel zones with a maximum concentration of 60 percent it is possible to reduce the radial peak-to-average power from 1.25 for the unzoned case to less than 1.10. The surprising part of this calculation is that the cost in reactivity is only about 0.7 percent.

A life study was then performed on this zoned core. A core power of 1.25 megawatts for 20 000 hours was assumed. For this, 3.1 percent of the initial U^{233} was consumed. The calculated reactivity loss was 1.8 percent $\Delta k/k$. The fuel depletion rate throughout the core was quite linear with time; so the change in local-to-average radial power distribution was less than 1 percent.

Heat-Transfer Capabilities

In order to estimate the amount of power that may be taken from these small, high-temperature reactors, this same $U^{233}N$, fuel-zoned core was used as a specific example. This core was 15.2 centimeters in diameter and 30.5 centimeters long. The average volume fractions of core materials were 47.5 percent $U^{233}N$, 42.2 percent W, and 10.3 percent Li^7. Many core geometries and fuel-element configurations would have fit these material volume fractions, but for the purposes of this example the following model was chosen. The $U^{233}N$ - W cermet fuel material was assumed to be solid with a triangular array of 0.48-centimeter-diameter holes for the Li coolant at a 1.43-centimeter spacing. Each of the coolant holes was clad with 0.038 centimeter of W.

With this model, consider each of the five limiting parameters that were discussed in reference 1:

(1) Li pressure drop through the core
(2) Li temperature increase
(3) Heat flux to the Li
(4) Fuel temperature
(5) Fuel burnup

In regard to the Li pressure drop, the flow area through this reactor model would be about 19 square centimeters. If an average flow velocity of 7 meters per second were arbitrarily selected, then the mass flow rate through the core would be about 7 kilograms per second, and the frictional pressure drop through the core would be about 1.2 newtons per square centimeter. This value would not increase by more than 20 percent if the core inlet and exit pressure losses were included. The total would still be a small pressure drop even for an electromagnetic pump.

In regard to the allowable Li temperature rise through the core, in a Rankine cycle system it is desirable to have this rise as small as possible in order to maximize the radiator temperature. However, making the temperature rise smaller for the same reactor power necessitates increasing the flow velocity with a consequent increase in pressure drop. Also, if the primary loop of this reactor were used as the heat source for a Brayton cycle system, a relatively large temperature rise might be used. As a compromise, an arbitrary temperature rise of 50° K was assumed.

A 7-kilogram-per-second Li flow rate through a 50° K temperature rise requires a power of about 1.4 megawatts. Since the heat-transfer area in this core is about 4700 square centimeters, the average heat flux is about 300 watts per square centimeter. The radial fuel-zoning study showed that the average heat flux through the wall of any particular coolant hole would not be greater than about 1.1 times the average (or about 330 W/cm^2). Furthermore, because of the cosine-shaped, axial power distribution, the maximum local heat flux would be about 1.25 times the flux through the wall of a coolant hole. Thus, if hot channel factors, which should be small in this type of core, are neglected, the maximum local heat flux at any point in the reactor will be about 413 watts per square centimeter. This flux value also neglects the change in power distribution with core life. In the 20 000-hour life study discussed in the section Core Power Flattening, this change in power distribution was almost negligible. Empirical equations in reference 10 for predicting the critical heat flux of liquid metals indicate that Li should be capable of sustaining heat fluxes in excess of 630 watts per square centimeter. Thus, in a long-term development, a maximum local heat flux of 413 watts per square centimeter should not be limiting. However, even if this flux value were reduced to 300 watts per square centimeter, this core would still be capable of transferring 1 megawatt of power to the Li.

In regard to the fuel temperature limit, reference 1 sets the maximum temperature drop through W-UO_2 fuel material at 278° K. This value is based on thermal cycling experiments on UO_2-W cermets. Figure 9 of reference 1 shows that a core with a 10.3-percent Li coolant fraction and 0.5-centimeter-diameter coolant holes can operate at a power density of about 250 watts per cubic centimeter without exceeding the 278° K temperature drop imposed on the W-UO_2-fueled core. Thus, a 15-centimeter-diameter by 30.5-centimeter-long core, with a volume of about 5400 cubic centimeters, would produce about 1.35 megawatts of power when operated at this same power density. Actually, at the temperatures of interest, UN has a thermal conductivity which is about eight times that of UO_2; so, if this core were operated at 1 megawatt power, the temperature drop through the fuel would be much less than 278° K.

In reference 1 it was found that, for nonvented fuel elements, fuel burnup is a major consideration in choosing a reactor size for a given power level. This finding was based on the stress generated in the fuel cladding by the fission products and not from a lack of reactivity or reactivity control. In these high-temperature reactors this stress probably could be almost eliminated if the fission product gas could be vented to a lower temperature plenum outside the core. Another arrangement that should be investigated is a fixed, long-life, 4π shield designed for replaceable reactors. In some applications, this arrangement might relax the burnup requirement while still maintaining a significant advantage over other sources of space power. In any event, the real potential of nuclear reactors as sources of space power for various applications cannot be obtained by allowing the reactor size to be set by a single, arbitrary core life and power level. Thus, for the purposes of this report, the fuel cladding stress limitation was relaxed. In regard to the required burnup reactivity, as previously described, a life study on this core which was assumed to operate at 1.25 megawatts showed that only 1.8 percent $\Delta k/k$ of burnup reactivity would be required for 20 000 hours of operation. In such a small core, this amount of reactivity control can be obtained in a number of ways.

SUMMARY OF RESULTS

Criticality calculations indicate that 30.5-centimeter-long, cylindrical cores which contain 45 volume percent molten uranium-233 (U^{233}) or plutonium-239 (Pu^{239}) and which use tungsten (W) as the structural material and lithium-7 (Li^7) as the coolant can be as small as 11 centimeters in diameter. Cores of the same length which are fueled with 45 volume percent uranium-233 nitride ($U^{233}N$) instead of molten U^{233} or Pu^{239} are about 2.5 centimeters larger in diameter.

These results were obtained for cores which were radially reflected by 10.2 centimeters of beryllium oxide (BeO). However, when BeO is used as the neutron reflector, there is a significantly larger peak-to-average power density at the reflector core interface than there is for a core with a heavy metal reflector for which the maximum power density is at the center of the core.

A tungsten-reflected reactor, 30.5 centimeters long, and composed of 45 volume percent $U^{233}N$, is about 16.5 centimeters in diameter. This diameter can be decreased to about 15 centimeters by increasing the $U^{233}N$ volume to 47.5 percent of the core. The smallest cores fueled with uranium-233 dioxide ($U^{233}O_2$) were 3.5 centimeters larger in diameter than the nitride cores. Cores fueled with uranium-235 (U^{235}) are considerably larger than those fueled with U^{233}. An alternate reflector of molybdenum (Mo) gives about a 2 percent greater neutron multiplication factor k_{eff} than does a W reflector of the same thickness, but a carbon (C) reflector gives about 1 percent less k_{eff} than does the W.

By using not more than four different fuel concentrations in the fuel elements and still holding the maximum fuel concentration to a reasonable value, the radial power distribution of a 15-centimeter-diameter, $U^{233}N$ reactor, which is reflected by 10 centimeters of W, can be made uniform within ±10 percent.

A one-dimensional, fuel-depletion study on this fuel-zoned core showed that in 20 000 hours of operation 1.8 percent $\Delta k/k$ of reactivity was lost and that the change in local-to-average radial power distribution was less than 1 percent. This depletion calculation was made in only the radial direction and it did not contain reactivity control features. But the small amount of reactivity control required for this fuel depletion indicates that the associated shift in power distribution in a controlled reactor would not be severe.

Finally, a consideration of the core pressure drop, the Li temperature rise, the heat flux, the fuel temperature, and the fuel burnup in this same 15-centimeter-diameter, fuel-zoned core indicates that it should be capable of transmitting 1 megawatt of thermal power to a Li coolant.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, November 3, 1967,
120-27-06-18-22.

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TABLE I. - CALCULATION OF CRITICAL EXPERIMENTS

[Spherical geometry.]

Core			Reflector			Neutron multiplication factor, k_{eff}	Reference
Material	Density, g/cm^3	Diameter, cm	Material	Density, g/cm^3	Thickness, cm		
Oralloy[a]	18.8	11.528	Be	1.84	10.16	1.032	3
Oralloy[a]	18.8	11.900	BeO	2.69	10.16	1.065	3
Oralloy[a]	18.8	12.822	W[b]	17.39	10.16	1.018	3
Oralloy[a]	18.8	15.22	W[b]	17.39	2.54	1.021	3
[c]U^{233}	18.62	9.200	Be	1.83	4.196	1.043	4
[c]U^{233}	18.62	10.088	Be	1.83	2.045	1.035	4
[c]U^{233}	18.62	9.200	W[d]	17.2	5.790	1.023	4
[e]Pu^{239}	15.62	10.088	Be	1.83	3.688	1.044	4
[e]Pu^{239}	15.62	10.088	W[d]	17.2	4.701	1.024	4

[a]Contains 93.5 vol. % U^{235} and 6.5 vol. % U^{238}.
[b]Contains 7 wt. % Ni and 3 wt. % Cu.
[c]Contains 1.1 wt. % U^{234} and 0.7 wt. % U^{238}.
[d]Contains 5.5 wt. % Ni, 2.5 wt. % Cu, and 0.7 wt. % Zr.
[e]Contains 4.9 at. % Pu^{240} and 0.31 at. % Pu^{241}.

TABLE II. - REACTOR FUEL PARAMETERS

Fuel	Density, g/cm^3	Atom density ratio of fuel to UO_2, N/N^{UO_2}	Fuel atom density, N, atoms/(b)(cm)
$U^{233}O_2$	10.7	1.00	0.0243
$U^{233}N$	13.5	1.35	.0329
U^{233}	18.8	2.00	.0488
Pu^{239}	19.7	2.04	.0498
[a]U^{233}	16.63	1.75	.0426
[a]Pu^{239}	16.54	1.70	.0412

[a]Melting-point temperature.

TABLE III. - ENERGY GROUP STRUCTURE[a]

13 Group, fast (b)		7 Group, fast (b)		5 Group, thermal	
Group	Lower lethargy	Group	Lower lethargy	Group	Lower lethargy
1	1.0	1	1.5	1	17.00
2	1.5	2	2.5	2	17.50
3	2.0	3	4.0	3	17.95
4	2.5	4	5.5	4	18.65
5	3.0	5	7.5	5	20.03
6	3.5	6	9.5		
7	4.0	7	17.0		
8	4.5				
9	5.5				
10	6.5				
11	7.5				
12	9.5				
13	17.0				

[a]Zero lethargy at 1.6×10^{-12} J.
[b]For tungsten-reflected cores, groups 13 and 7 were sometimes omitted because they contained very few neutrons; this resulted in a 12-group and a 6-group set.

TABLE IV. - COMPARISON OF REFLECTOR MATERIALS

[Fuel composition, 0.45 $U^{233}O_2$, 0.10 void, and 0.45 W; radial reflector thickness, 10.16 cm; axial reflector composition, 0.85 W and 0.15 Li^7; axial reflector thickness, 10.16 cm (8.5 cm axial reflector saving).]

Reflector composition	Core size, cm (a)	
	25.4	30.48
	Neutron multiplication factor, k_{eff}	
Carbon (0.92C-0.03W-0.05Li^7)	1.024	1.055
Tungsten (0.95W-0.05Li^7)	1.042	1.069
Molybdenum (0.92Mo-0.03W-0.05Li^7)	1.063	1.091

[a]Core height same as core diameter.

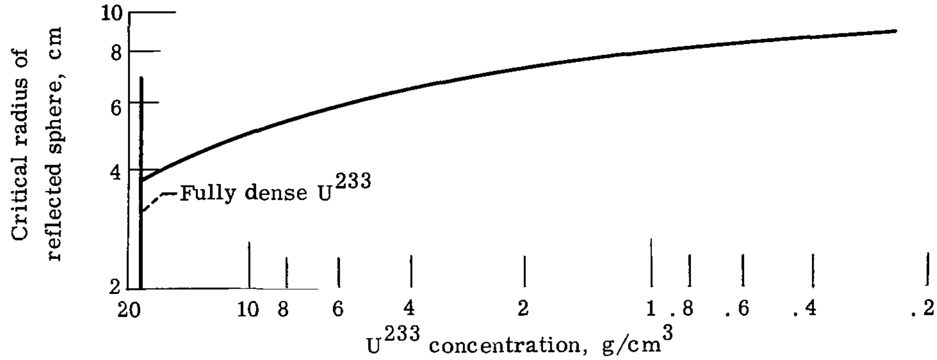

Figure 1. - Minimum critical radii of water-reflected spheres of U^{233}-H_2O solution (from ref. 2).

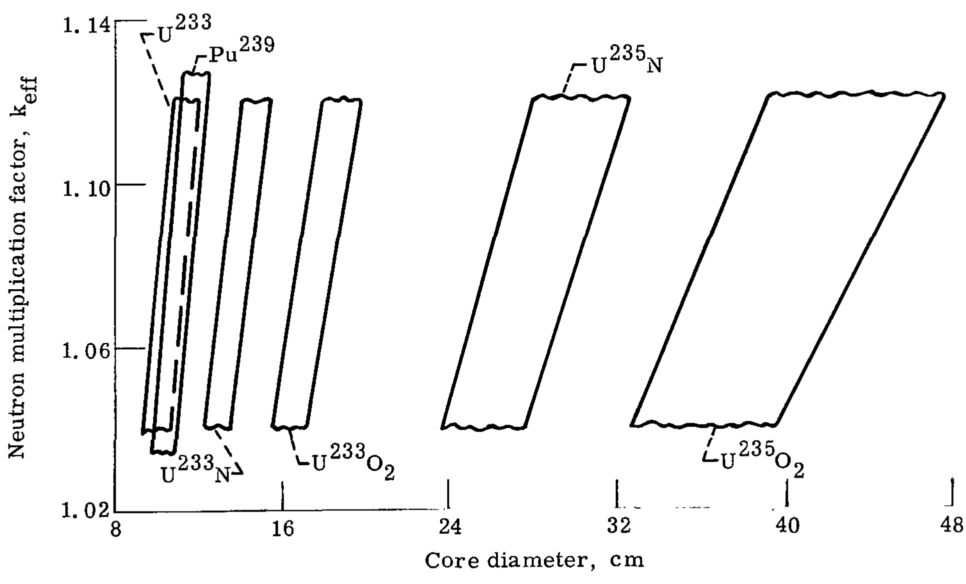

Figure 2. - Multiplication factors for 30.48-centimeter-long cylindrical reactors with 10.16-centimeter-thick radial reflectors. Enclosed areas correspond to fuel material range of 43.3 to 47.5 volume percent of core.

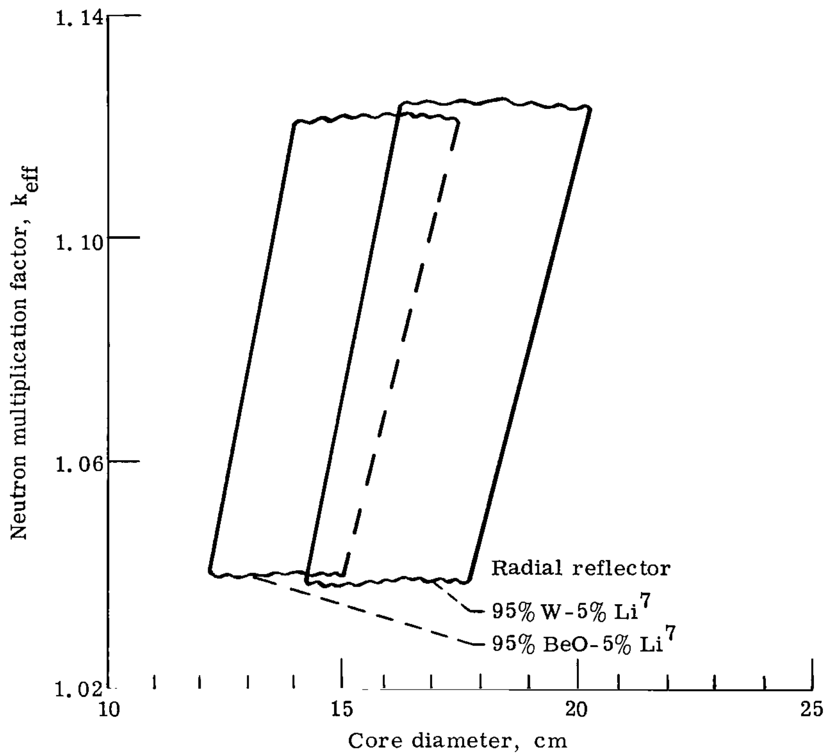

Figure 3. - Multiplication factors for 30.48-centimeter-long cylindrical reactors with different radial reflectors. Fuel, $U^{233}N$; reflector thickness, 10 centimeters. Enclosed areas correspond to fuel material range of 38.9 to 47.5 volume percent of core.

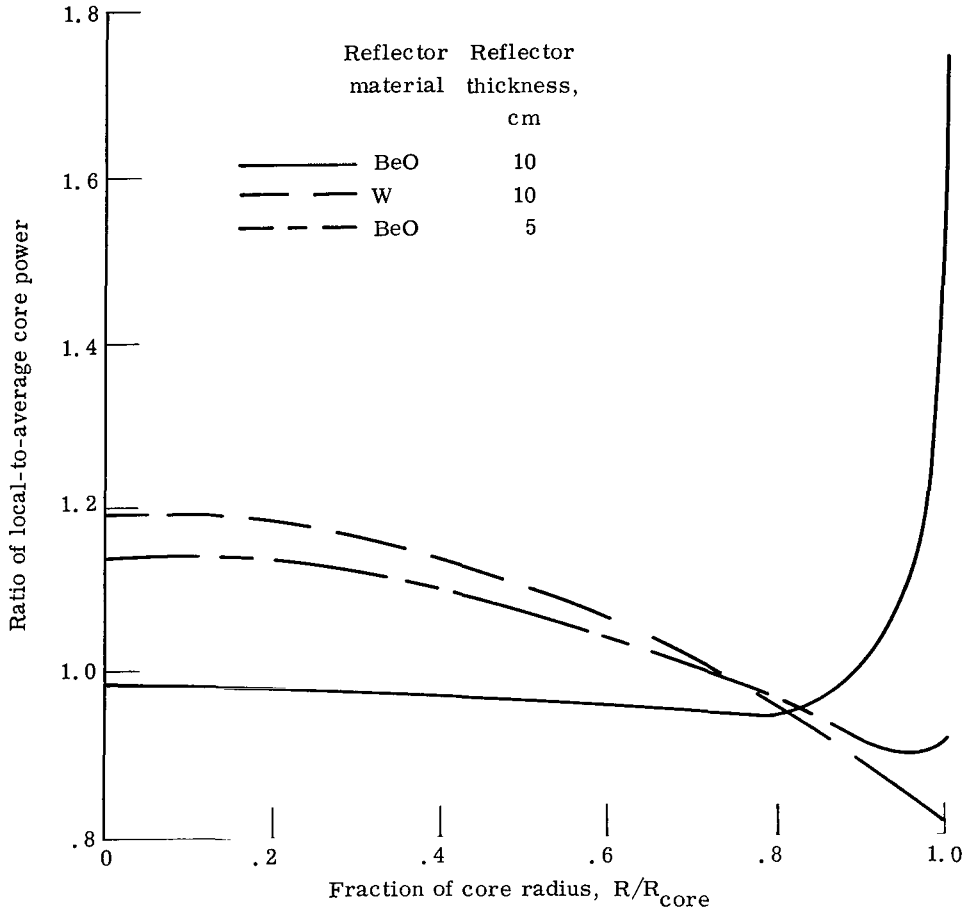

Figure 4. - Radial power distributions (local-to-average) in BeO- and W-reflected reactors. $U^{233}N$ - W cores with 10 percent Li^7 coolant.

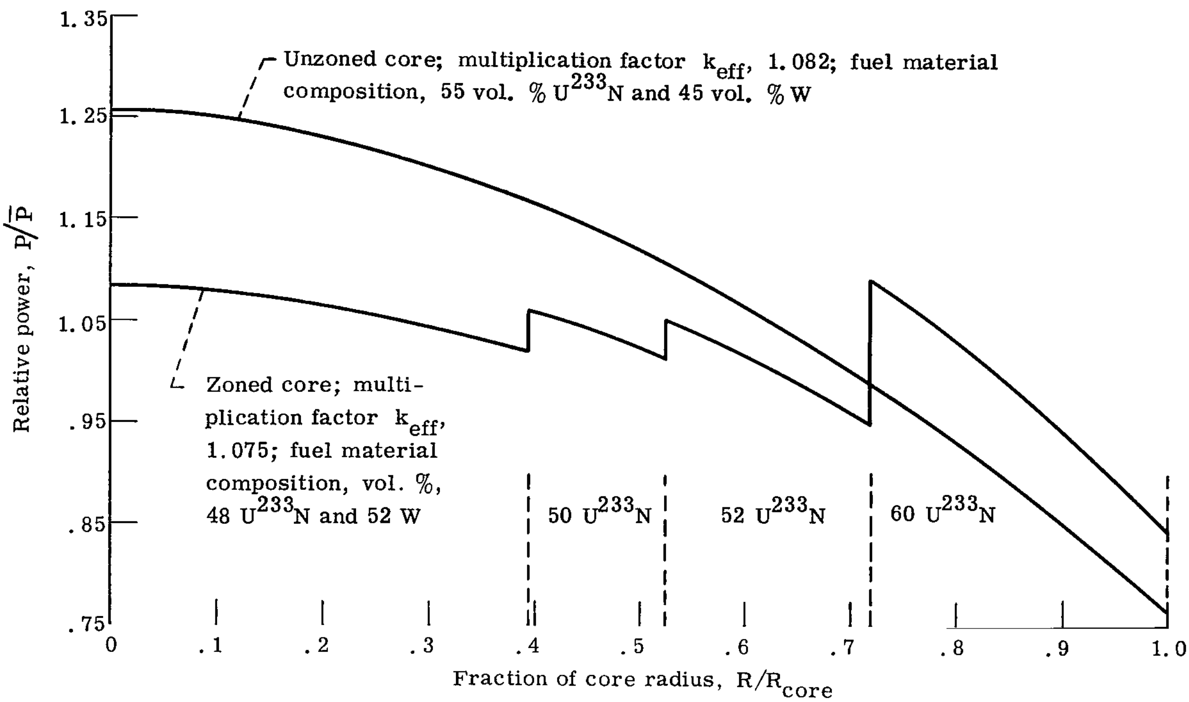

Figure 5. - Radial power distribution for fuel-zoned and unzoned, W-reflected cores. Core length, 30.5 centimeters; core diameter, 15.2 centimeters.