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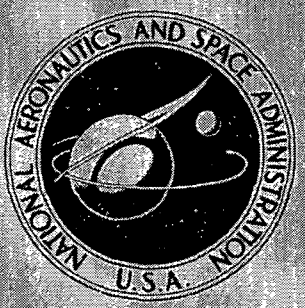

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CRITICALITY STUDY OF MELTDOWN CONFIGURATIONS FOR NUCLEAR AIRCRAFT REACTORS

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16. Abstract

As part of a containment system study for the nuclear aircraft reactors, a description of the meltdown problem is presented along with curves of reactivity for reflected and unreflected slabs, spheres, and cylinders as a function of UO_2 fuel loading, composition, and size. These are presented to help the early design concepts avoid configurations that would cause excursions in the event of a core meltdown and thereby obtain a feasible containment system. Results are presented which indicate that masses that have melted down from reactors can be kept subcritical provided certain geometrical dimensions for specific dilutions of fuel are not exceeded within the containment system.

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SUMMARY

This report presents a set of curves for slabs, cylinders, and spheres showing reactivity as a function of size, fuel loading, and composition for both reflected and unreflected geometries. These curves were generated to help determine design restraints when the designer considers the possibility of a reactor meltdown following a nuclear airplane accident. Since a reactor for a nuclear airplane would have fuel loadings exceeding several hundred kilograms, the prevention of excursions in the case of a meltdown is of concern for the survival of a containment system.

The critical dimensions are presented for each basic shape (slab, cylinder, and sphere) below which no critical excursion can take place. With no dilution of the uranium dioxide, this thickness for slabs is approximately 8 centimeters; for cylinders the maximum radius is 10 centimeters; and for spheres the maximum radius is 15 centimeters. Diluting the fuel, if it can be accomplished, can markedly increase the allowable dimensions and uranium dioxide content. In one case, for example, a slab of uranium dioxide diluted with four parts by volume of molybdenum increased the allowable thickness from 8 to 30.48 centimeters.

INTRODUCTION

Safety of a nuclear airplane is a major concern of the nuclear powerplant designer. To make the nuclear airplane safe, it is necessary to prevent the release of fission products to the atmosphere. One way to prevent this release is to put the reactor inside of a containment vessel and then prevent the vessel from rupturing. Figure 1 is a schematic drawing of a water-moderated gas-cooled reactor inside a containment vessel that has been considered for aircraft use.

There are two primary causes of containment vessel rupture: the crash (impact)

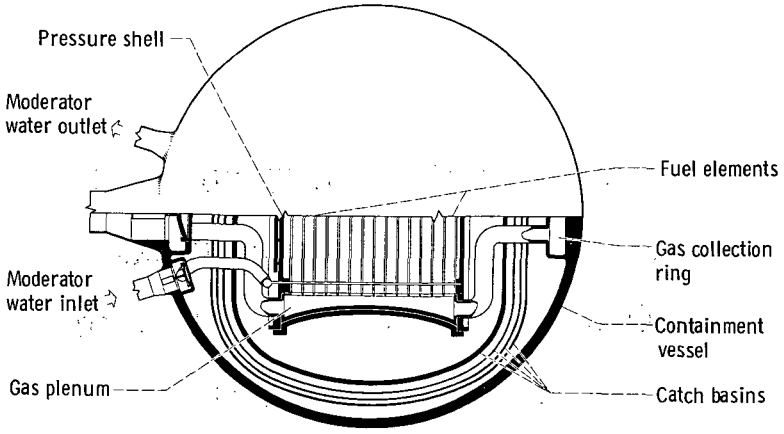


Figure 1. - Thermal, gas-cooled, water-moderated reactor.

itself and heat release after the crash. The concern of this report is the heat release. The heat is released by the decay of fission products after impact when there is no longer any coolant flow through the core. The problems associated with preventing containment vessel rupture due to impact are discussed in references 1 and 2. The fission product decay heat can have several effects. Among them, hot spots could be created on the containment vessel wall, and the pressure inside the vessel would be increased. This combination of effects could cause vessel rupture. These problems are discussed in reference 1.

In addition to these problems the fission product decay heat could cause the core to melt. The molten core material could then flow out of the core and collect in a supercritical configuration. The energy release from the resulting nuclear excursion could increase the internal pressure and hot-spot temperatures of the containment vessel thus increasing the chance of vessel rupture.

This report analyzes a part of the meltdown criticality problem. Only the situation where molten core materials have flowed out of the core region is considered. The probability of criticality on meltdown increases rapidly as core loadings increase. Since the core loading for a nuclear aircraft reactor will exceed several hundred kilograms of ^{235}U, the possibility of criticality on meltdown must be carefully considered in the design to insure its avoidance.

Meltdown criticality depends primarily on the size, shape, and composition of a collection of core material and on the presence or absence of neutron reflecting material around the collection. The purpose of this report is to indicate some of the combinations of the parameters that could lead to critical assemblies in the containment vessel of a nuclear aircraft reactor after meltdown. The intent here is to provide information to help the reactor designer recognize some of the conditions inside the containment vessel that might lead to a critical or supercritical configuration. Knowing these conditions,

the reactor designer can provide designs that avoid the possibility of critical assemblies that might occur during a meltdown.

To determine in detail the meltdown criticality problem of a particular reactor design requires a comprehensive study of the entire system design. However, useful information to help guide the design can be obtained early in the design process by a relatively gross review of the system. The usefulness of this report lies in its applicability to the one problem common to all meltdown criticality analyses - the determination of the reactivity of a given collection of core material. The curves presented herein should help the reactor designer to estimate the reactivity of his configuration and indicate design changes to be made in size, shape, and composition to avoid the possibility of criticality.

Since this report deals with the reactivity of core materials after they have flowed out of the core, the results can be applied to a variety of reactor types and designs.

DESCRIPTION OF MELTDOWN CRITICALITY PROBLEM

This section has two purposes: To acquaint the reader with some of the events that will probably occur in the containment vessel after a severe aircraft crash which stops normal cooling of the reactor and to describe the factors that could cause a fast spectrum nuclear excursion and discuss those that are most important. Only configurations of molten core materials that have collected outside of the core region are considered.

Sequence of Events from Impact to Excursion

It is assumed that the reactor is shut down before impact. In a fast reactor or a thermal reactor with solid moderator, the reactor will be at least 2 percent subcritical. In a water-moderated thermal reactor, the moderator could be drained out of the core, and the reactor made more subcritical than a fast reactor. It should be noted that after impact the reactor containment vessel may have any orientation with respect to the ground. Some possible events caused by release of fission product decay heat after impact and possible sequences are briefly described to illustrate the nature of the problem.

(1) The reactor cooling system which normally carries away the heat released by decaying fission products operates up to impact. When impact is detected, the coolant line valves located at the containment vessel are closed. This seals the containment vessel but also stops the decay heat coolant flow. The heat release rate due to fission product decay is shown in figure 2 as a function of time after reactor shutdown.

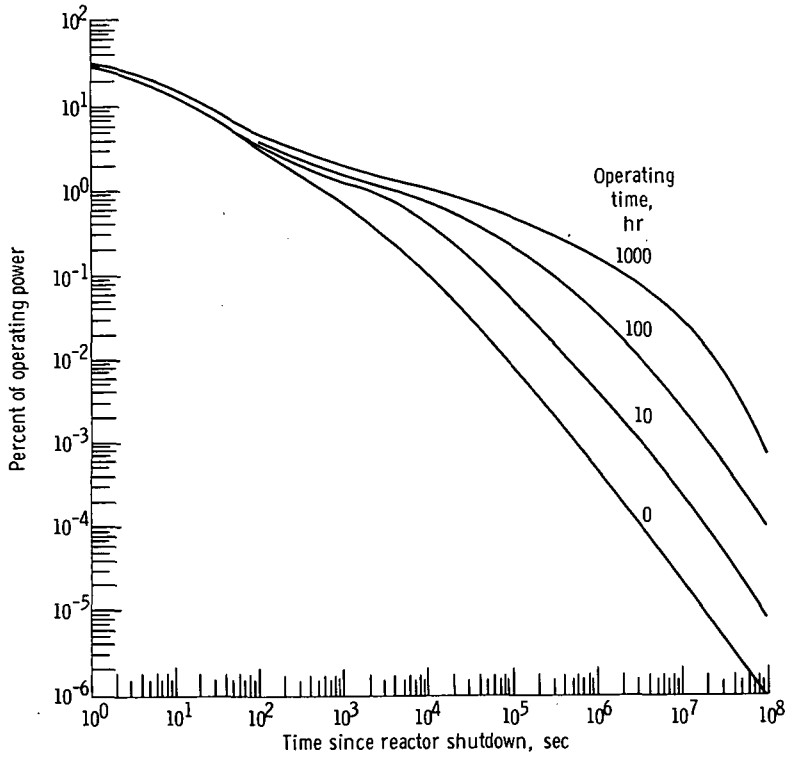


Figure 2. - Afterheat power as function of time. Negative reactivity insertion, 2 percent.

(2) After impact the only way to get rid of the decay heat is to transfer it to the containment vessel. This heat is transferred from the containment vessel by radiation and natural convection of air.

(3) If the fuel elements have not started to melt, the temperature of the fuel material is primarily affected by (a) the resistance to heat transfer between the fission product heat sources and the containment vessel and (b) the heat capacity of the core and other materials (structure and shield materials) which are heat-transfer coupled to the core. The heat capacity is important because the fission product decay heat decreases with time (see fig. 2). If the heat capacity is large, then the temperature rises slowly and long times are required to raise the core to melting temperature. During this time the fission product heat generation rate is decreasing but the heat-transfer rate to the containment vessel and from the vessel to the air is still increasing. The core temperature peak occurs when the rate of heat transfer away from the core equals the heat generation rate. For reference, figure 3 illustrates core temperature as a function of time, showing the effect of structural heat capacity and resistance to heat transfer.

(4) If the fuel starts to melt, there will be gross release and transport of the fission product heat sources from the hot fuel to colder places inside the containment vessel. This could reduce the fuel temperature, and in some cases cause the fuel to resolidify

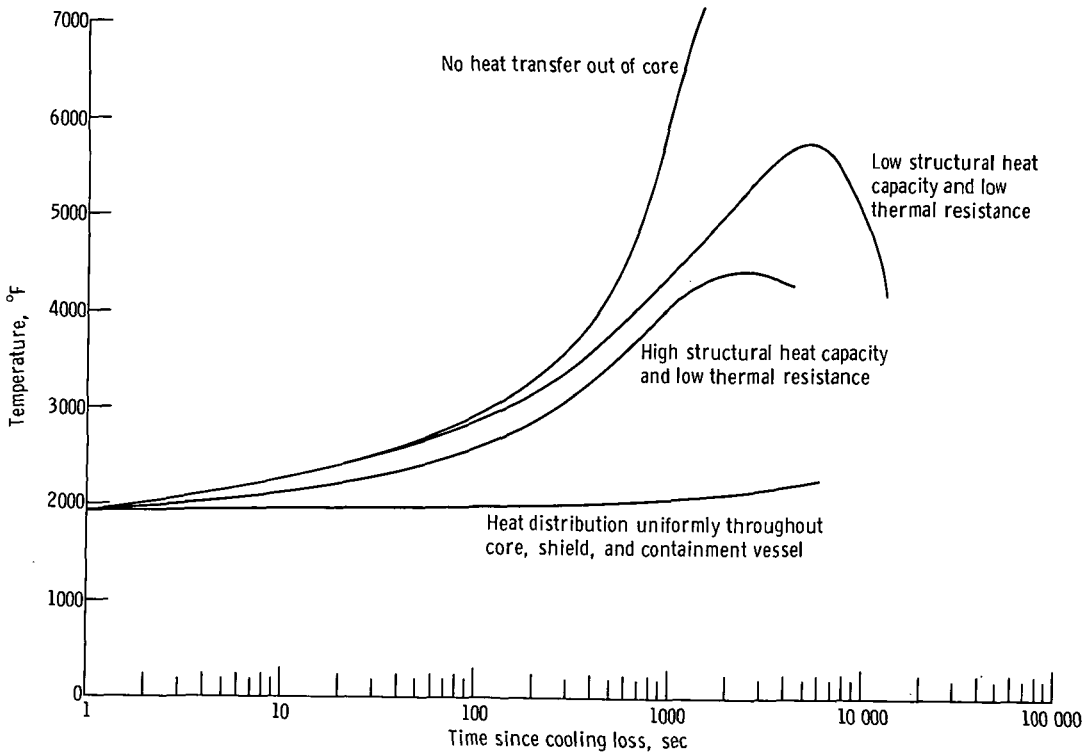


Figure 3. - Reactor core temperature from afterheat as function of time since cooling loss.

before a significant amount has melted or flowed. The heat sources transport because, at the melting temperature of uranium dioxide (UO_2) (3000 K), many of the fission products are gases and the others are either above their boiling points or have high vapor pressures.

(5) In many reactor designs the core structure material will have a lower melting point than the fuel elements. In these cases some of the structure may melt before the fuel elements. This material will drop or flow out of the core and solidify at places of lower temperature. If this low melting-point material flowed across the metal fuel elements, it could combine with the clad material and form a lower melting point alloy. The melted flowing structure material could then erode or wash away the clad material exposing the fuel material before the clad or fuel reached melting temperature. In this case, fuel still containing fission product heat sources could drop out of the core and collect someplace below the core. This collection of material would increase in temperature due to fission product decay heat, and then melt, compact, homogenize, increase in reactivity, and possibly cause an excursion if sufficient material is permitted to collect by the particular design.

(6) Material may melt and flow in various sequences of melting, flowing, solidifying, melting, etc. Melted material could flow or drip, then solidify as it transfers heat to

the colder materials it contacts. If material continues to flow to this area, collections of materials may form. The temperature at a collection site will increase because of the addition of hot molten material, poor heat transfer from the collection site, and transport of heat sources to the collection site. The temperature will stop increasing when the collection begins to melt. Material will then begin to flow to new collection sites. Several collections, dams, or pools of melted material could be formed. Material will stop moving and gradually cool because of the combination of a decrease in total heat released, the transport of heat sources from hot places to cold places by vapor transport, and an increase in amount of heat transfer surface.

Factors Affecting Criticality of Materials after Meltdown

The probability of a nuclear excursion is a function of the configuration (size and shape) of the collection site and the composition of the melted material that collects at the site. This section will discuss the probable configurations and compositions of collections of melted materials.

Configuration. - When molten materials drip and flow through passages in a structure, an infinite variety of configurations can form. Materials can pile up in mounds or stalagmite configurations. Materials can collect in pools or basin-like shapes or in long strips like icicles or miniature lava flows. Piles or mounds could be formed on previously formed basin structures resulting in spheroids. One cannot predict the exact configuration that will be formed. It is more difficult to predict the configuration after a nuclear airplane accident than after a land powerplant accident because the containment vessel can roll free of the airframe and have any orientation relative to the ground.

It would be very tedious and beyond the scope of this report to calculate the reactivity for all possible configurations. Configurations for which reactivities can be readily calculated are spheres, cylinders, and slabs. The problem of defining a meltdown configuration becomes one of finding probable collection sites and determining which standard shape (sphere, cylinder, or slab) best approximates the site. For a given amount of melted material the sphere has the highest criticality factor, and the slab has the lowest. To be conservative one looks first for possible spherical collection sites, then for cylindrical sites and last for slab sites. This report presents analytical results to help determine whether or not a reactor of specified mass will, upon melting and possible dilution, become critical in a newly defined shape. If the size, shape, and composition of the melted material can be estimated, the reactivity can be estimated from the figures that are presented in the RESULTS AND DISCUSSION section.

Composition. - Materials can collect in layers or they can collect uniformly mixed. The amount of each material can vary greatly. How they collect is a function of many

factors. Some of the important factors are the melting temperature of the core and structure materials, solubility, volume, and mass fractions of the core materials, after-heat generation rates, release of heat sources, and heat capacity and ease of heat transfer in core along the melt flow path and at the collection site. From a fast critical excursion point of view, the important questions are (1) what is the dilution of fuel by nonfuel materials and (2) do nonfuel materials surround the fuel and reflect neutrons?

Nonfuel materials mixed with fuel materials affect the reactivity significantly because they affect the fuel concentration. Fuel concentration, atoms of ^{235}U per cubic centimeter, in a fast spectrum is the most important quantity affecting the reactivity. The lower the fuel concentration, the less reactive the configuration is. In general adding nonfuel and nonmoderating materials to a collection of melted materials tends to reduce the reactivity of the collection by diluting the fuel concentration. The nonfuel materials also reduce the reactivity slightly because of the absorption of neutrons. This effect is small because their fast neutron absorption cross sections are small compared with those of the fuel.

The effect on reactivity of adding nonfuel materials and diluting the fuel concentration will be shown in the results. These curves were obtained for cores composed of molybdenum-clad, UO_2 fuel, and stainless-steel structure. However, since the type of material causing the dilution does not significantly affect reactivity (ref. 3), these curves can be used for other clad and structure materials.

As explained, nonfuel materials decrease reactivity when mixed with fuel. However, when nonfuel materials surround a collection of melted materials containing fuel, they can reflect neutrons back into the fueled mass increasing the reactivity. Each possible collection of melted materials must be examined to determine whether it is a reflected configuration. There are three ways to get a reflected configuration:

(1) Reflector material is deposited on top of solid fuel mixture.

(2) In the molten state the fuel and diluent separate to form distinct fuel and reflector regions.

(3) The fuel mixture is deposited on solid materials.

ANALYSIS

Assumptions and Limitations

The purpose of this section is to indicate the assumptions made in the analysis and some of the limitations of the results.

Neutron spectrum. - This report considers only fast spectrum neutron systems, that is, fast reactor or water-moderated thermal reactors that have the water removed.

It is intended to apply only to configurations of core material that have melted and have left the core region. It does not consider the criticality of a core deformed by impact or partially melted fuel in the reactor region.

Fuel-diluent mixing. - One of the major assumptions used is that, when the fuel and diluent are mixed, the mixture is homogeneous. There will, of course, be cases for which this is not true. Each of the parametric studies includes calculations with no dilution.

Diluent material. - Molybdenum was assumed to be the diluent since it is a contender for aircraft reactor fuel clad. It is shown that the results are of use even if other diluent materials are present.

Infinite slabs and cylinders. - In the slab and cylinder calculations infinite transverse dimensions were used to provide conservative results. For each shape, however, some calculations were performed with finite transverse dimensions to indicate the degree of conservatism.

Reflector thickness. - Except for two calculations in connection with the slab configuration, only one reflector thickness was used throughout. Therefore, there is no direct determination of the effect of reflector thickness.

Configurations and Compositions

Reactivities were calculated for slab, sphere, and cylinder configurations. The parameters of interest in these systems are the reactor dimensions, the fuel enrichment, and the amount of diluent (clad structure, etc.) present.

The volume ratios of diluent to fuel are based on the total fuel volume present. The only diluent other than uranium-238 was the clad material, molybdenum. Since, as previously pointed out, molybdenum is somewhat typical (for purposes of this report) in nuclear properties of the materials used in a core, it is felt that the results obtained using molybdenum can be extended to cases of other diluent materials.

The dilution effect of $^{235}UO_2$ was taken care of by varying the fuel enrichment. The range of fuel enrichments was chosen to bracket the most commonly used values considered for aircraft reactors. In computing the enrichment, the mass ratio of $^{235}UO_2$ to the total UO_2 mass was used. The ratios of diluent volume to fuel volume used in the slab calculations were 0, 1, 2, 4, and 5. The enrichments for ^{235}U were 100, 90, 75, and 60 percent.

Figure 4(a) illustrates a bare sphere of UO_2 and diluent. For this case the fuel enrichment was held fixed at 100 percent $^{235}UO_2$. This fuel enrichment was chosen to produce conservative results. Diluent to fuel volume ratios of 0 and 4 were used. The fuel loadings employed were 200, 400, 600, 800, and 1000 kilograms of $^{235}UO_2$. For all

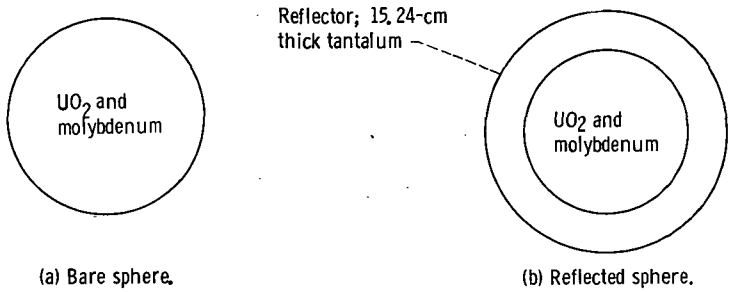


Figure 4. - Schematic of sphere with and without reflector.

the sphere calculations, the sphere size was determined solely by the mass of material present.

Figure 4(b) shows schematically a sphere surrounded by a reflector. In these calculations the mass of $^{235}UO_2$ was held fixed at 500 kilograms. Two values of the dilution ratio (0 and 4) were used, and enrichments of 60, 75, 90, and 100 percent were employed. A reflector consisting of 15.24 centimeters of tantalum completely surrounded the spheres.

Cylinder. - Molten material will not form cylinders unless it is confined by walls, for example, coolant ducts. Thus, it is not reasonable to choose the sizes of cylinders for our calculations on the basis of fuel mass. Cylinder diameters from 15 centimeters to 90 centimeters were calculated assuming infinite lengths. The infinite length was chosen to provide results as conservative as possible. In order to determine the degree of conservatism, four finite length calculations were made for length to diameter ratios of 1 and 2. All cylinders contained 100 percent enriched $^{235}UO_2$. Diluent to fuel volume ratios of 0 and 4 were used, and all cylinders were unreflected.

CALCULATIONS

The calculations were performed on an IBM 7094 II computer using two diffusion calculation programs available at Lewis. These will be described in the appendix. A check with a transport code indicated no appreciable difference in reactivity, and the time for computation was reduced for the RP-1 diffusion calculations.

The slab model requiring a large number of reactivity calculations was done in two steps. The first consisted of a set of calculations as a function of enrichment and dilution ratios using one fixed fuel mass for each combination of enrichment and dilution ratios. These calculations were done with the RP-4 computer program which used 72 neutron energy groups. The second set of calculations was performed with the RP-1 computer program using eight neutron energy groups obtained by group reduction from

the 72 RP-4 groups. This set of calculations determined the reactivity as a function of slab thickness for each of the enrichment and dilution ratios from the first set of calculations. Two separate computer programs were used to save time because the RP-1 program runs much faster than RP-4. Because of the relatively few numbers of calculations required for the sphere and the cylinder models, only the RP-4 program was used.

RESULTS AND DISCUSSION

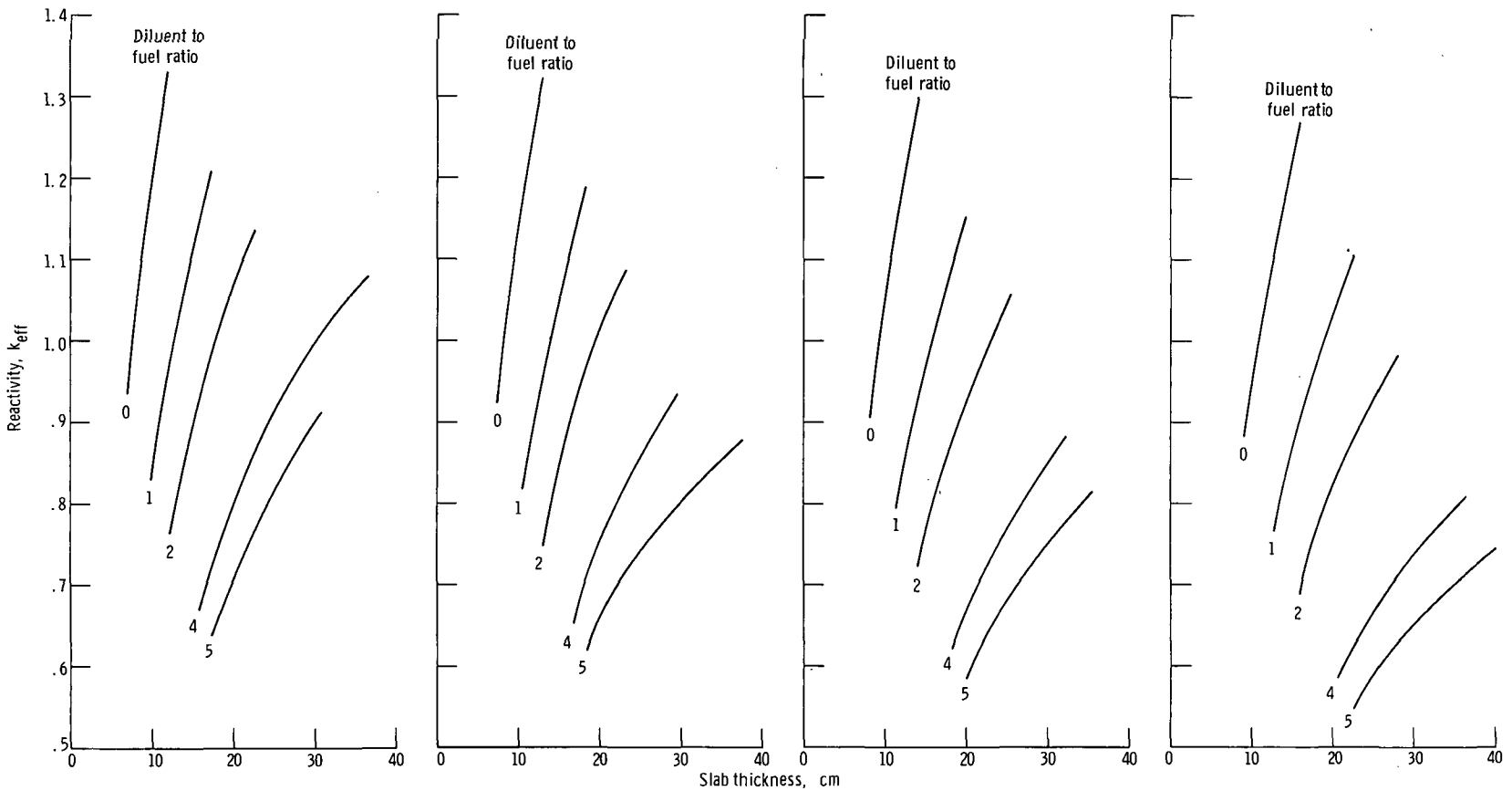
Slab

Table I contains the results of the slab calculations with a 15.24 centimeter reflector on one side. These results are plotted in figure 5 as a function of slab thickness. Since the calculation are based on infinite transverse dimension, the thickness is the controlling parameter for the reactivity of a slab of given composition. This point is very important

TABLE I. - REACTIVITY OF SLABS

[Infinite transverse dimensions; reflected on one side by 15.24 cm of tantalum.]

Fuel enrichment, percent	Diluent to fuel volume ratio									
	0		1		2		4		5	
	Thickness, cm	Reactivity	Thickness, cm	Reactivity	Thickness, cm	Reactivity	Thickness, cm	Reactivity	Thickness, cm	Reactivity
100	7.00	0.935	9.92	0.830	12.19	0.763	15.82	0.671	17.37	0.637
	8.58	1.08	12.19	.97	14.99	.893	19.48	.784	21.40	.741
	9.92	1.19	14.12	1.07	17.37	.986	22.60	.862	24.83	.814
	11.11	1.27	15.82	1.15	19.48	1.06	25.37	.921	27.89	.867
	12.19	1.34	17.37	1.21	21.14	1.11	27.89	.967	30.68	.908
	-----	-----	-----	------	-----	-----	33.42	1.03	-----	-----
	-----	-----	-----	------	-----	-----	36.47	1.081	-----	-----
90	7.37	0.924	10.47	0.817	12.86	0.784	16.70	0.654	18.34	0.619
	9.05	1.07	12.86	.954	15.82	.875	20.57	.762	22.60	.718
	10.47	1.17	14.90	1.05	18.34	.965	23.86	.837	26.23	.787
	11.73	1.26	16.70	1.13	20.57	1.03	26.79	.892	29.47	.837
	12.86	1.32	18.34	1.19	22.60	1.09	29.47	.936	32.42	.875
75	8.08	0.905	11.48	0.794	14.12	0.722	18.34	0.624	20.14	0.587
	9.92	1.05	14.12	.927	17.37	.843	22.60	.723	24.83	.677
	11.48	1.15	16.35	1.02	20.14	.928	26.23	.791	28.86	.738
	12.86	1.23	18.34	1.10	22.60	.992	29.47	.841	32.42	.783
	14.12	1.30	20.14	1.15	24.83	1.04	32.42	.880	35.69	.817
60	9.05	0.883	12.86	0.765	15.82	0.689	20.57	0.585	22.60	0.546
	11.11	1.02	15.82	.8924	19.48	.801	25.37	.673	27.89	.625
	12.86	1.13	18.37	.983	22.60	.879	29.47	.732	32.42	.677
	14.42	1.21	20.57	1.05	25.37	.973	33.13	.775	36.47	.714
	15.82	1.27	22.60	1.11	27.89	.983	36.47	.808	40.18	.742



(a) 100 Percent enriched fuel (b) 90 Percent enriched fuel. (c) 75 Percent enriched fuel. (d) 60 Percent enriched fuel.

Figure 5. - Reactivity as function of slab thickness for various uranium dioxide enrichments. Infinite transverse dimensions; reflected on one side by 15.24 centimeters of tantalum.

because it allows the analyst the use of these curves for any size or shape that can be approximated by a slab.

The only limitation imposed, the infinite transverse dimension, is not a major factor. The infinite slab is only slightly more reactive than a finite one. A few finite slab calculations indicated that a finite slab is less reactive than an infinite one by about 0.05. Also, an error introduced will be in the direction of safety as the infinite slab will be more reactive than the actual finite slab. The results show that any time the fuel in UO_2 form melts into a slab configuration having a thickness of approximately 8 centimeters or less, it will not be critical. In addition, as the fuel is diluted either by another material such as molybdenum or uranium-238, this allowable thickness can be increased. For example, a 90 percent enriched fuel with a dilution of four parts molybdenum will not become critical until it is more than 30.48 centimeters (1 ft) thick.

Reflector effects. - Figure 6 shows the reactivity of a 30.68-centimeter thick slab

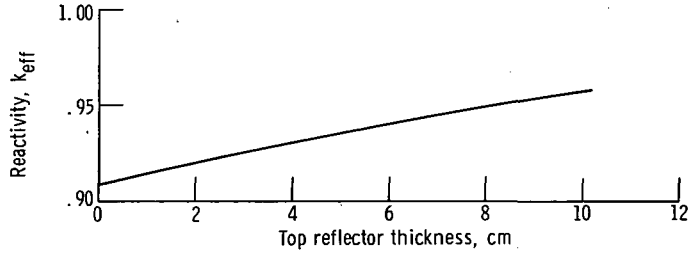


Figure 6. - Slab reactivity as function of top reflector thickness. 100 percent enriched fuel; diluent to fuel volume ratio, 5. Bottom reflector thickness, 15.24 centimeters of tantalum; fuel to diluent thickness, 30.68 centimeters; infinite width.

of 100 percent enriched fuel with a diluent to fuel ratio of 5 plotted as a function of top reflector thickness. This curve can be used to estimate the reactivity of a slab between two reflectors. Thus, a 5.08-centimeter thick reflector increases the reactivity by only 0.026. Although this is only one example, it does indicate that for thick slabs of fuel and diluent mixtures an additional reflector does not appreciably increase reactivity.

Sphere

Figures 7 and 8 show the results of the sphere calculations listed in table II. These results can be used whenever a sphere is the best approximation to a molten core material configuration. Calculations were performed on both bare spheres and reflected (Ta, 15.24 cm thick) spheres.

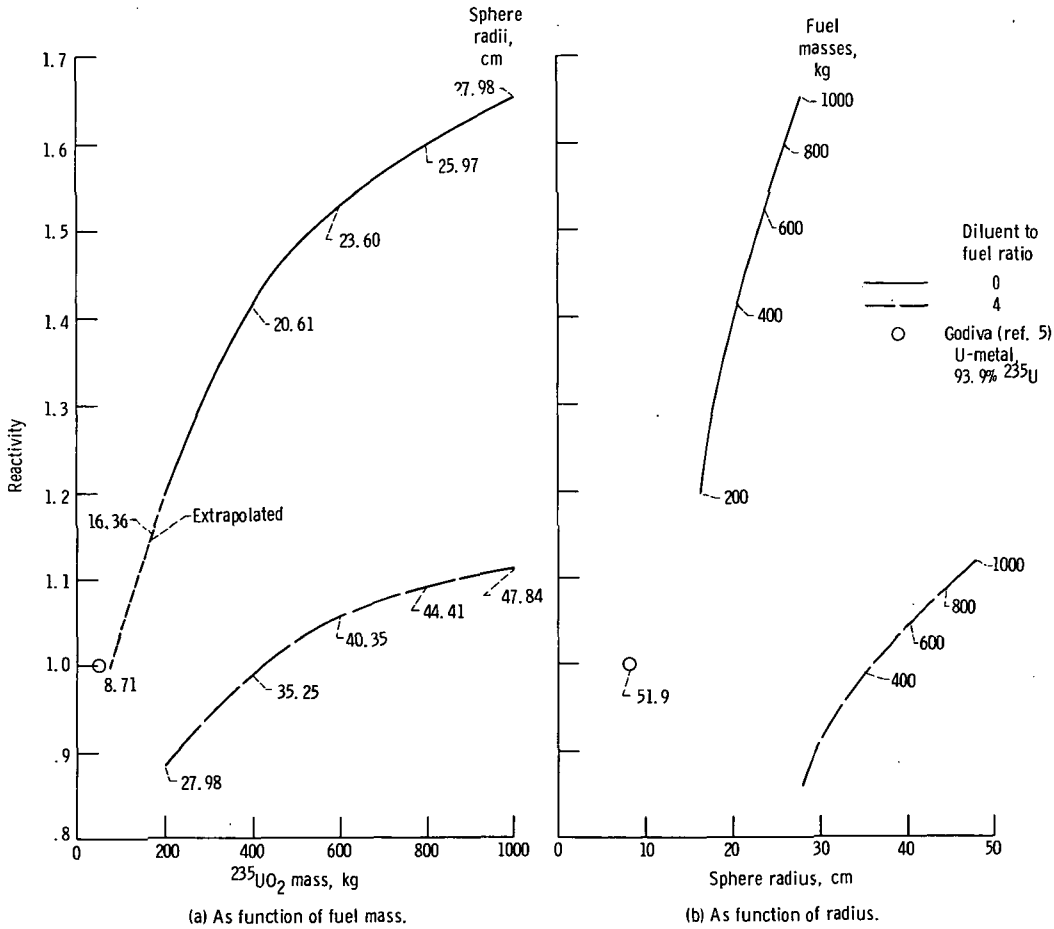


Figure 7. - Sphere reactivity as function of fuel mass and sphere radius. Unreflected sphere; $^{235}UO_2$ enrichment, 100 percent.

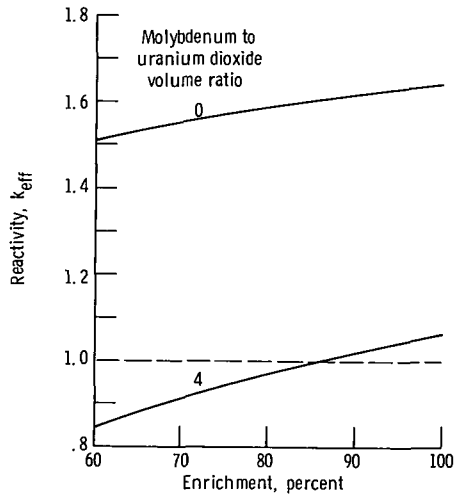


Figure 8. - Reactivity as function of enrichment for uranium dioxide containing 500 kilograms $^{235}UO_2$. Sphere reflected by 15.24 centimeters of Ta.

TABLE II. - REACTIVITY OF SPHERES

Unreflected sphere; 100 percent enriched fuel

Diluent to fuel volume ratio	Mass $^{235}UO_2$, kg	Radius, cm	Reactivity, k_{eff}
0	200	16.36	1.20
	400	20.61	1.41
	600	23.60	1.53
	800	25.97	1.60
	1000	27.98	1.65
4	200	27.98	0.859
	400	35.25	.988
	600	40.35	1.05
	800	44.41	1.09
	1000	47.84	1.12

Reflected by 15.24 centimeter of tantalum; 500 kilograms $^{235}UO_2$

Diluent to fuel volume ratio	Enrichment, percent	Radius (without reflector), cm	Reactivity, k_{eff}
0	100	22.21	1.64
	90	23.00	1.62
	75	24.44	1.57
	60	26.33	1.51
4	100	37.80	1.07
	90	39.33	1.02
	75	41.79	.942
	60	45.50	.848

Reactivity as function of mass and size. - Figure 7 shows the reactivities of an unreflected sphere of 100 percent enriched fuel plotted as functions of the fuel mass and sphere radius, respectively. Diluent to fuel volume ratios of 0 and 4 are used.

This plotting of the reactivity as functions of the two related parameters separately show the reactivity dependence on sphere dimension and fuel mass. Note that the reactivity increases much more linearly with radius than with fuel mass. Also, it is evident that the undiluted sphere is supercritical for even the smallest values of the parameters. For a fuel loading of 200 kilograms, which is a sphere of 16.35 centimeters radius, the reactivity is 1.2. Extrapolating the curve to a reactivity of less than 1.0 results in a sphere 15 centimeters or less in radius.

The reactivity of the diluted system again increases much more slowly than the undiluted system. This is in addition to the overall reduction in reactivity due to dilution. This reduction results in the critical mass being more than 400 kilograms for a dilution of 4 which is a 35.25-centimeter radius sphere.

Also plotted on figure 7 are the results of the Godiva experiment (ref. 4). This was a critical experiment using a sphere of 93.9 percent enriched uranium metal (i.e., no dilution by oxygen). This point serves as a reference for our calculations. We see that an extrapolation of the calculated curve for 100-percent enriched UO_2 gives slightly higher fuel mass for a critical configuration as compared with Godiva. Taking into account the diluting effect of the oxygen and the difference in enrichment, the calculations compare reasonably well with the experimental results.

Reflectors and enrichment. - Figure 8 shows the reactivity of a reflected sphere of

UO_2 containing 500 kilograms of $^{235}UO_2$ plotted as a function of the enrichment. The reflector (15.24-cm-thick tantalum) is the same as that used for the slabs. Again, the diluent volume ratios are 0 and 4.

There are several interesting features in this figure. First, of course, is the substantial reactivity increase due to the reflector. The unreflected sphere for 500-kilogram, 100-percent enrichment, and no dilution (from fig. 7(a)) has a reactivity of 1.48, while the reflected sphere has a reactivity of 1.64. For the dilute cases, the values are 1.03 and 1.07. This also shows that dilution, because it increases the sphere volume, reduces the effect of the reflector.

Cylinder

Figure 9 is a plot of reactivity as a function of cylinder radius for an infinitely long

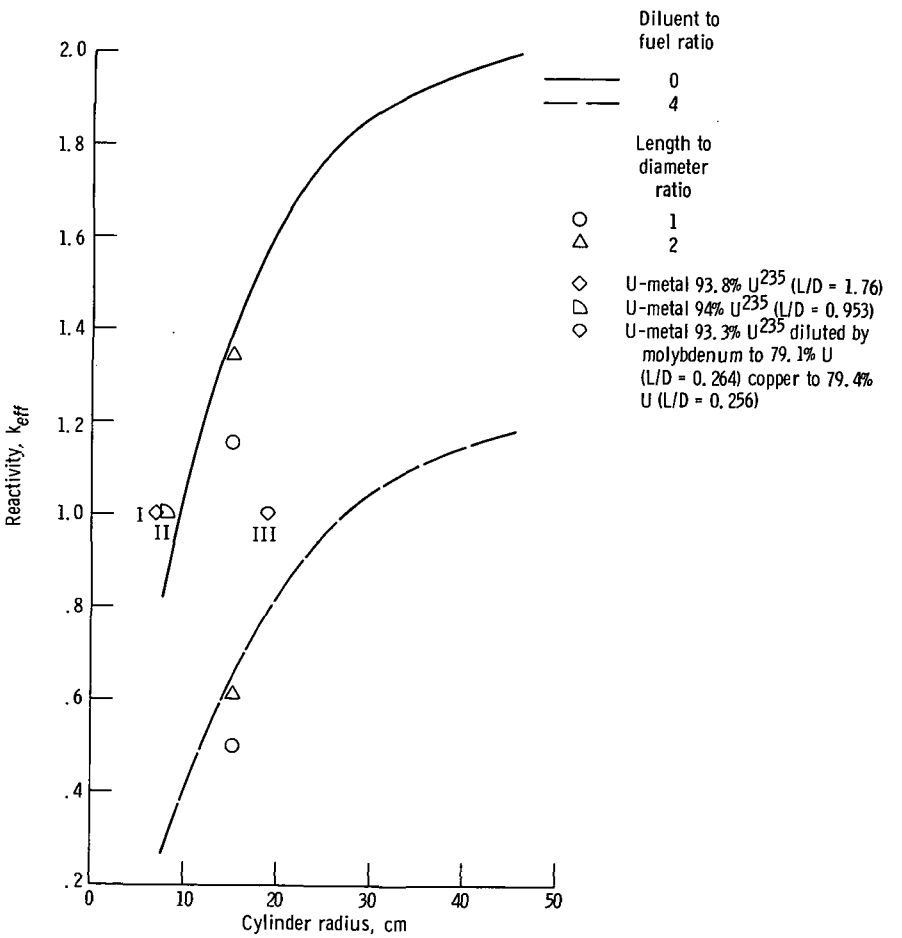


Figure 9. - Reactivity as function of cylinder radius. Infinitely long cylinder, $^{235}UO_2$ fuel enrichment, 100 percent.

cylinder. The fuel is 100 percent $^{235}UO_2$, and the dilution volume ratios are 0 and 4. As in the slab results of figure 5, it was possible to reduce the size dependence to one dimension.

Indicated on the figure are results for four finite length calculations. These finite results show that there is only about 0.08 difference in reactivity between the infinite length model calculations and the finite calculations that have a length to diameter ratio of 2. This means that we can make general use of these cylinder results. If the length to diameter ratio of a cylinder being studied is much less than 1, a change to a slab approximation is indicated. For a length to diameter ratio near 1, a sphere approximation can be used. This can be seen by comparing results with those on figure 7.

Some experimental points are also plotted on figure 9. These were obtained from reference 3. The points labeled I and II are undiluted uranium metal cylinders having 93.0 and 94.0 percent enrichment, respectively. The same kind of argument (i.e., dilution with oxygen) is used when made with respect to the agreement between the sphere calculations and the Godiva experiment. Thus, our cylinder results also appear to be reasonable.

The point labeled III is really two points. It represents critical experiments using diluted 93.3-percent enriched uranium metal cylinders of 19 centimeter radius. The two experiments were with copper dilution to 79.4 volume percent uranium and with molybdenum dilution to 79.1 volume percent. The difference in the experimental critical mass of uranium is less than 3.3 percent. Since molybdenum and copper are fairly dissimilar in their neutronic properties, we have some experimental evidence to support our assumption that our calculated reactivities are not very sensitive to the type of diluent material.

The undiluted cylinder becomes critical at a radius of about 9.5 centimeters. The diluted cylinder, however, remains subcritical until a radius of about 27.5 centimeters. Also, the reactivity increases more slowly with radius for the diluted fuel.

CONCLUDING REMARKS

The intent of this report is to provide the reactor designer with information that will help him in the preliminary analysis of meltdown and thus prevent criticality problems. Specifically, the curves presented provide assistance in the one area common to all meltdown criticality problems - the estimation of the reactivity of a particular collection of core material.

Although the reactivity curves are the main contribution of this report, some interesting points were established. Every attempt should be made in the design of a reactor-containment system to avoid places where, in the event of a meltdown, the fuel (UO_2) can

collect in a spherical shape greater than 15 centimeters in radius unless dilution can be proven. The same idea on avoiding possible collection sites shaped as cylinders or slabs can be made. For cylinders the critical radius is 10 centimeters, and for slabs approximately 8 centimeters thick. When comparing mass for different geometrical shapes, it is clear that slabs can contain more material for the critical dimension when the lateral dimensions are free to expand. And in fact it was shown that a slab of $^{235}UO_2$ and four parts molybdenum was not critical until a slab 30.48 centimeters was formed.

Another point shown was that any form of dilution will permit a larger size and mass before becoming critical. In conjunction with this it was shown that the type of diluent is of secondary importance. Finally it should be stated that the curves presented were not intended to give a final answer as to determining criticality of molten cores. But merely to guide and assist in designing reactor-containment systems that will not be critical on meltdown. It is recognized that a more detailed calculation would be needed in obtaining the final design.

Lewis Research Center,
National Aeronautics and Space Administration,
Cleveland, Ohio, February 12, 1970,
126-15.

APPENDIX - DESCRIPTION OF COMPUTER CALCULATIONS

The RP-4 program is a one-dimensional consistent P-1 solution of the neutron diffusion problem written by D. Fieno of Lewis. The basic equations of the consistent P-1 solution are given in reference 4. The program incorporates its own nuclear cross section library which is based on 72 neutron energy groups. The program has been used reliably at Lewis and, in addition, was checked in our case by the calculation of a Godiva (ref. 6) assembly. This assembly, consisting of a bare sphere of fully enriched uranium metal which has an experimental reactivity of 1.0, was calculated by RP-4 to have a reactivity of 0.995, which is very close agreement for our purposes.

Because of the large energy group structure, the RP-4 program requires considerable calculation time, on the order of 25 minutes for each case. However, RP-4 has a built-in editing scheme that can be used to obtain diffusion coefficients and other nuclear cross-section information with spatial and energy weighting into a reduced number of fast groups. We used RP-4 to edit this information into seven fast and one thermal group for combinations of fuel enrichment and diluent ratio with a constant mass of $^{235}UO_2$ for each combination. This procedure was used since the diffusion coefficients and other averaged nuclear data do not vary significantly (less than a fraction of a percent) with changes in system size (i.e., $^{235}UO_2$ mass). However, significant changes do occur for variations in enrichment and diluent ratio. The condensed group information was then used as input to the RP-1 program.

The RP-1 program is a solution to the one-dimensional neutron diffusion theory equations and was written by D. Fieno. This program has no cross-section library of its own and must be given nuclear data as input. To check the accuracy of the RP-1 solution, the cases calculated by RP-4 were recalculated with RP-1. The results agreed to within a few percent with RP-1 having the higher result.

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