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041	Gd / ROW 1 (EDGE)	14	15.22
016	Cd / ROW 8 (MIDDLE)	14	21.94
020	BORAL / ROW 8 (MIDDLE)	14	28.29
042	Gd / ROW 8 (MIDDLE)	14	20.91
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(1) "ROW" REFERS TO LINE OF LATTICE HOLES RUNNING PARALLEL TO LONG SIDE OF TEMPLATES. (SEE FIGURE 2). THE "EDGE" ROW MEANS THE ROW OF TEMPLATE HOLES NORMALLY OCCUPIED BY THE FIRST ROW OF FUEL. THE MIDDLE ROW, LIKEWISE, REFERS TO THE ROW OF HOLES IN THE ARRAY CENTER.

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(3) A Gd PIN IS POSITIONED AT THE CENTRAL POSITION OF EVERY 5x5 CLUSTER OF FUEL. THE ENTIRE ARRAY, IN TURN, IS PUT TOGETHER IN A REPITITIOUS MANNER, OF THESE 5x5 CELLS.

(4) THE BORAL AND Cd PLATES STRETCHED TO THE EDGE OF THE FULL 508 mm TEMPLATE LENGTH. CONSEQUENTLY, SOME PORTION OF THE PLATE WAS OUT IN THE WATER REFLECTOR REGION. THE Gd PINS, HOWEVER, DID NOT EXTEND OUT INTO THE WATER BEYOND THE FUEL.

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Fig 3 Critical array size for 15 39-mm lattice for various neutron absorber positions in fuel array

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5. Critical Experiments Using High-Enriched Uranyl Nitrate with Cadmium Absorber, *W. E. Converse, R. C. Lloyd, E D. Clayton (BNWL), W. A. Yuill (Allied Chem)*

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This series of experiments was performed using a 241 8-mm-i.d stainless-steel (Type 304) cylindrical vessel reflected on the sides and bottom with at least 200 mm of water or water plus cadmium nitrate The experimental vessel thickness was 0.79 mm on the sides and 6 35 mm on the top and bottom. The cylindrical reflector tank was made of carbon steel with a 1016-mm o d and 2 77-mm-thick walls on the sides and bottom. For all experiments, the reflector solution height was maintained at the top of the experimental vessel. A detailed description of the experimental assembly is shown in Fig. 1.

The uranyl nitrate used in these experiments contained ~500.0 g U/litre with 0 1 *M* excess nitric acid. The average enrichment was ~85 wt% ^{235}U

Each experiment was performed by incrementally increasing the uranium solution level in the experimental vessel until the critical level was determined. A safety and

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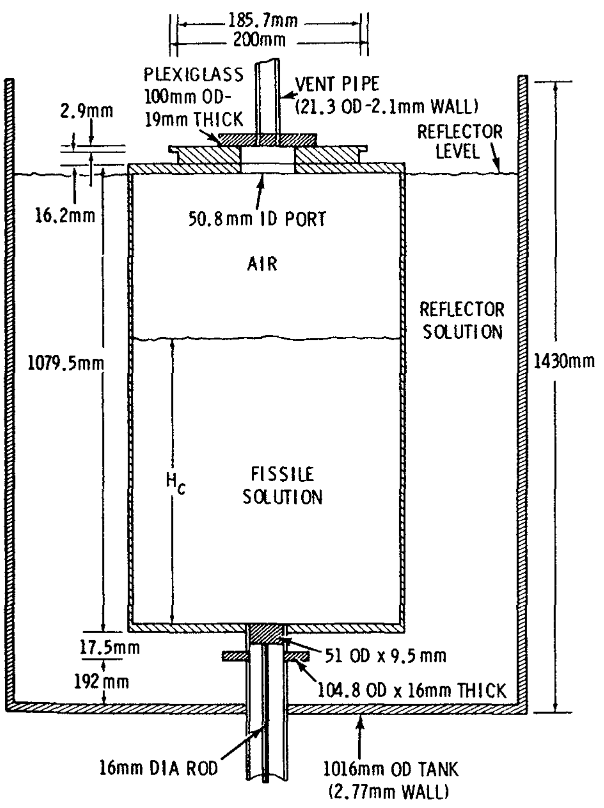

Fig 1. *Description of experiment*

control rod (positioned external to the experimental tank) were used At each solution level, the safety rod was fully withdrawn and the control rod was withdrawn until it no longer influenced the reactivity of the experiment. The first two experiments were done with no cadmium added to the uranyl nitrate The first experiment had water only in the reflector tank and the second had water plus cadmium nitrate (at ~15 g Cd/litre) in the reflector tank For the remaining experiments, water alone was used in the reflector tank and the cadmium concentration in the uranyl nitrate to the experimental vessel was varied The results of the experiments including details of solution concentrations and critical solution heights are presented in Table I

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0	0	22.33	1.024 ± 0.008
0	15	30.58	1.040 ± 0.010
1.98	0	28.20	0.971 ± 0.010
3.98	0	37.75	0.998 ± 0.012
6.35	0	76.05	0.985 ± 0.007

*As uranyl nitrate, $UO_2(NO_3)_2$, 478 7 g U/litre, S G. = 1 6592, 0 194 *M* excess nitrate Isotopics (wt%) = ^{234}U = 0.94, ^{235}U = 85 02, ^{236}U = 3 46, ^{238}U = 10 58

[a] As cadmium nitrate, $Cd(NO_3)_4$ $4H_2O$

[b] KENO-IV calculations using 18-group-averaged cross sections from EGGNIT One sigma (1σ) values are derived from KENO code statistics.

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It is a well-known fact that the maximum reactivity (k_∞) attainable in a homogeneous fuel-moderator mixture can increase when the fuel is lumped A familiar illustration of this effect is the observed difference in the minimum critical enrichments for homogeneous (~1 wt% ^{235}U) and heterogeneous (natural U) uranium-water mixtures[1] There are many computational methods and considerable experimental data available to the criticality safety analyst to allow the evaluation of the safety of either homogeneous fuel-moderator mixtures or lattices of fuel rods immersed in a moderator. However, the analyst is sometimes faced with the practical problem of evaluating an "intermediate" case consisting of a mixture of fuel particles in a moderator Both the absence of experimental data and the lack of simple computational methods to address this specific problem have led us to develop a computational model for this type of system This paper presents the results of a series of survey calculations performed using the model to examine the reactivity effects of spherical uranium fuel particles in a water moderator The computational model is described in a companion paper[2] Some of the variables considered in the survey, in addition to spherical particle diameter, include fuel material form (metal, oxide), uranium enrichment, and particle volume fraction (degree of moderation)

Reactivity changes from lumping fuel into particles result from two primary effects The first effect is that a larger fraction of the neutrons is slowed down in the moderator, thus escaping resonance capture in ^{238}U This increases the reactivity of the system The second effect is that neutrons that slow down in the moderator travel some distance in the moderator before encountering a fuel particle. Therefore, a larger fraction of the neutrons is absorbed in the moderator in the heterogeneous mixture than in the homogeneous system. This effect produces changes in the thermal utilization and tends to decrease reactivity

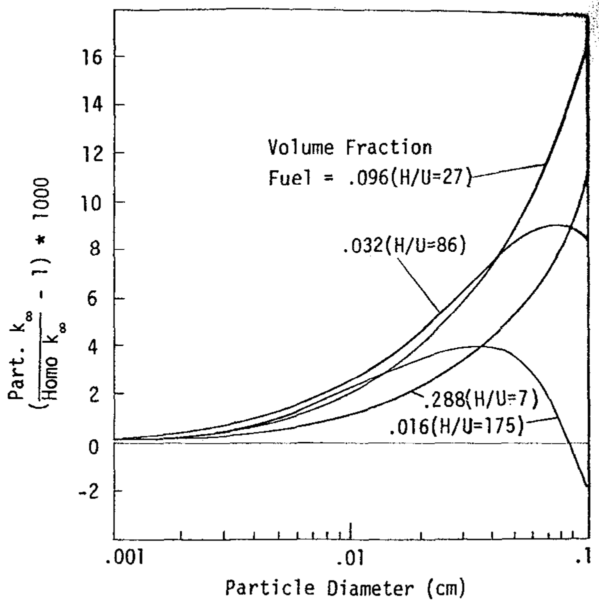

Fig. 1 The ratio of the heterogeneous k_∞ to the homogeneous k_∞ for a solution of 5 wt% enriched UO$_2$ spherical particles in water.

Modified versions of the GRANIT[3] and EGGNIT[4] computer codes were used to calculate the thermal and epithermal parameters and k_∞ The epithermal resonance calculation in EGGNIT was done using integral transport theory[5] The thermal calculation in GRANIT was also done using an integral transport theory[6] method k_∞ was calculated based on two-group parameters from a 30-thermal-group and 68-epithermal-group spectrum In both the thermal and epithermal calculations a Dancoff factor is required for the fuel particles An expression for the Dancoff factor for an infinite array of spherical particles is derived and discussed in a companion paper[2]

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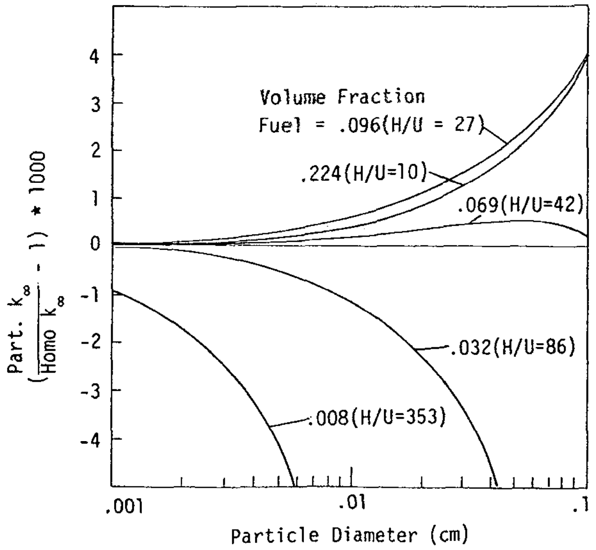

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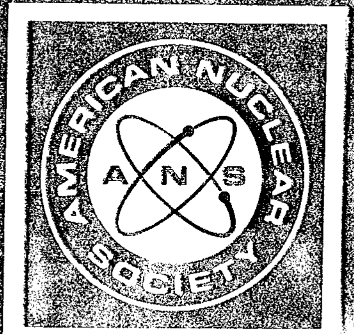

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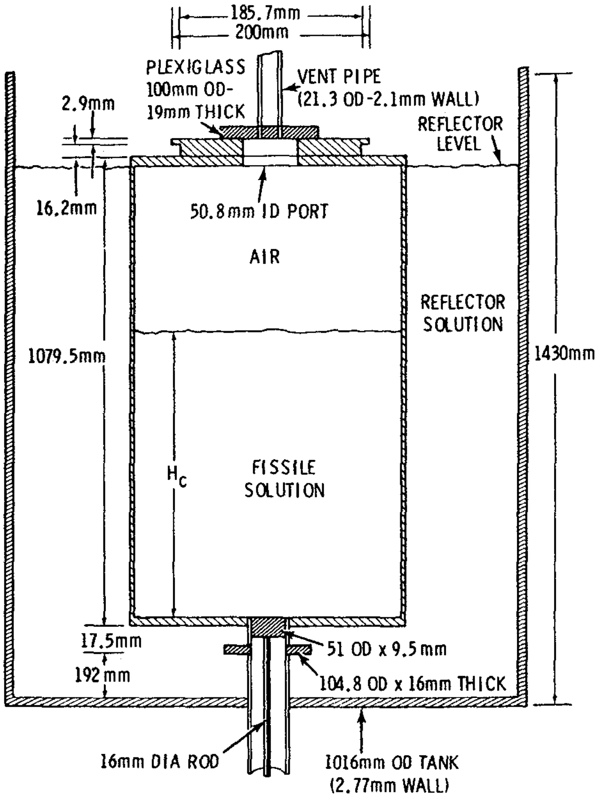

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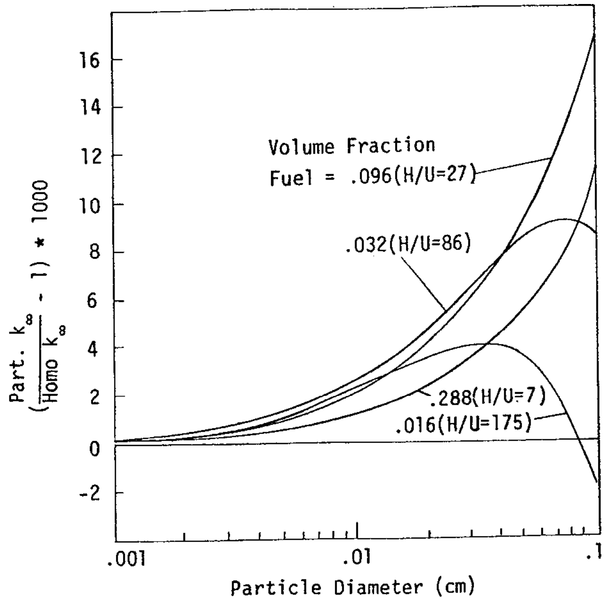

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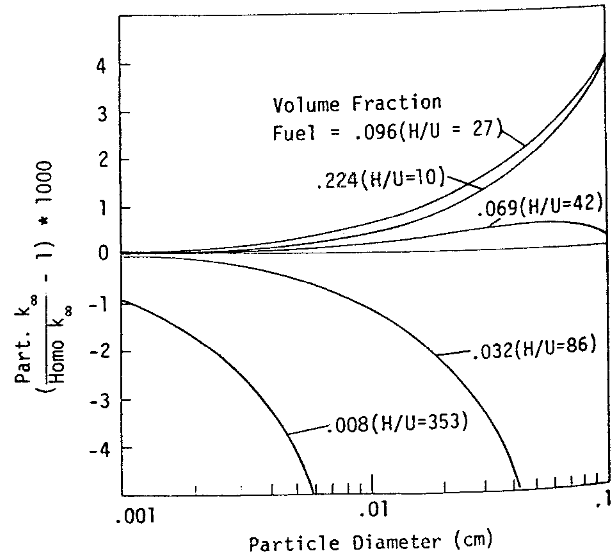

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