75

NASA TECHNICAL MEMORANDUM

NASA TM X-52302

NASA TM X-52302

AN ANALYSIS OF CRITICAL FUEL SOLUTION REACTORS CONTAINING ARRAYS ON VOID TUBES

by Wendell Mayo
Lewis Research Center
Cleveland, Ohio

TECHNICAL PAPER proposed for presentation at Thirteenth Annual Meeting of the American Nuclear Society
San Diego, California, June 11-15, 1967

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION · WASHINGTON, D.C. · 1967

AN ANALYSIS OF CRITICAL FUEL SOLUTION REACTORS CONTAINING ARRAYS OF VOID TUBES

by Wendell Mayo

Lewis Research Center
Cleveland, Ohio

TECHNICAL PAPER proposed for presentation at

Thirteenth Annual Meeting of the American Nuclear Society
San Diego, California, June 11-15, 1967

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

E-3808

AN ANALYSIS OF CRITICAL FUEL SOLUTION REACTORS CONTAINING ARRAYS OF VOID TUBES

by Wendell Mayo

Lewis Research Center
National Aeronautics and Space Administration
Cleveland, Ohio

The problem of neutron streaming out of empty tubes penetrating critical reactors has been considered by many people in the past few years. Homogenization is one possible method of treating the void effect but this generally leads to poor predictions of reactor criticality. Behrens (ref. 1) derived equations for anisotropic diffusion coefficients which may be used in criticality calculations to account for neutron streaming. Zimmerman treated the critical cylindrical reactor with a single void tube on the axis (ref. 2). However, the reactor geometry sketched in Figure 1 is of such a heterogeneous nature that Behrens' and Zimmerman's methods do not work very well. The 37 void tubes in the reactor shown are 7.62 cm in diameter and arranged with a triangular spacing. The height of the core depends on the UO_2F_2 concentration in the fuel solution surrounding the void tubes. An aqueous solution of uranyl fluoride enriched to 93.2 percent in uranium 235 is used as fuel.

Peak and Cohen (ref. 3) calculated a 37 tube configuration such as shown in Figure 1 using Behrens' method. The observed critical height of the core, radially reflected by 15.24 cm of water, was about 75 cm. However, the calculated multiplication factor for this core was 0.94

X-52302

compared with the experimental value of unity. The multigroup leakage rates, parallel to the void tubes, were greatly overestimated by the calculations and indicated that an approach different from Behrens' was required.

The purpose of this study is to investigate a calculational method involving direct application of existing computer programs to extreme cases of heterogeneous voids. Extreme heterogeniety, here, refers to arrays of large diameter voids in relatively compact reactors where the total leakage rate per source neutron may approach 50 percent.

The geometry of the reactors considered in this study is illustrated by the sketch at the top of Figure 2 which shows a 19 void tube configuration. Another ring of 18 additional void tubes would give a 37 void tube assembly while omitting the six corner tubes from the 37 void tube array results in a 31 void tube array. The pitch (center-to-center tube spacing) was either 9.65 cm or 10.92 cm. The core tank is 76.2 cm in diameter.

The method used in the present analysis consists primarily of two parts. The first is to obtain the multigroup axial leakage rates out of the void tubes using transport theory solutions of an explicit cell containing a void tube and proportional part of the surrounding fuel solution. The second step is to incorporate these axial leakage rates into a gross cylindrical reactor calculation of the effective multiplication factor.

The dotted lines around the center tube in the reactor geometry shown in Figure 2 are lines of symmetry, i.e., cell boundaries. It is assumed that the entire void tube array is made up of these cells. With the hexagonal boundary of the cell converted to an equivalent circular boundary the cell may be represented in r-z geometry. The circular boundary is chosen to conserve the proportional volume of fuel associated with the void tube. A multigroup transport solution of the explicit r-z cell as shown, which is extracted from the geometry of a critical reactor, gives the axial leakage rate per source neutron out of the cell. Five energy group S_4P_1 transport theory calculations with the TDSN (ref. 4) program are used. Cross sections are obtained from the GAM-II (ref. 5) and TEMPEST (ref. 6) programs. An "isotropic return current" boundary condition is used at the curved surface and a "no return current" boundary condition is used at the cell ends. The axial symmetry of the cell is exploited so that only the half height of the cell need be specified. The axial leakage rate from the cell is assumed to be applicable to the entire void region of the core consisting of 19, 31, or 37 cells.

The gross cylindrical reactor model used in the one-dimensional radial diffusion theory calculations consists of a homogenized void tube-fuel solution region, a void-free fuel solution driver region, the aluminum core tank and the water reflector. The representation of the complex outer boundary of the void tube array by a circular boundary so as to permit one-dimensional calculations is probably the poorest approximation in this calculational model, particularly for the 37 void tube array which nearly fills the core tank. The axial leakage rate out of the homogenized void region is based on the r-z cell solution. An axial leakage probability per unit flux, or more precisely, an axial leakage cross section by group is defined by dividing the axial leakage rate per source neutron by the volume integral of the group flux. Spatial flux weighting, instead of volume averaging, is used to homogenize the voided region in order to account for the fuel "disadvantage" factors.

The high leakage rates out of the void tubes, for the shorter reactors with the more concentrated fuel solutions, depresses the flux in the void. The "disadvantage" factors (ratio of the average flux in the fuel to the average flux in the cell) for the fuel are thus greater than unity. This effect is evident in Figure 3 which shows experimental thermal fluxes obtained by dysprosium activations in a 37 tube assembly for two fuel concentrations (ref. 7). The traverse shown was made along a major radius of the hexagonal array. The fuel concentrations are given as the ratio of the hydrogen-to-uranium 235 atoms in the solution. Note that the flux peaking is considerably greater for the shorter reactor at H/U-235 = 150 than for the taller core at H/U-235 = 720. Additional experiments have shown that the magnitude of the flux peaking in the fuel solution varies with core height at a given fuel concentration. A seven tube reactor at H/U-235 = 150 with a height of 16 cm shows an increase of the peak flux of 14 percent compared to this 37 tube core which is 32 cm tall. The r-z cells are used to obtain the "disadvantage" factors for use in the homogenized void region of the reactor calculations.

Figure 4 shows calculated multiplication factors for 19 tube reactors as a function of the atom ratio of hydrogen to uranium 235. Three curves are shown:

Curve A is obtained using the geometric buckling with energy dependent extrapolation distances to compute the axial leakage. The void tube region is homogenized by volume averaging the void tube cells and one-dimensional radial calculations are performed. The critical heights of the voided reactors are shown on this curve.

Curve B is obtained from radial calculations using the reactor calculational geometry in the same manner as for Curve A except that the axial leakage cross sections obtained from the r-z cell are used. Homogenization of the voided region is by spatial flux weighting.

Curve C is for unvoided reactors. The infinite multiplication factor of the fuel solution does not change with void content; therefore, the unvoided critical reactors have the same process rates per source neutron as the critical voided reactors using the same fuel solution regardless of the void content. The unvoided reactors are relatively simple systems from a calculational standpoint, and therefore provide an excellent basis with which to compare results of calculations for voided reactors.

A positive reactivity correction for the 1.27 cm thick aluminum core tank bottom worth is included in all of these calculations; S_4P_0 axial calculations give tank bottom corrections of 1.8 percent, 0.8 percent and 0.4 percent ΔK at H/U-235 values of 324, 789, and 1082, respectively.

The discrepancy between curves B and C is about 0.5 percent as compared to several percent for Curves A and C where the r-z cell results are not incorporated. Comparison of Curve C with the experimental value of the multiplication factor of unity shows a reasonably uniform positive deviation of from about 0.2 percent to 0.8 percent ΔK.

An examination of the axial leakage cross sections for a reactor with a critical fuel solution height of 41.9 cm shows qualitatively why the reactor calculations using the r-z cell model predicts smaller and more accurate multiplication factors than those computed using the completely homogenized cell model. Figure 5 is a histogram showing leakage cross sections for each of the five neutron groups used in the calculations. The lower energy of each group is shown along the abscissa. The leakage cross sections are calculated using the indicated equations. The trends in this histogram are typical of all of the cases examined. The r-z cell leakage cross sections are almost constant for the first four groups but decrease in the thermal group; this decrease is attributed to the shorter mean free path of thermal neutrons in the fuel solution. On the other hand, the homogenized cell values show more variation with the greatest, and most important difference being a smaller thermal group cross section. Since the group flux integrals are nearly the same in these two calculations, the relative leakages rates by group are indicated by the histogram. For example, the ratio of the axial leakage cross sections for group 5 by the two methods of calculation is about 2.8 which is also the ratio of the axial leakage rates.

The reactor results to this point have been for 19 void tube configurations. Several additional reactors were calculated. Table 1 presents a summary of the reactors studied and the calculated multiplication factors for both voided and unvoided reactors. All the results have been corrected for the worth of the core tank bottom. The largest discrepancy between the voided and unvoided calculations of 1.2 percent occurs for the third case. Most of this discrepancy can be attributed to the geometric approximations at the void region-driver region interface; an additional contributing factor may be the high cell multiplication factor of 1.098 which indicates substantial leakage radially into the thin driver region. This is to be contrasted with the second case which has 31 void tubes; this case has a larger void tube spacing and has about the same thickness of driver region. The cell multiplication factor in this second case is nearer to unity indicating little interaction of the driver and voided region. The discrepancy in the calculated reactor results is only 0.4 percent. The last case listed is essentially the same configuration as the second (though the fuel solution, critical height and reflector are different), but has a high cell multiplication factor of 1.152 and shows a 1 percent discrepancy. Although the number of cases studied is limited, a reasonably consistent finding is that the major part of the small observed differences are associated with the geometric representation in the gross cylindrical reactor calculations rather than in the r-z cell representation.

It should be pointed out that while one-dimensional radial calculations were used in this gross cylindrical reactor calculation, the method does not preclude two-dimensional calculations. A triangular mesh diffusion theory program would permit more accurate representation of the geometry of the reactors.

Considering the wide range of reactors calculated the model used appears to be reasonably accurate and amenable to direct application. Further tailoring of the model to a particular reactor or to a limited range of reactors could result in improved accuracy. The use of this technique for other reactors containing voids depends primarily upon an adequate definition of the appropriate r-z cell.

TABLE I

CALCULATED MULTIPLICATION FACTORS FOR CRITICAL REACTORS

(r-z Cell Method)

H/U-235	Reflector	No. Tubes	Pitch	Height	K_{cell}	K_{eff}
324	water	19	9.65 cm	21.0 cm	0.624	0.999
		0	---	15.5		0.999
324	water	31	10.92	28.4	1.024	1.003
		0	---	15.5		0.999
324	none	37	9.65	40.8	1.098	1.014
		0	---	15.7		1.002
789	water	37	9.65	64.7	1.093	1.016
		0	---	22.4		1.007
789	none	31	9.65	67.9	1.116	1.010
		0	---	22.8		1.008
1082	water	31	10.92	83.7	1.152	1.016
		0	---	29.1		1.006

References

1. E. L. ZIMMERMAN, "Boundary Values for the Inner Radius of a Cylindrical Annular Reactor," ORNL-2484, Oak Ridge National Laboratory (1958).

2. D. J. BEHRENS, "The Effect of Holes in a Reacting Material on the Passage of Neutrons, with Special Reference to the Critical Dimensions of a Reactor," AERE-T/R-103, United Kingdowm Atomic Energy Authority (1958).

3. R. G. BARDES, et al, "Tungsten Nuclear Rocket, Phase 1, Part 1," GA-6890 (NASA CR-54909), General Dynamics Corporation (1966).

4. C. E. BARBER, "A FORTRAN IV Two-Dimensional Discrete Angular Segmentation Transport Program," TN D-3573, National Aeronautics and Space Administration (1966).

5. G. D. JOANOU and J. S. DUDEK, "GAM-II. A B_3 Code for the Calculation of Fast-Neutron Spectra and Associated Multigroup Constants," GA-4265, General Dynamics Corporation (1963)

6. R. H. SHUDDE and J. DYER, "TEMPEST. A Neutron Thermalization Code," Atomics International (1960).

7. T. A. FOX, R. A. MUELLER and C. H. FORD, "Critical Mass Studies with the NASA Zero Power Reactor-II," TN D-3555, National Aeronautics and Space Administration (1966).

E-3808

CALCULATED MULTIPLICATION FACTORS
FOR CRITICAL REACTORS

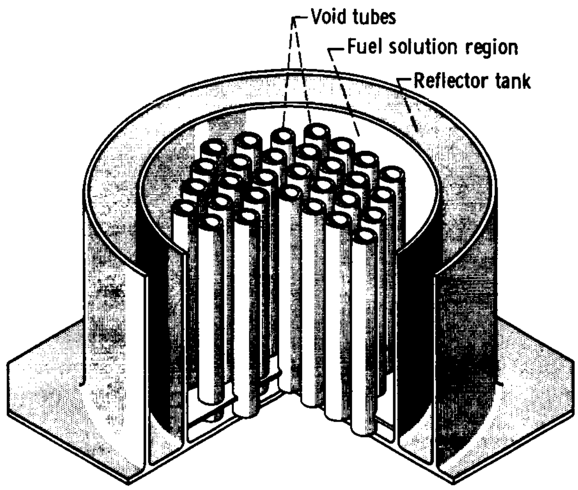
r - z CELL METHOD

H/U-235	REFLECTOR	NO. OF TUBES	PITCH, CM	HEIGHT, CM	K_{CELL}	K_{EFF}
324	WATER	19	9.65	21.0	0.624	0.999
		0	—	15.5		0.999
324	WATER	31	10.92	28.4	1.024	1.003
		0	—	15.5		0.999
324	NONE	37	9.65	40.8	1.098	1.014
		0	—	15.7		1.002
789	WATER	37	9.65	64.7	1.093	1.016
		0	—	22.4		1.007
789	NONE	31	9.65	67.9	1.116	1.010
		0	—	22.8		1.008
1082	WATER	31	10.92	83.7	1.152	1.016
		0	—	29.1		1.006

TABLE 1

CS-43354

NASA ZPR-II REACTOR WITH VOIDS



CD-9096

Figure 1

E-3808

MODEL GEOMETRY

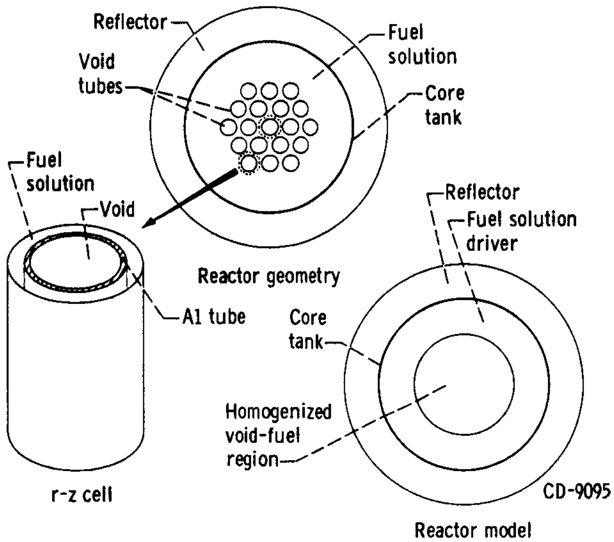


Figure 2

THERMAL NEUTRON FLUX DISTRIBUTIONS

37 TUBES, 9.65 CM PITCH

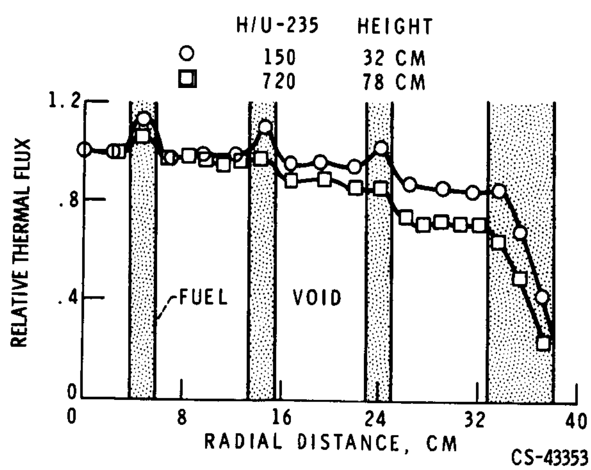


Figure 3

E-3808

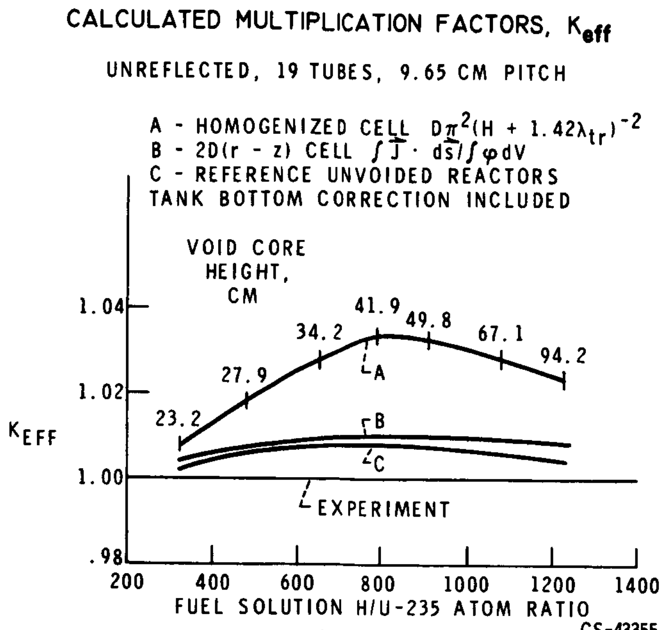


Figure 4 CS-43355

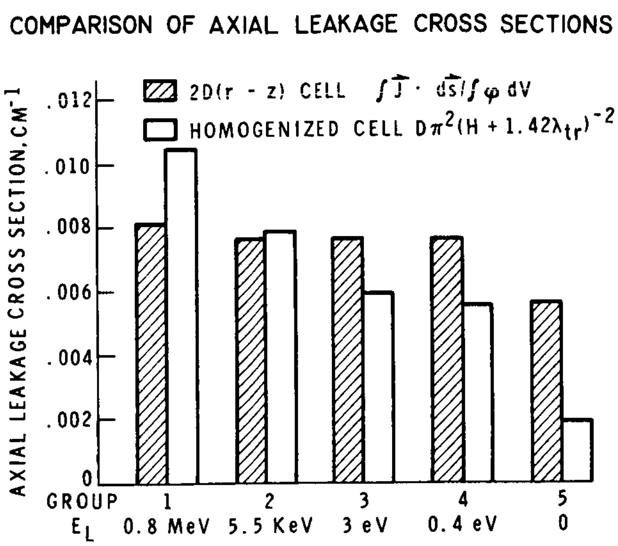


Figure 5 CS-43356