NASA TECHNICAL NOTE

NASA TN D-3555

CRITICAL MASS STUDIES WITH THE NASA ZERO POWER REACTOR II

II - HETEROGENEOUS ARRAYS OF CYLINDRICAL VOIDS

by Thomas A. Fox, Robert A. Mueller, and C. Hubbard Ford

Lewis Research Center
Cleveland, Ohio

CRITICAL MASS STUDIES WITH THE NASA ZERO POWER REACTOR II

II - HETEROGENEOUS ARRAYS OF CYLINDRICAL VOIDS

By Thomas A. Fox, Robert A. Mueller, and C. Hubbard Ford

Lewis Research Center
Cleveland, Ohio

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

CRITICAL MASS STUDIES WITH THE NASA ZERO POWER REACTOR II

II - HETEROGENEOUS ARRAYS OF CYLINDRICAL VOIDS

by Thomas A. Fox, Robert A. Mueller, and C. Hubbard Ford

Lewis Research Center

SUMMARY

The NASA Zero Power Reactor II (ZPR-II) has been used to determine experimentally several critical cylindrical configurations of aqueous fuel solutions that contain heterogeneous arrays of voids. These voids are cylindrical, are symmetrically arranged parallel to the axis of the reactor, and extend the height of the core. The study covered a wide range of highly enriched (93.2 percent U^{235}) aqueous uranyl fluoride fuel concentrations. The ZPR-II core has a diameter of 76.2 centimeters and a height that permits the solution level to be increased to about 90 centimeters. The reactor system may be operated either bare or radially reflected by about 15 centimeters of water. The specific reactor void configurations consisted of symmetrical arrays of 1, 7, 19, 31, and 37 tubes approximately 7.6 centimeters in diameter arranged in hexagonal geometry with pitches of 9.652 or 10.922 centimeters.

In addition to the critical mass and geometry, data are presented on the thermal neutron flux distributions in the central radial plane, and on the variation of void reactivity importance with radial position. These data explain qualitatively some of the reactivity effects associated with the different void spacings.

INTRODUCTION

Heterogeneous reactor systems in which gas-cooled high-temperature fuel elements are operated in conjunction with surrounding low-temperature moderating media have been designed and operated for aircraft propulsion (ref. 1) and for portable land-based power plants (ref. 2). For scientific missions of interest to the national space program, nuclear rocket stages provide a large potential advantage over chemical stages (ref. 3). A program of research was undertaken to determine the feasibility of an isotopically enriched tungsten, water-moderated reactor for use as a nuclear rocket (ref. 4).

One of the physics problems of these heterogeneous reactors is concerned with the analytical treatment of the essentially void gas-flow passages of the fuel elements. This problem is complicated by the strong anisotropic neutron currents generated in these gas cooled fuel elements by neutron absorbing and scattering materials. However, no data using absorbing and scattering materials in the voided regions are presented in this report.

One of the objectives considered in the design of the NASA Zero Power Reactor-II (ZPR-II) was the provision of a tool for the study of these anisotropic heterogeneous effects. The ZPR-II is a flexible, fuel solution critical system; the configurations, as described in the next section, consist of a homogeneous reactor with which the criticality effects of heterogeneous arrays of voids and absorbers are studied. The configurations provide the experimental data for checking various calculational models even though the fissionable material is located in the moderating media.

This is the second report that covers critical mass studies using the ZPR-II. The first (ref. 5) contained experimental data for clean homogeneous cores. The present report contains the experimental data for cylindrical voids in hexagonal geometry in a core studied over a wide range of fuel concentrations. The voided regions are achieved by means of thin-walled empty aluminum tubes in the core region.

DESCRIPTION OF THE EXPERIMENTAL EQUIPMENT

The ZPR-II is a solution critical system in which the fuel is an aqueous solution of uranyl fluoride $(UO_2F_2 + H_2O)$. The uranium is fully enriched (93.2 percent) in the uranium 235 (U^{235}) isotope, and the fuel-water concentration can be varied over a wide range. The fuel solution is stored in a geometrically safe configuration of tanks and pipes The basic operation consists of remote pumping of the fuel solution from the storage system into a 76.2-centimeter-diameter cylindrical tank in which criticality is achieved. The ZPR-II reactor tank is schematically shown in figure 1. Reactor control is achieved by careful manipulation of the solution level in the core tank. The addition or removal of solution is accomplished by a positive displacement pump with the capability of fuel transfer in very small increments when necessary. The reactor can be operated as an unreflected system or as a system radially reflected by about 15 centimeters of water.

The ZPR-II reactor vessel consists of two concentric aluminum cylinders welded to a common bottom plate to form the cylindrical core region and the annular reflector region. Both surfaces of the inside cylinder and the upper surface of the bottom plate were machined with precision in the core region (see table I for dimensions and tolerances).
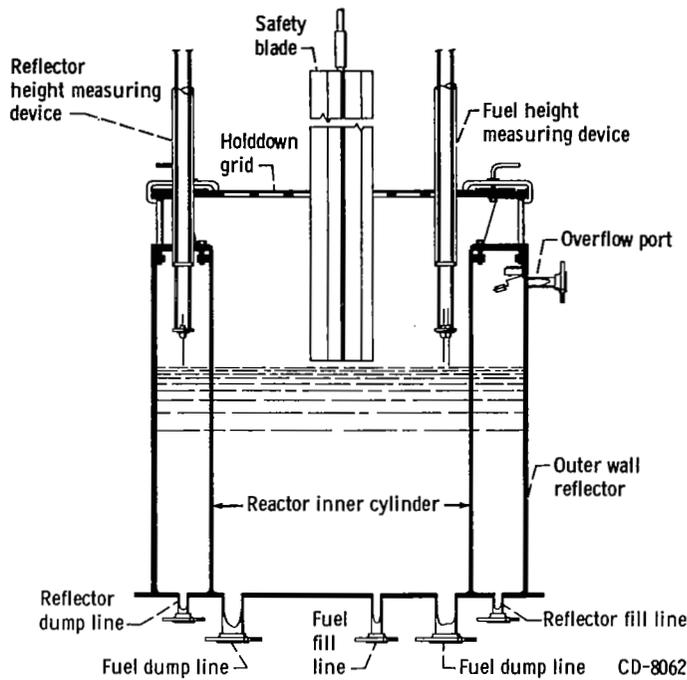
Figure 1. - ZPR-II reactor tank.

TABLE I. - DIMENSIONS OF ZPR-II COMPONENTS

Component	Diameter, cm		Wall thickness, cm
	Inside	Outside	
Reactor inner cylinder (15 cm from bottom)	76.152±0.013	77.559±0.025	0.704±0.018
Out of roundness near top	76.137±0.076	-----------	-----------
Reactor outer cylinder	107.95±0.64	110.5±0.64	1.27
Reactor bottom { Core region	-----------	-----------	1.270±0.064
{ Reflector region	-----------	-----------	1.270±0.157
Void tubes { Wall	6.965±0.023	[a]7.658±0.023	0.347±0.023
{ Bottom	-----------	-----------	[b]0.953
Spacer grid holes { 10.922-cm grid	(c)	(c)	(c)
{ 9.652-cm grid	(d)	(d)	(d)

[a]Does not include 0.002- to 0.005-cm Glyptal coating.

[b]Rounded edges.

[c]Pitch, 10.922±0.0025 cm; diameter, 7.691±0.0025 cm.

[d]Pitch, 9.652±0.0025 cm; diameter, 7.706±0.0025 cm.

3

Fuel and water are introduced into and removed from the core and reflector, respectively through penetrations in the bottom plate. A description of the basic ZPR-II system and the method of operation can be found in reference 5.

The ZPR-II has been designed to permit the placement of heterogeneous arrays of voids and absorbers into an otherwise homogeneous core. The voids in this study were made of aluminum tubes, with nominal outside diameters of 7.6 centimeters, that are capped at the bottom end. These tubes were positioned and constrained from movement by a set of matched aluminum grids within the existing reactor vessel structure (fig. 2(a)) A set of grids includes two spacer grids and a holddown grid. The grids can be changed to accomodate different geometrical arrangements of the void tubes. This study used grids giving hexagonal arrays with pitches or center-to-center tube spacings of 9.652 or 10.922 centimeters (3.8 or 4.3 in.). The maximum number of void tubes that were inserted into the core was 37 as shown in figure 2(b) although partial arrays of less than 37 tubes were also used. The dimensions of the aluminum tubes and spacer grids are also given in table I. All components subject to corrosion are protected by a coating of Glyptal.

A description of the fuel solution and water-height-measuring devices and the fuel solution temperature measuring system is presented in reference 5. Briefly, the height-measuring devices are direct-current probes mounted on precision lead screws that provide readouts to ±0.001 centimeter for fuel height and to ±0.01 centimeter for water height. The fuel solution temperature is measured by a thermocouple with readings accurate to ±0.25° C but capable of detecting temperature changes of 0.005° C.

EXPERIMENTAL PROCEDURES

The criticality studies reported have experimentally determined critical core heights for several hexagonal arrays of voided tubes using uranyl fluoride solution external to the tubes. As is typical of many criticality studies, this experimental program required the determination of the critical mass of U^{235} as a function of fuel concentration for several geometric configurations. The basic clean core geometry was the 76.2-centimeter-diameter cylindrical core tank with a 15-centimeter annular reflector region. The void array of interest was then introduced into the empty reactor core, and the critical mass was determined. The precision of an experimental determination of critical mass is dependent on the accuracy with which uranium concentration, volume, and temperature of the fuel solution at the steady-state critical condition may be determined, and on uncertainties in the determination of the steady-state critical condition itself.

After attaining a steady-state critical condition, the fuel solution height and temperature are measured remotely. To determine the fuel concentration at the moment of crit-

4

Height-measuring
devices

Safety blade

Holddown
grid

Upper
spacing
grid

Lower
spacing
grid

Voided
tubes

Reflector
fill line

Fuel
fill
line

Fuel
dump
valves

CD-8247

(a) Cutaway drawing.

CD-8245

(b) Plan view of 37-tube array (typical).

Figure 2. - ZPR-II tube array.

icality is unnecessary since changes in concentration vary gradually over days of operation.

For most of the configurations involved in these criticality studies, the limiting factor in the critical mass determination was the measurement of the fuel solution concentration. Two methods of measuring fuel solution concentration are employed. One method makes use of the variation of solution density with fuel concentration. The second method is a gravimetric analysis technique. Both methods are described briefly in reference 5.

The determination of the solution volume in these experiments required an accurate measure of the displacement volume of the void tubes. The void tube dimensions stated in table I were checked by a displacement calibration for all 37 aluminum tubes with water instead of fuel in the reactor tank. The agreement between calculated and experimentally determined volumes at several solution heights was excellent. Measured critical height is, therefore, a known function of solution volume.

The steady-state critical condition was determined by operating the reactor system without an external neutron source at a constant neutron flux and power level for at least 15 minutes. For most of the experiments, critical height was not the limiting variable so that determination of the exact steady-state condition presented no unusual problem.

The study of cylindrical void arrays was limited to symmetrical configurations of 1, 7, 19, 31, and 37 void tubes. The 7-, 19-, and 37-tube arrays were achieved by completing the appropriate hexagons (fig. 2(b)).

The single tube case was the central void tube (tube 19), while the 31-tube array was achieved by removing corner tubes 1, 4, 16, 22, 34, and 37 from the 37-tube array. Spacing grids of 9.652- and 10.922-centimeter pitch were used. The measurements for each geometric array of voids covered a wide range of fuel concentrations. A hydrogen to uranium 235 atom ratio H/X of about 150 (169 grams of U^{235}/liter of solution) was the most concentrated fuel solution used. The most dilute fuel solution used was determined by the condition at which the critical height of the fuel solution reached the nominal 90-centimeter maximum height of the reactor tank.

The general experimental procedure followed these steps at each fuel solution concentration:

(1) The desired fuel solution was prepared. This required thorough mixing of the solution and a preliminary determination of the concentration.

(2) A set of grids was inserted into the reactor tank, and the reference criticality measurements were carried out with no voids inserted for both the bare and reflected cores.

(3) Criticality measurements for void arrays of 1, 7, 19, 31, and 37 tubes were made with this fuel solution concentration if possible. Again these measurements were made for both the bare and reflected cores.

(4) The first set of grids were then replaced by the second set of grids, and the entire process was repeated.

(5) Temperature coefficients of reactivity were measured as needed to correct the critical heights to a reference value of 20° C. The change in critical height per degree change in temperature was found to be approximately the same for both bare and reflected systems with the same critical height. This finding reduced the number of temperature coefficient measurements that had to be made. Many cases required very small height corrections.

(6) The fuel solution concentration was determined during or immediately following the series of measurements just described. Where experiments covered long periods of time, more than one determination of concentration was made during the course of the measurements.

(7) The fuel solution concentration was changed to a new value and the entire procedure repeated.
Experimental measurements were made at enough fuel solution concentrations so that the criticality characteristics were essentially determined for all concentrations.

Measurement of the thermal neutron flux distributions in the radial dimension were carried out for several void arrays. The experimental technique involved the use of bare dysprosium foils (5 percent dysprosium-95 percent aluminum) that are approximately 0.32 centimeter (1/8 in.) in diameter and 0.013 centimeter (0.005 in.) thick. The foils placed in fuel regions were packaged in Teflon tape for protection from the fuel. These

foils were suspended in the fuel region by mounting on an aluminum grid that was adjustable axially to permit mapping in any radial plane. The foils inserted in the void tubes presented no packaging problems and were generally suspended from a wire frame that also permitted adjustment in the axial dimension. The relative foil activities were counted by means of 2π gas-flow thin window proportional counters.

A measure of the relative importance of a cylindrical void located at various radial positions was obtained by one of two procedures. For a clean homogeneous core case, a single void tube was inserted into the reactor at the location of interest and its worth measured in terms of increased critical height. For a case with an array of void tubes, a single void tube was removed from the array at the location of interest and the worth measured in terms of decreased critical height.

RESULTS AND DISCUSSION

Criticality Data

With the fuel solution concentration as the independent variable, the critical core height and, therefore, critical mass was measured as a function of the number of void tubes in several arrays, for two pitches or spacings of the void tubes, and for either bare or reflected systems. The data obtained are given in tables II to IV. The final criticality data are also plotted according to the various geometric factors in figures 3 to 8. The various tables and graphs are discussed in greater detail in the sections that follow.

Criticality data tables. - The criticality data from the studies of heterogeneous arrays of void tubes are listed in tables II to IV. Table II presents the data obtained for 37-, 31-, 19-, and 7-void-tube arrays. This table compiles the critical heights and critical masses for a number of fuel concentrations, for 9.652- and 10.922-centimeter pitches, and for bare and reflected systems. Table III presents data for a single, centrally located void tube. Since this configuration results in criticality data quite close to that obtained for the reference or zero-void-tube configuration, the differential in critical height as a fraction of the reference critical height $\Delta h/h$ for a single void tube is also given in table III. The reference case, or zero-void-tube data, is given in table IV. This differs from the clean homogeneous cases reported in reference 5 by a very small amount because of the presence of the lower spacer grid near the bottom of the core.

Table V presents data of pertinent physical properties for the various fuel solution concentrations as determined by NASA studies. The solution densities and uranium atom concentrations were measured by gravimetric methods as described in reference 5. A table containing the atom concentrations for all constituents of the fuel solution can be

TABLE II. - CRITICALITY DATA AT 20° C

Atom ratio, H/X	Critical height, cm				Critical mass, kg				Atom ratio, H/X	Critical height, cm				Critical mass, kg			
	Bare		Reflected		Bare		Reflected			Bare		Reflected		Bare		Reflected	
	Pitch, cm									Pitch, cm							
	9.652	10.922	9.652	10.922	9.652	10.922	9.652	10.922		9.652	10.922	9.652	10.922	9.652	10.922	9.652	10.922
37 - Void-tube reactor									**19 - Void-tube reactor**								
151.6	32.46	29.41	26.37	26.15	15.40	13.94	12.49	12.38	151.6	19.40	20.14	17.77	18.36	11.82	12.28	10.81	11.18
151.8	-----	29.47	-----	26.21	-----	13.95	-----	12.40	151.8	-----	20.21	-----	18.43	-----	12.30	-----	11.21
324	40.76	36.39	32.81	31.99	9.19	8.20	7.39	7.20	324	23.20	24.12	20.98	21.77	6.72	6.99	6.07	6.30
438	48.45	-----	38.49	-----	8.11	-----	6.44	-----	476	27.68	-----	-----	-----	5.48	-----	-----	-----
476	51.23	-----	-----	-----	7.89	-----	-----	-----	482	27.93	28.90	24.87	25.78	5.46	5.66	4.86	5.04
482	51.42	45.08	40.70	38.88	7.82	6.85	6.18	5.91	484	-----	29.23	-----	26.06	-----	5.70	-----	5.08
484	52.20	45.75	41.20	39.40	7.92	6.93	6.24	5.97	650	34.32	35.35	29.97	31.07	4.99	5.15	4.36	4.52
493	-----	46.37	-----	39.88	-----	6.91	-----	5.94	718	37.90	-----	-----	-----	5.00	-----	-----	-----
511	54.37	-----	42.70	-----	7.81	-----	6.13	-----	789	41.92	42.59	35.87	36.89	5.04	5.12	4.31	4.43
636	66.94	-----	51.08	-----	7.74	-----	5.90	-----	870	-----	47.73	-----	40.91	-----	5.21	-----	4.46
650	67.27	57.75	51.49	48.31	7.61	6.53	5.82	5.46	908	49.79	50.13	41.82	42.76	5.21	5.24	4.37	4.47
718	78.16	65.23	57.58	-----	8.02	6.69	5.90	-----	993	57.94	-----	47.18	-----	5.55	-----	4.51	-----
789	90.11	74.85	64.72	59.45	8.42	6.99	6.04	5.55	1082	67.13	67.04	54.17	55.13	5.90	5.90	4.76	4.48
870	-----	-----	75.07	67.92	-----	-----	6.36	5.75	1225	94.20	92.42	70.86	71.22	7.33	7.19	5.51	5.53
908	-----	97.49	79.76	71.95	-----	7.92	6.48	5.84	1247	-----	-----	74.63	-----	-----	-----	5.70	-----
1001	-----	-----	-----	86.79	-----	-----	-----	6.39	**7 - Void-tube reactor**								
31 - Void-tube reactor									151.6	15.64	15.79	15.14	15.26	10.93	11.03	10.57	10.66
									324	18.03	18.24	17.38	17.54	5.99	6.06	5.77	5.83
151.6	26.93	26.47	22.71	23.39	13.99	13.75	11.78	12.14	482	20.84	21.07	19.98	20.16	4.67	4.73	4.48	4.52
151.8	-----	26.56	-----	23.47	-----	13.78	-----	12.16	650	24.36	24.67	23.16	23.41	4.07	4.12	3.86	3.91
324	33.45	32.44	27.78	28.43	8.26	8.01	6.85	7.01	718	26.03	-----	-----	-----	3.94	-----	-----	-----
482	41.65	39.67	34.05	34.33	6.94	6.61	5.67	5.72	789	28.29	28.54	26.65	26.85	3.90	3.94	3.67	3.70
650	53.20	49.90	42.51	42.27	6.60	6.19	5.26	5.23	870	31.17	-----	29.14	-----	3.60	-----	3.65	-----
718	59.15	-----	-----	-----	6.65	-----	-----	-----	908	32.13	32.38	29.99	30.20	3.86	3.89	3.60	3.62
789	67.94	62.28	52.61	51.31	6.96	6.38	5.38	5.25	993	35.71	-----	-----	-----	3.92	-----	-----	-----
870	-----	-----	59.63	-----	-----	-----	5.54	-----	1082	39.67	40.20	36.39	36.84	4.00	4.06	3.67	3.72
908	85.41	76.88	63.21	60.82	7.61	6.85	5.62	5.41	1217	49.22	-----	-----	-----	4.42	-----	-----	-----
1001	-----	-----	75.31	-----	-----	-----	6.08	-----	1225	49.20	49.69	44.11	44.54	4.39	4.43	3.91	3.97
1082	-----	-----	88.39	83.69	-----	-----	6.61	6.26	1394	73.54	-----	61.95	-----	5.78	-----	4.87	-----
1082	-----	-----	-----	83.09	-----	-----	-----	6.21	1490	-----	-----	72.58	72.87	-----	-----	5.34	5.36

TABLE III. - CRITICALITY DATA AT 20° C FOR 1-VOID-TUBE REACTOR

Atom ratio, H/X	Critical height, cm		Critical mass, kg		Differential height,[a] $\Delta h/h$	
	Bare	Reflected	Bare	Reflected	Bare	Reflected
151.6	14.37	14.14	10.68	10.50	0.0241	------
324	16.33	16.05	5.77	5.67	.0245	0.0203
482	18.50	18.15	4.42	4.33	.0249	.0208
650	21.20	20.73	3.76	3.68	.0326	.0272
789	23.96	23.33	3.51	3.42	.0377	.0314
870	25.78	25.03	3.43	3.33	.0437	.0364
908	26.60	25.80	3.39	3.29	.0415	.0345
1225	37.39	35.58	3.55	3.38	.0613	.0505
1490	57.27	52.15	4.48	4.08	.0902	.0706
1571	-----	69.02	-----	5.13	------	------

[a]$\dfrac{\Delta h}{h} = \dfrac{\text{(1-void-tube critical height)} - \text{(zero-void-tube critical height)}}{\text{Zero-void-tube critical height}}$.

found in reference 5 for most of the fuel solution concentrations reported in table V.

Critical height data. - The data on critical heights as a function of fuel solution concentration are plotted on eight different graphs in figures 3 and 4. Figure 3 presents the criticality data for the 9.652-centimeter-grid bare (fig. 3(a)), 10.922-centimeter-grid bare (fig. 3(b)), 9.652-centimeter-grid reflected (fig. 3(c)), and 10.922-centimeter-grid reflected (fig. 3(d)) configurations. Although these four figures appear very similar, the actual data points are different of course. The trend of the data shows that the effect on critical height caused by changing the hexagonal void arrays is similar for all configurations studied. Figure 4 presents the same criticality data sorted according to 37-, 31-, 19-, and 7-void-tube arrays. This presentation affords a good comparison of the differences caused by both the grid pitch and the effect of a radial water reflector.

The grid pitch effect is of particular interest. For the 37-void-tube cases (fig. 4), the smaller 9.652-centimeter pitch results in higher critical height for all fuel solutions. For the 7-tube cases (fig. 4(d)), the larger 10.922-centimeter pitch gives slightly higher critical heights for all fuel solutions. For the 31- and 19-tube cases, no general statement of this kind can be made. For the 31-tube reflected cases (fig. 4(b)) and the 19-tube bare cases (fig. 4(c)), a crossover occurs in which one grid pitch gives a higher critical height with the more concentrated solution and the other grid pitch gives a higher critical height with the less concentrated solution. This behavior can be explained qualitatively by interpretation of the thermal flux distributions and reactivity importance data presented subsequently.

The range of fuel solution concentrations covered depends on the configuration in-

TABLE IV. - CRITICALITY DATA AT 20° C FOR ZERO-VOID-TUBE REACTOR

Atom ratio, H/X	Critical height,[a] cm	
	Bare	Reflected
151.6	14.03	-----
151.8	14.09	13.91
324	15.94	15.73
438	17.31	17.06
482	18.05	17.79
484	18.11	17.84
493	18.22	17.94
636	20.20	19.85
650	20.53	20.18
718	21.60	-----
789	23.09	22.62
870	24.70	24.15
908	25.54	24.94
993	27.64	-----
1001	27.86	27.10
1082	30.24	29.33
1217	35.31	-----
1225	35.23	33.87
1247	36.37	34.90
1394	46.14	43.38
1490	52.53	48.71
1500	55.73	51.24
1600	71.84	-----
1650	-----	71.72
1659	-----	74.36

[a]Lower spacer grid was present in core (adds about 0.25 cm to clean core height).

TABLE V. - ATOM CONCENTRATIONS FOR URANYL FLUORIDE SOLUTIONS

Atom ratio, H/X	Solution density (at 20° C), g/cu cm	Atom concentration of uranium 235, N/cu cm	Uranium 235, g/liter
151.6	1.20884	4.2998×10^{20}	167.74
151.8	1.20855	4.2938	167.51
324	1.09844	2.0405	79.60
438	1.07260	1.5135	59.04
476	1.06662	1.3916	54.29
482	1.06580	1.3749	53.64
484	1.06558	1.3704	53.46
493	1.06434	1.3452	52.47
511	1.06201	1.2977	50.63
636	1.04958	1.0445	40.74
650	1.04844	1.0213	39.84
718	1.04374	.9256	36.11
789	1.03966	.8425	32.87
870	1.03581	.7642	29.82
908	1.03424	.7322	28.57
993	1.03118	.6700	26.14
1001	1.03090	.6643	25.92
1082	1.02848	.6151	23.99
1217	1.02514	.5472	21.34
1225	1.02495	.5433	21.19
1247	1.02449	.5339	20.83
1394	1.02172	.4776	18.64
1490	1.02021	.4469	17.44
1500	1.02007	.4440	17.33
1571	1.01909	.4241	16.54
1600	1.01871	.4164	16.25
1650	1.01809	.4038	15.75
1659	1.01798	.4016	15.66

Figure 3. - Criticality data sorted by pitch and reflector configuration.
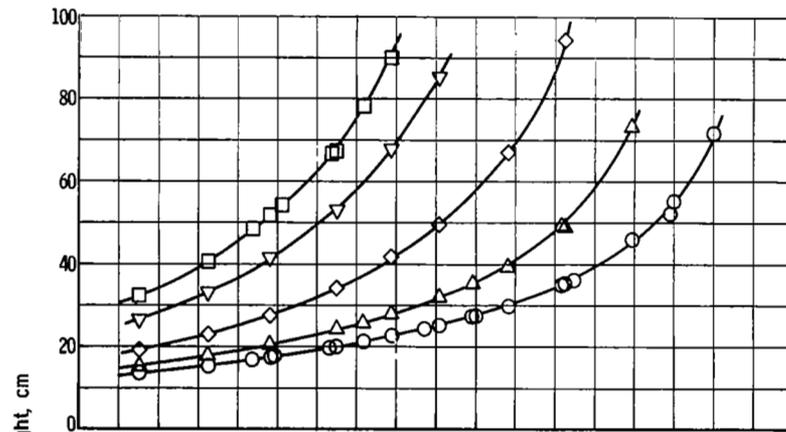
(a) Bare core with 9. 652-centimeter pitch.
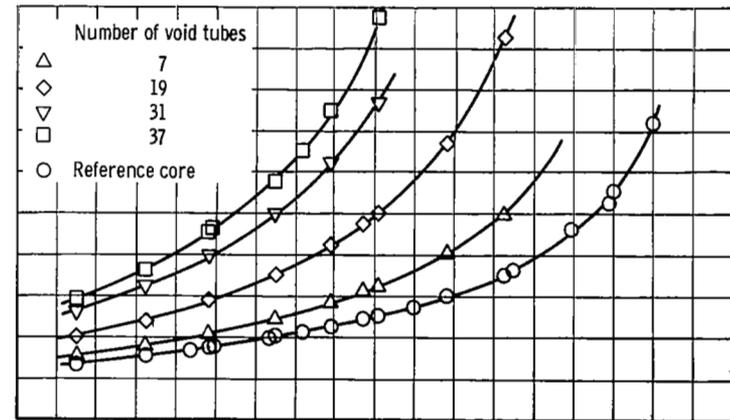
(b) Bare core with 10. 922-centimeter pitch.
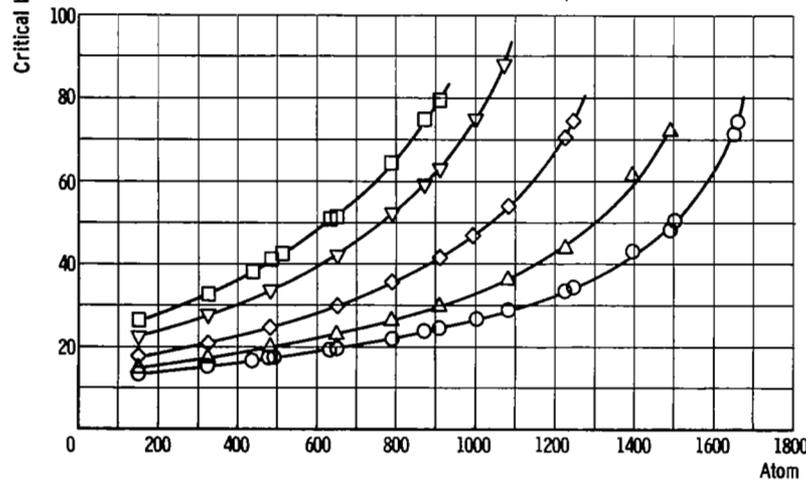
(c) Reflected core with 9. 652-centimeter pitch.
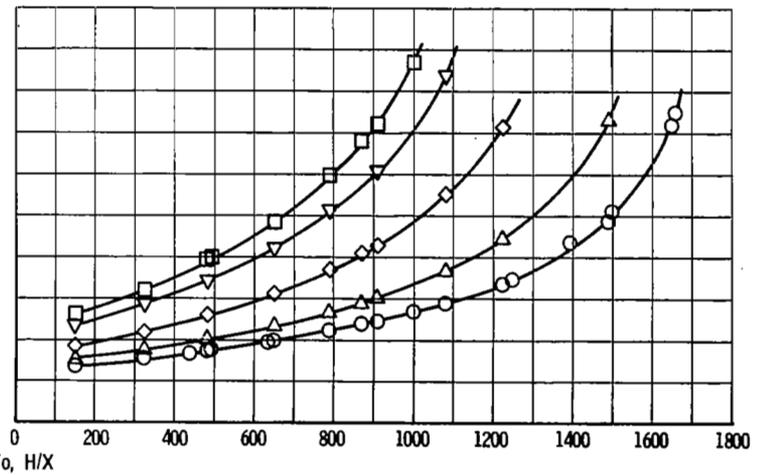
(d) Reflected core with 10. 922-centimeter pitch.
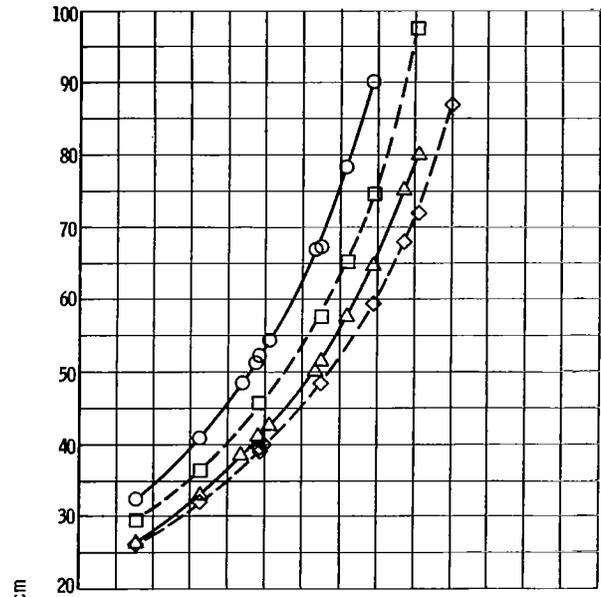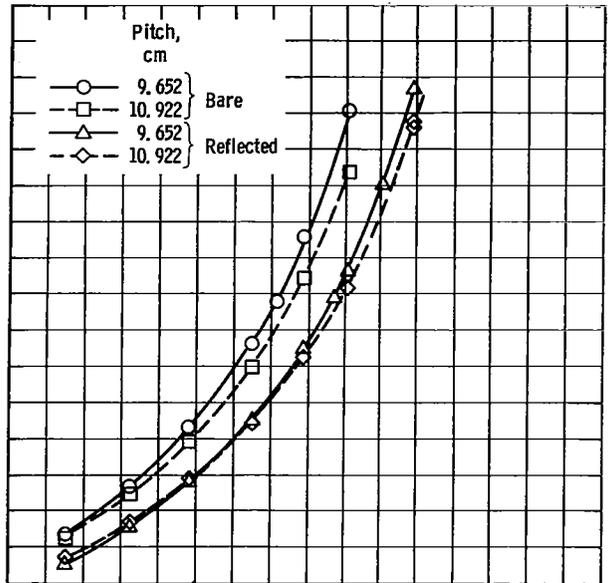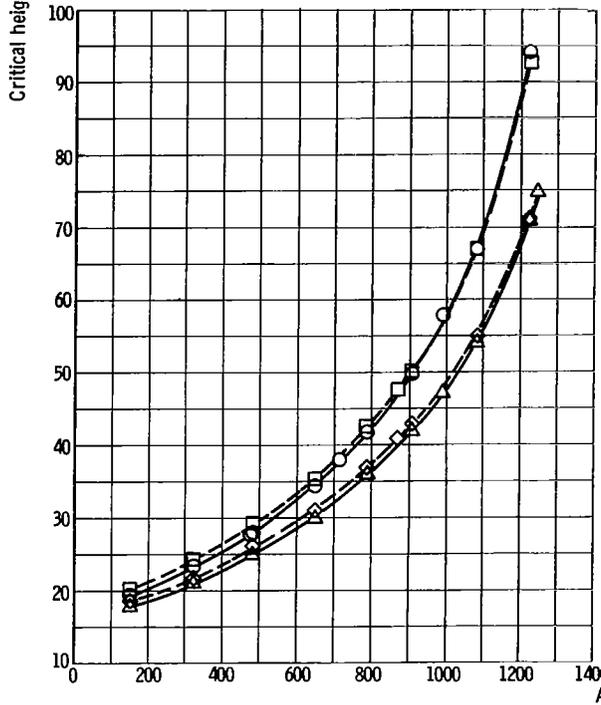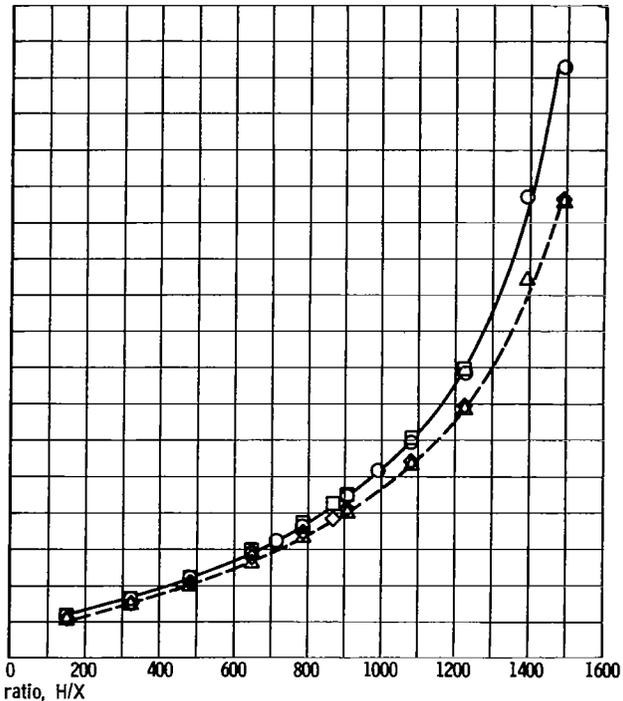
Figure 4. – Criticality data sorted according to void tube configuration.
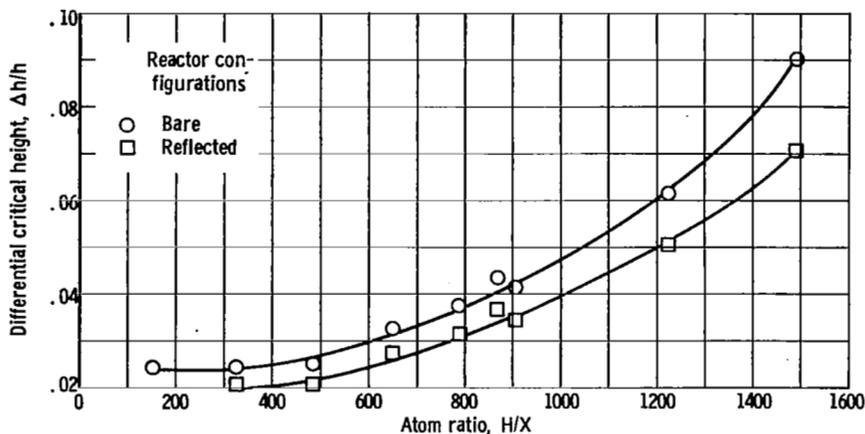
Figure 5. - Differential worth of single void tube at center of core.

volved. The experimental study began with measurements at $H/X = \sim150$ for the most concentrated case. The measurements were continued as the fuel was diluted until it became physically impossible to achieve criticality with the given array. This occurred when the critical height exceeded the 90- to 95-centimeter-height limit of the ZPR-II tank. The dilute concentration limit varied from $H/X = \sim800$ for the 9.652-centimeter-pitch, 37-void tube bare case to $H/X = \sim1650$ for the reference cases.

Figure 5 presents a plot of the differential critical height as a function of concentration for a single, centrally located void tube compared with the reference case with no void tubes. The differential referred to is the fractional change in critical height as compared with the reference case critical height, or $\Delta h/h$. The scatter of the data is attributed primarily to the fact that the single void tube data were obtained by using either 9.652- or 10.922-centimeter tube spacing grid at random. Small differences in grid reactivity worth have been measured. These are accounted for, but because of the small effect being measured some scatter is introduced.

Figure 6 is a plot of critical heights as a function of the number of void tubes in the arrays studied for a fixed atom ratio at $H/X = 908$. The curves are used only as a guide; these data cannot be regarded as a continuous function of the variables plotted. This plot is representative of the data that are obtained at any particular fuel concentration. The experimental procedure outlined earlier produces a series of similar plots at various fuel concentrations that have been combined to produce the data compiled in tables II to IV and figures 3 and 4.

Critical mass (uranium 235) data. - Data on critical mass cover the same number of cases as the critical height data and would also require a set of eight graphs to present it in a comparable manner. Since the graphs do have a marked similarity and show similar trends, only two representative configurations are presented. Figure 7 presents critical masses for the 9.652-centimeter-grid bare systems. Figure 8 presents the critical masses for the 37-void-tube configurations. These figures present critical mass data for

13
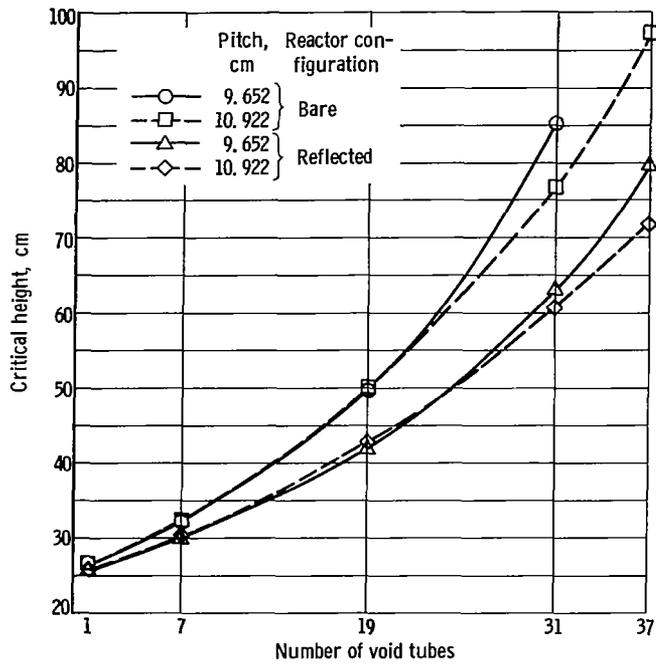
Figure 6. – Criticality data for atom ratio H/X = 908.
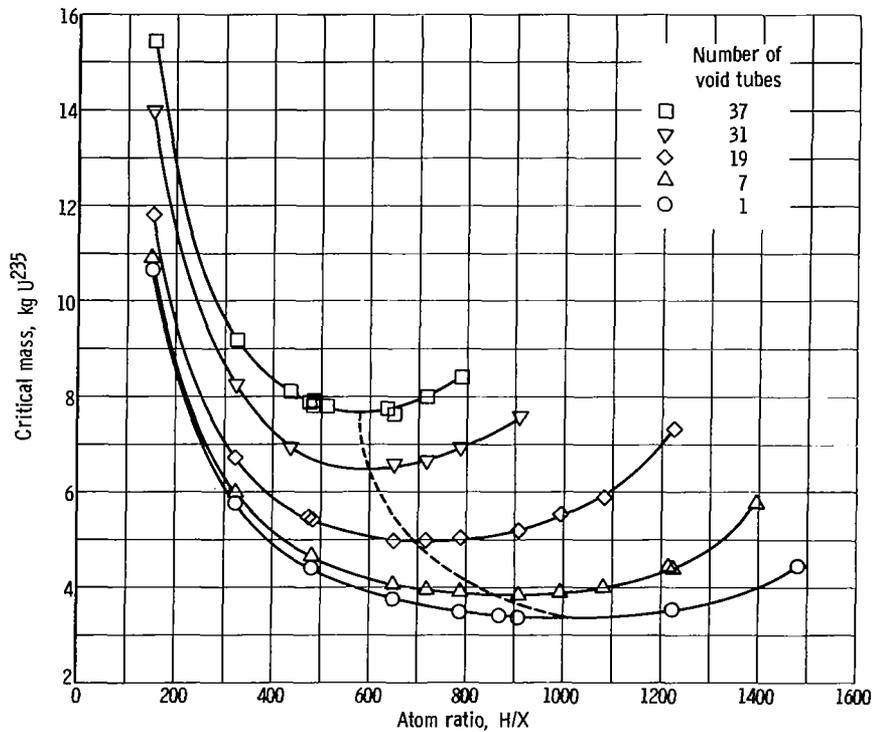
Figure 7. – Critical mass plotted against atom ratio H/X for 9.652-centimeter-pitch bare core.
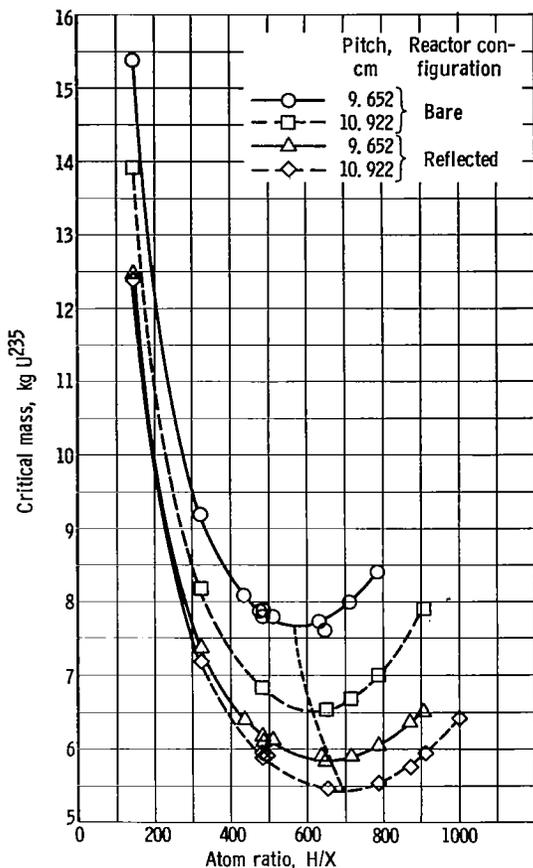
Figure 8. - Critical mass plotted against atom ratio H/X for 37-void-tube core.

the comparable cases for which critical heights are presented in figures 3(a) and 4(a), respectively.

The minimum critical mass shifts in the direction of higher H/X (lower fuel concentration) as the critical mass requirement of the particular family of critical systems is reduced (see both figs. 7 and 8). The curves in figure 7 demonstrate the change in critical mass typical for the family of arrays studied. The curves in figure 8 demonstrate the effect on critical mass due to both the change in radial reflector and the change in void spacing.

A reduction in critical mass for the reflected systems as compared with the bare systems is obtained because of positive reactivity effects of the water reflectors. However, the changes in critical mass associated with various void spacings involve the relative reactivity roles of the voided and unvoided regions of the core. In the top two curves of figure 8 (and fig. 4(a)), the total void fractions were identical, but the distribution of voids differed and consequently the critical heights were quite different.

The core configurations can be divided into two approximately concentric regions, a central region with void and fuel solution interspersed surrounded by a region of fuel solution only. The change in void spacing results in a composition change in the central voided region and a geometric change in both regions. The reactivity effects caused by these changes are always opposite and competitive. For example, closer spacing of the voids decreases the size of the central voided region and reduces the fuel to void ratio. Both of these changes in the central voided region cause a reduction in reactivity. On the other hand, the surrounding unvoided fuel solution region is increased in size, which causes an increase in reactivity. The net effect on reactivity is dependent on the particular configuration involved. Specific examples are explained qualitatively in terms of measured thermal flux distributions in the sections that follow.

15
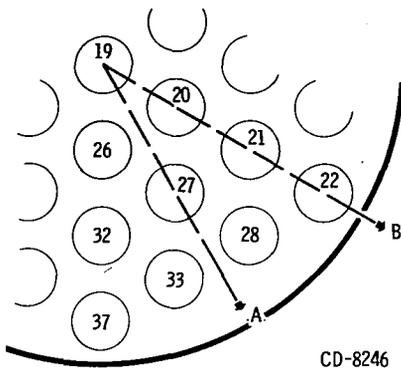
Figure 9. - Radial flux traverses.

CD-8246

Thermal Flux Distributions

The radial thermal flux measurements and individual void worth measurements are used to interpret the net reactivity effects observed in these studies. Of special interest is the understanding of the effects observed in the partly voided 19- and 31-tube cores.

The thermal flux was determined by two radial traverses A and B as indicated in figure 9. A one-sixth symmetrical section of the hexagonal array of void tubes is shown with tube 19 as the central tube. The flux measurements were made in the radial plane that goes through the center of the reactor.

Thermal fluxes were measured for the 37- and 19-void-tube cases with both grid spacings and at two fuel concentrations corresponding to a short and a tall reactor. A single 7-void-tube case was also traversed. All configurations traversed were bare reactors. The interface between the central void region and the outer unvoided region is not cylindrical for most of the arrays since the grid arrangements are hexagonal.

In the section Critical height data, it was observed that the 37-void-tube arrays have a higher critical height (greater critical mass) when the smaller pitch is used. An examination of the thermal flux distributions (fig. 10) shows that the thermal flux is higher in the central voided region than in the unvoided region in all 37-void cases studied. Traverse A of the 9.652-centimeter-pitch short reactor (H/X = 150) shows a peak near the interface of the central voided and outer unvoided regions; however, this peak is confined to a rather small core volume. The central voided region is indicated to be the more important core region for the 37-tube arrays. Consequently, a change to the central region is worth more in reactivity than the same change in the outer region.

The higher critical heights measured for the smaller pitch cases can be explained qualitatively by the fact that there is a loss in reactivity for two similar reasons: (1) the transfer of fuel from the center region of higher importance and higher flux to an outer location of lower importance and lower flux, and (2) the transfer of voided volume from the outer, lower importance region to a central location of higher importance and higher flux. This replacement of fuel with void in the more important location and void with fuel in the less important location gives the same fuel solution to void volume ratio in the core but requires more fuel to achieve criticality.

For all bare 7-void-tube cases, the thermal neutron flux peaks in an outer fueled region. The flux pattern shown in figure 11 is representative of the 7-void arrays. The central voided region has a low thermal neutron flux in this array. There is evidence from experimental and analytical data to indicate that a major factor in causing this effect is the axial leakage of the thermal neutrons in the voids.

Figure 10. - Experimental thermal neutron flux distributions for bare 37-void-tube arrays.
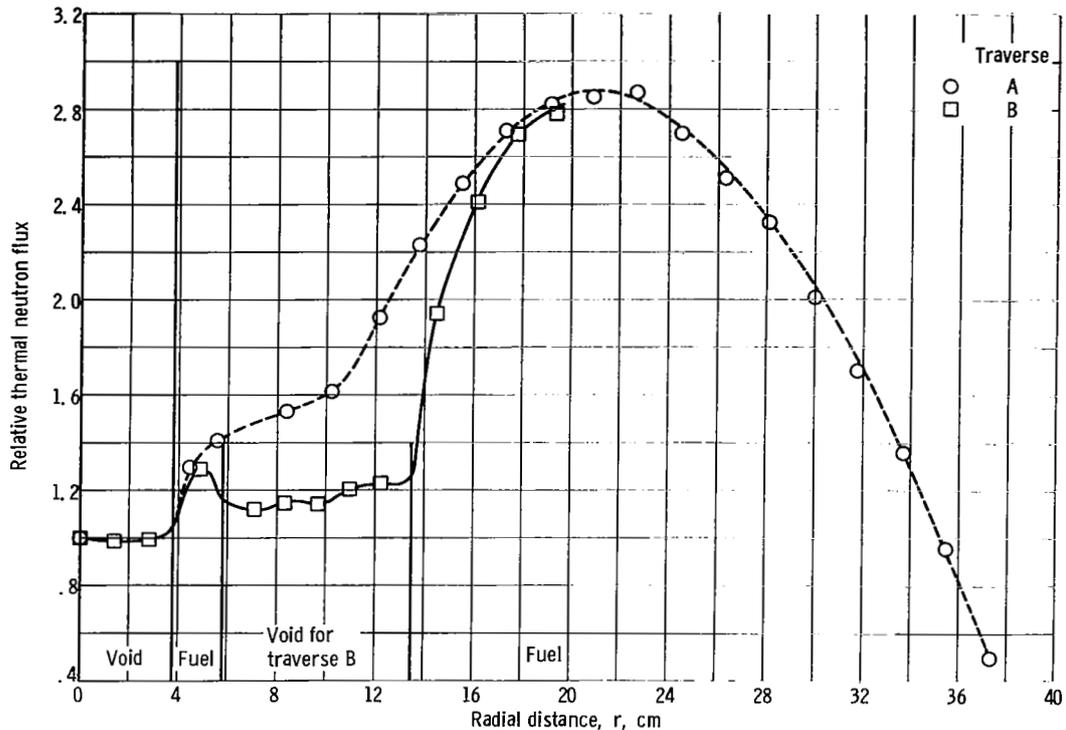
Figure 11. - Experimental thermal neutron flux distributions for bare 7-void-tube array with 9. 652-centimeter pitch and atom ratio H/X = 150.

For the 7-tube cases studied, the larger void pitch gave the slightly higher critical heights. The explanation is the same as before: the movement of voids to a region of higher thermal flux, and the movement of fuel to a region of lower thermal flux cause a net loss in reactivity. The effect is quantitatively much smaller because of the smaller voided core region involved.

For the 19-void bare reactors (fig. 4(c), p. 12) and the 31-void reflected cases (fig. 4(b)), the data indicate a crossover in critical heights between grid spacings for the range of fuel concentrations studied. The 19-tube bare configurations were chosen for a more detailed study and flux maps were made for four cases. These included both grid spacings at H/X = 150 and 1175 as plotted in figure 12. The flux distributions for the tall reactors with H/X = 1175 have a shape approaching the 37-void-tube cases, and the closer grid spacing requires a greater critical height accordingly. The short reactors with H/X = 150 have flux distributions with a shape more like the 7-tube case and the larger grid spacing requires a greater critical height. It is expected that the same explanation applies to the 31-tube reflected configurations.

Figure 12. – Experimental thermal neutron flux distributions for bare 19-void-tube arrows.
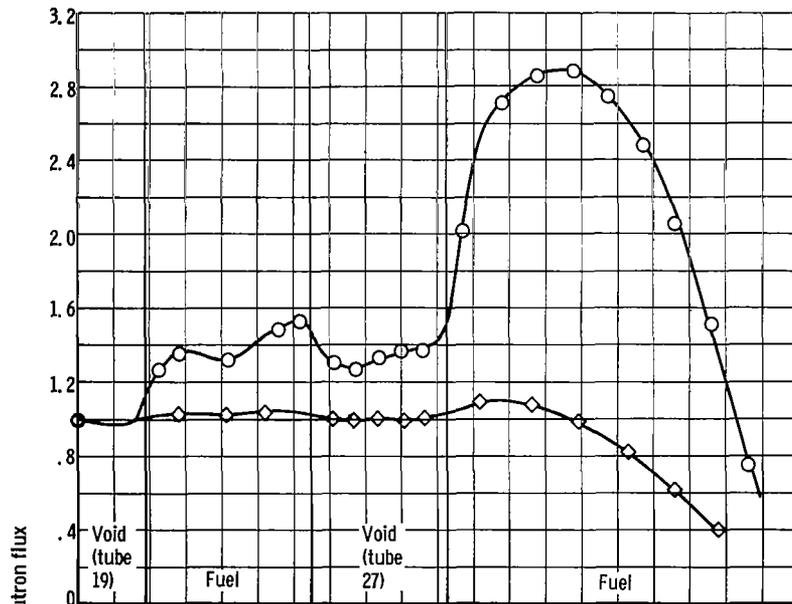
(a) Traverse A; pitch, 9.652 centimeters.
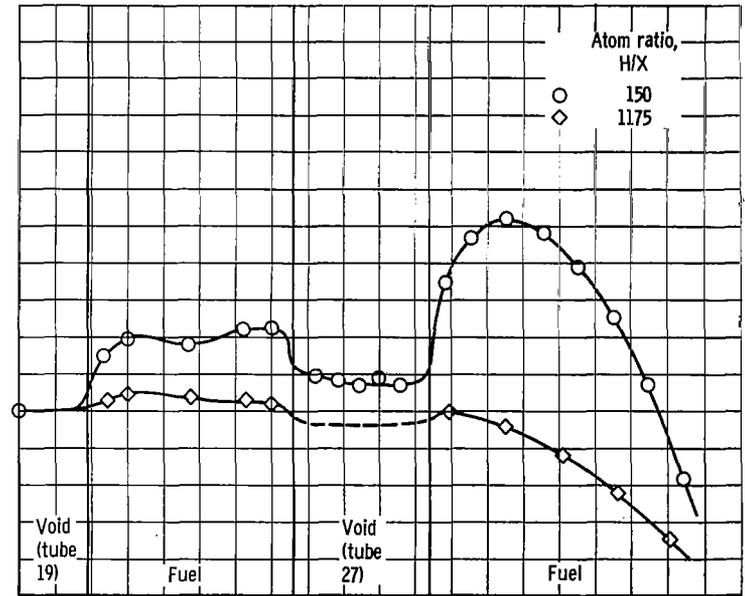
(b) Traverse A; pitch, 10.922 centimeters.
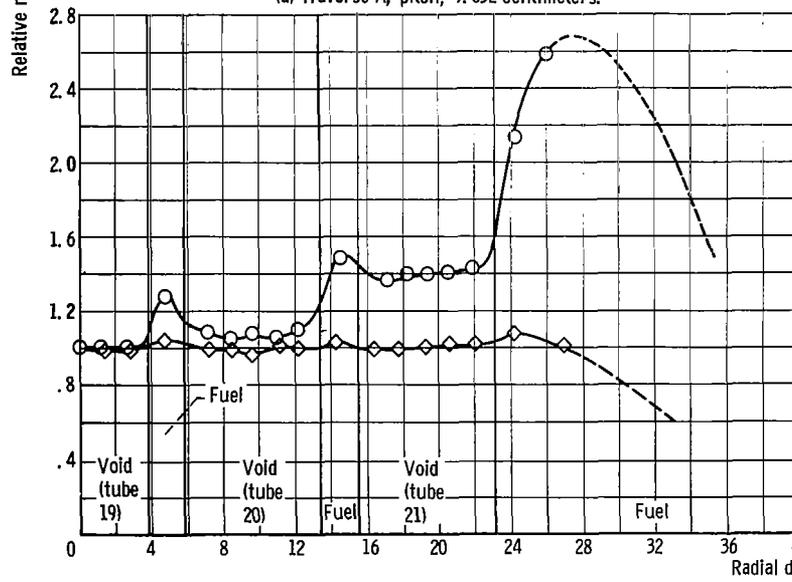
(c) Traverse B; pitch, 9.652 centimeters.
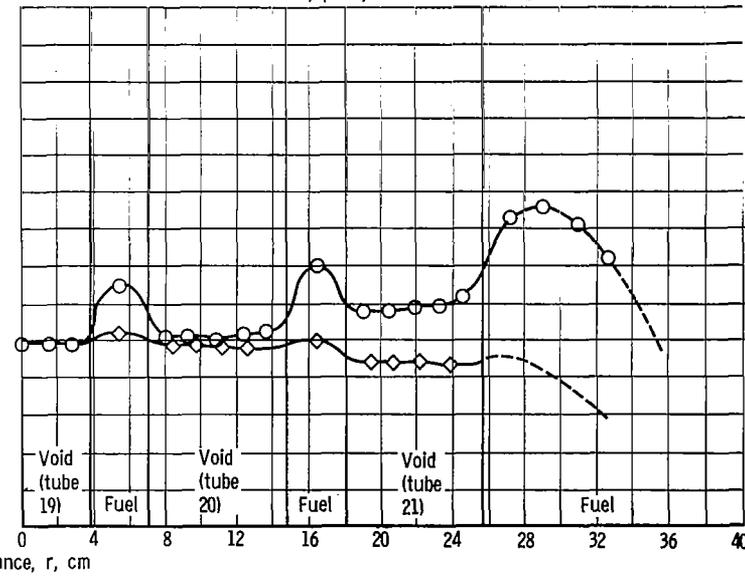
(d) Traverse B; pitch, 10.922 centimeters.

Reactivity Measurements of Individual Void Tubes

The smallest void used in this investigation was a single 7.6-centimeter-diameter tube. This constitutes a fuel solution displacement and is a larger perturbation than ordinarily used for measuring the spatial variation of reactivity worth. The worth of a single void tube was measured as a function of radial position in a clean homogeneous core and compared with the variation expected for an absorber in a bare homogeneous thermal reactor. If a one energy group model is assumed for the neutrons in a bare homogeneous core, the spatial importance function of a local perturbation in absorptivity should vary approximately as the square of the neutron flux φ. The individual void tubes used constitute a fuel solution displacement and should follow this φ^2 law. The neutron flux distribution in the radial dimension in a critical homogeneous reactor of cylindrical geometry is

$$\varphi(r) = AJ_0\left(\frac{2.405\,r}{R}\right)$$

where

r   radial distance from axis

A   arbitrary constant

J_0 Bessel function of the first kind, zero order

R   extrapolated radial boundary of reactor, $R_0 + \ell$, 42 cm

R_0 core radius, 38.1 cm

ℓ extrapolation length, 3.9 cm

If the value of $A = 1$ is chosen so that $\varphi(0) = 1$ and $\varphi(42) = 0$, $\varphi^2(r)$ is plotted in figure 13 as a function of the radial distance. The experimental reactivity worths for individual void tubes as measured by the change in critical height associated with locating the void tube in various radial positions of a core with a fuel concentration H/X = ~1175 are also shown in figure 13. The experimental reactivity values have been normalized to unity at the center. The experimental and $\varphi^2(r)$ results show reasonable agreement for a range of values of extrapolation length.

The worth of an individual void tube in the hexagonal void-tube arrays was measured by removing the single void tube at various radial positions. Removal of the void tube resulted in a reduced critical height Δh that was taken as a measure of the relative reactivity of the tube. The results for the hexagonal arrays do not appear to follow exactly
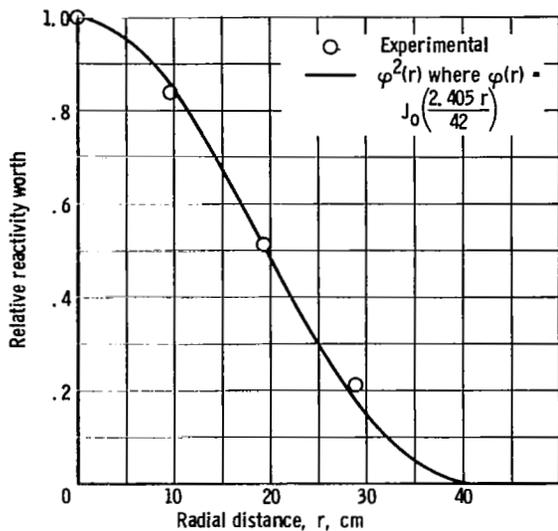
Figure 13. - Relative reactivity worth of single void tube as function of radial position in homogeneous cylindrical reactor with atom ratio H/X = 1175.

the same law as those obtained in the measurements made by inserting a void tube, but they do give an indication of the importance of radial position. In all cases, the radial traverse B (fig. 9, p. 16) was used in the importance measurements. For example, tube 19 was removed, a measurement was made of the new critical height, and then the tube was replaced. This procedure was repeated for each tube in traverse B.

The results of the measurements made with representative 37-void-tube arrays are shown in figure 14(a). The reactivities are estimated by using the experimentally determined reactivity worths for small increments in fuel height near critical. These are extended to larger height increments by assuming a linear relation. Calculations indicate

(a) 37-Void-tube array.

(b) 19-Void-tube array.
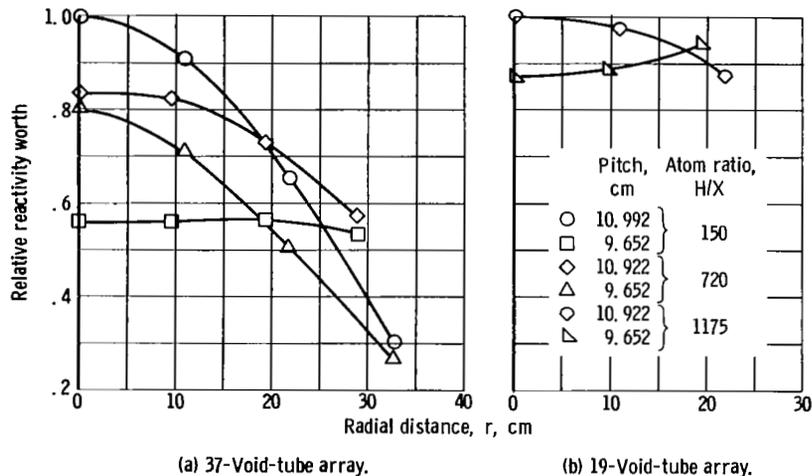
Figure 14. - Relative reactivity worth of single void tube removal as function of radial position for bare reactors.

that no more than 10 percent error is involved for the changes in critical height reported herein. The 37-tube cases reported are the same ones for which flux maps were presented earlier. The flux distributions and individual void-tube reactivity measurements are in good agreement. If each radial traverse of both the reactivity data and the flux data were normalized to unity at the center of the core, the agreement would be more readily seen.

The results of importance measurements made with 19-void-tube cases are shown in figure 14(b). Only the tall cores with atom rations H/X of 1175 are compared. For the larger pitch spacing, the central void region is more important; for the smaller pitch spacing, the outer fueled region is more important.

Precision of Experimental Data

The critical heights in all cases are corrected to a temperature of 20° C. The estimated maximum uncertainty in the height is ±0.075 centimeter. Most of this uncertainty is associated with tank dimensional tolerances and calibrations of the height-measuring device. The reproducibility of the height measurement itself is ±0.0025 centimeter. The reproducibility of any given experiment in terms of critical height is estimated to be ±0.050 centimeter for all experiments where spacer grids were involved.

The fuel solution density can be measured to ±0.00025 gram per milliliter (95 percent confidence limit). This gives a considerable variation with concentration in the precision to which H/X is known. For example, the error limits at several fuel atom ratios H/X are 150±0.2, 500±2, 1000±8, and 1600±20 for a percentage uncertainty varying from about 0.13 to 1.3 percent over the range of fuel concentrations used.

For most of the configurations, therefore, the precision of the critical mass reported is determined by the precision of the fuel concentration determination. The exceptions are the short reactors where the uncertainty in height measurement is larger than the uncertainty in the fuel density measurements. The estimated error in critical mass varies from about 0.5 percent at $H/X = 150$ to 1.5 percent at $H/X = 1600$.

The uncertainties in foil data measurements are attributed primarily to counting statistics and foil positioning. The error associated with counting statistics and counting equipment had a standard deviation varying from about 0.2 to 1.7 percent depending on the foil activity. The greatest potential error is associated with the foil positioning. At some locations (flat flux regions) position is not critical; at others it can lead to large errors. For high flux regions where the flux does not vary rapidly, the total error

(95 percent confidence limit) is estimated from repeated measurements to be as small as ±3 percent; while for the worst locations, it is estimated to be ±10 percent.

Lewis Research Center,
    National Aeronautics and Space Administration,
        Cleveland, Ohio, May 10, 1966,
            122-28-03-02-22.

REFERENCES

1. Thornton, Gunnar; and Blumberg, Ben: ANP HTREs Fulfill Test Goals. Nucleonics, vol. 19, no. 1, Jan. 1961, pp. 45-51.

2. Anon.: Army Gas-Cooled Reactor Systems Program. The ML-1 Design Report. Rept. No. IDO-28550, Aerojet General Nucleonics, May 16, 1960.

3. Rom, Frank E.; Lietzke, Armin F.; and Johnson, Paul G.: Nuclear Rockets of Unmanned Missions. Nucleonics, vol. 20, no. 11, Nov. 1962, pp. 53-57.

4. Bogart, Donald; and Lantz, Edward: Nuclear Physics of Solid-Core Gas-Cooled Rocket Propulsion Reactors. Nuclear Rocket Propulsion, NASA SP-20, 1962, pp. 17-25.

5. Fox, Thomas A.; Mueller, Robert A.; Ford, C. Hubbard; and Alger, Donald L.: Critical Mass Studies With NASA Zero Power Reactor II. Pt. I - Clean Homogeneous Configurations. NASA TN D-3097, 1965.

NASA SCIENTIFIC AND TECHNICAL PUBLICATIONS

TECHNICAL REPORTS: Scientific and technical information considered important, complete, and a lasting contribution to existing knowledge.

TECHNICAL NOTES: Information less broad in scope but nevertheless of importance as a contribution to existing knowledge.

TECHNICAL MEMORANDUMS: Information receiving limited distribution because of preliminary data, security classification, or other reasons.

CONTRACTOR REPORTS: Technical information generated in connection with a NASA contract or grant and released under NASA auspices.

TECHNICAL TRANSLATIONS: Information published in a foreign language considered to merit NASA distribution in English.

TECHNICAL REPRINTS: Information derived from NASA activities and initially published in the form of journal articles.

SPECIAL PUBLICATIONS: Information derived from or of value to NASA activities but not necessarily reporting the results of individual NASA-programmed scientific efforts. Publications include conference proceedings, monographs, data compilations, handbooks, sourcebooks, and special bibliographies.

Details on the availability of these publications may be obtained from:

SCIENTIFIC AND TECHNICAL INFORMATION DIVISION

NATIONAL AERONAUTICS AND SPACE ADMINISTRATION

Washington, D.C. 20546