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SUBJECT: OAK RIDGE NATIONAL LABORATORY FAST BURST REACTOR: CRITICAL EXPERIMENTS AND CALCULATIONS

TO: Distribution

FROM: W. E. Kinney and J. T. Mihalczo

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I. INTRODUCTION

Critical masses and void coefficients of reactivity of three unmoderated, unreflected assemblies, and the critical masses of ten unmoderated, reflected assemblies have been determined at the Oak Ridge National Laboratory Critical Experiment Facility to support the design of an unmoderated and unreflected research reactor capable of producing intense bursts of fast neutrons. Multigroup and one-group transport theory calculations, as well as perturbation theory calculations, have been performed to predict the results of the experiments with the unreflected assemblies.

The Fast Burst Reactor is first briefly described, followed by a discussion of the experiments. The calculational methods are considered next and finally the results of both experiment and calculation are presented and compared.

II. DESCRIPTION OF THE REACTOR

The United Nuclear Corporation, under contract to the Oak Ridge National Laboratory, is completing the design of the ORNL Fast Burst Reactor.[1] The Fast Burst Reactor (FBR) is an unmoderated and unreflected cylinder of uranium-molybdenum alloy, designed to produce 50-μsec bursts of 10^{17} fissions. It will also be capable of operation at 10 kw thermal power for ten minutes with forced-air cooling, or of steady state operation at 1 kw with natural convection cooling. It is intended for use as a research tool at Oak Ridge National Laboratory for radiobiological, dosimetry, materials testing and radiochemical research.

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1. G. Breidenbach, et al., Preliminary Design of the ORNL Fast Burst Reactor, NDA 2136-1, July 30, 1960.

The principal difference between the FBR and the first fast-burst reactors, Godiva I and II,[2] lies in the use of the U-Mo alloy rather than unalloyed uranium. The greater strength and, in general, better physical properties of the alloy will allow the production of bursts a factor of five larger than those produced by Godiva II. The better dimensional stability of the alloy under thermal cycling will increase the reactor operating lifetime.

The design reactor is essentially a 20.32-cm (8-in.) -dia annulus of U-Mo alloy about 19.05 cm high with a 5.08-cm-dia movable cylindrical steel core. To the lower section of this magnetically supported steel core is attached a 8.58-cm-dia annulus of alloy, which serves as a safety device. The reactor also will be fitted with movable rods, made of the alloy, to adjust to delayed critical and to provide excess reactivity for pulse generation. The steel core reduces the peak-to-average temperature ratio in the alloy.

The sequence of steps in the production of a burst of radiation is similar to that of Godiva II.[2]

III. CRITICAL EXPERIMENTS

The principal purposes of the experiments reported here were to establish over-all dimensions of the reactor sufficient to compensate for reactivity losses in voids and other structural features, and to determine the fission-rate distribution within the U-Mo. It was also possible to measure the critical dimensions of both solid and annular cylinders of the alloy and of an alloy cylinder with a steel core, in order to evaluate methods of reactor analysis.

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2. T. F. Wimett, et al., Nuclear Sci. and Eng. 8, 691 (1960).

Four critical assemblies were studied. The first was a solid 20.32-cm-dia cylinder of U-Mo alloy. The second was a 20.32-cm-dia cylinder of U-Mo alloy with a 5.08-cm-dia axial hole. The third resembled the second except that the hole was filled with stainless steel. The fourth was a modification of the second, but with the lower 12.70 cm of the hole enlarged to 8.89 cm and with various conditions of reflection.

The material used in the experiments was a U-Mo alloy containing 10 wt% molybdenum and 90 wt% uranium (93.17 wt% U^{235}). The alloy had a density of 17.08 g/cm^3. The impurity content of the U-Mo alloy is shown in Table A-1 of Appendix A and its physical properties are shown in Table A-2 of Appendix A.

The stainless steel used in the critical experiments was type 304, containing 18% nickel and 8% chromium.

The material used as reflector was a methacrylate plastic (Plexiglas) containing 5.8×10^{22} atoms of hydrogen and 3.6×10^{22} atoms of carbon. Its density was 1.2 g/cm^3.

The assembly machine used in the critical experiments is a vertical, hydraulically operated piston which is remotely controlled. A small steel table is fastened to the piston rod by an electromagnet as shown in Fig. 1.

The hydraulic cylinder is of the double-action type: pressure is applied to the top of the piston to lower the table and to the bottom of the piston to raise the table. Two independent gas-pressure systems, at 100 psi and 350 psi respectively, are used to move the hydraulic oil. The normal operating pressure of 100 psi is applied to the oil in the cylinder to raise or lower the piston. At a scram signal, a solenoid valve opens and the 350-psi-pressure gas forces the hydraulic fluid into the top of the

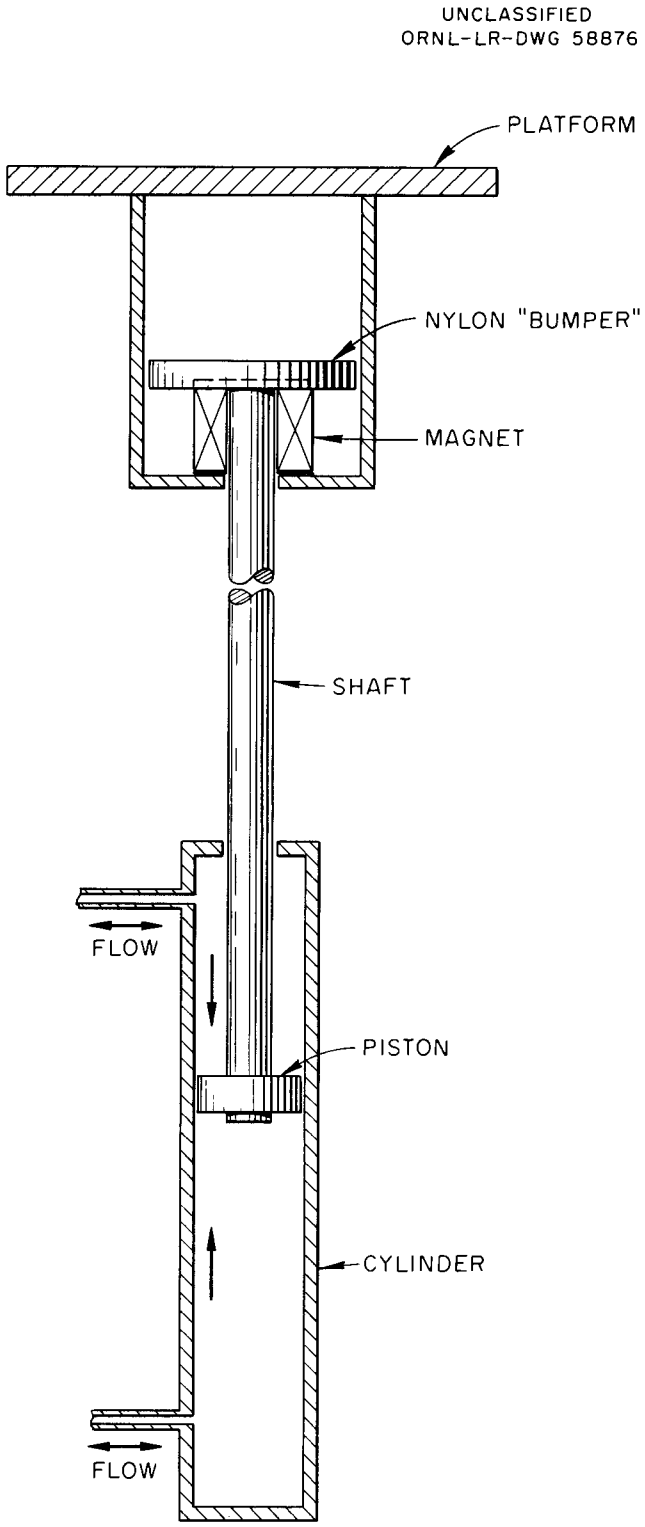

Fig. 1. Schematic of Magneto-Hydraulic Support Device

cylinder. At the same time a valve opens allowing oil to drain from the bottom of the cylinder. The valves involved in producing a scram are normally open, requiring electric power to close. In addition to the hydraulic scram, the signal interrupts power to the magnet and the steel table under gravity falls 6 in. This combined hydraulic and gravity scram provides a rapid initial separation either upon power failure or by actuating the scram. The rate of travel of the lower half of the assembly after a signal to scram is shown in Fig. 2.

On the steel table is bolted a low-mass aluminum column by which the lower part of the core is supported. The upper part of the core is supported by a 10-mil-thick stainless steel diaphragm held in tension. Figure 3 is a photograph of the equipment prior to assembly of the core.

Source-neutron multiplication measurements were performed in all initial approaches to criticality. The critical dimensions and the critical masses for the unreflected assemblies are shown in Table 1. In determining the critical heights of unreflected systems, corrections for the various reflecting materials near the assembly were made. The effect of the support structure for the lower part of the assembly was measured by placing an identical structure on the top of the assembly and again measuring the reactivity of the system. The effect of the 10-mil-thick diaphragm supporting the upper part of the assembly was evaluated by increasing the diaphragm thickness to 0.025 in. in 0.005-in. steps, measuring the reactivity after each step change. By extrapolating a curve of reactivity vs diaphragm thickness to zero diaphragm thickness, the reactivity of the assembly without the diaphragm was obtained. The return of leakage neutrons from the thick walls and floor of the room is

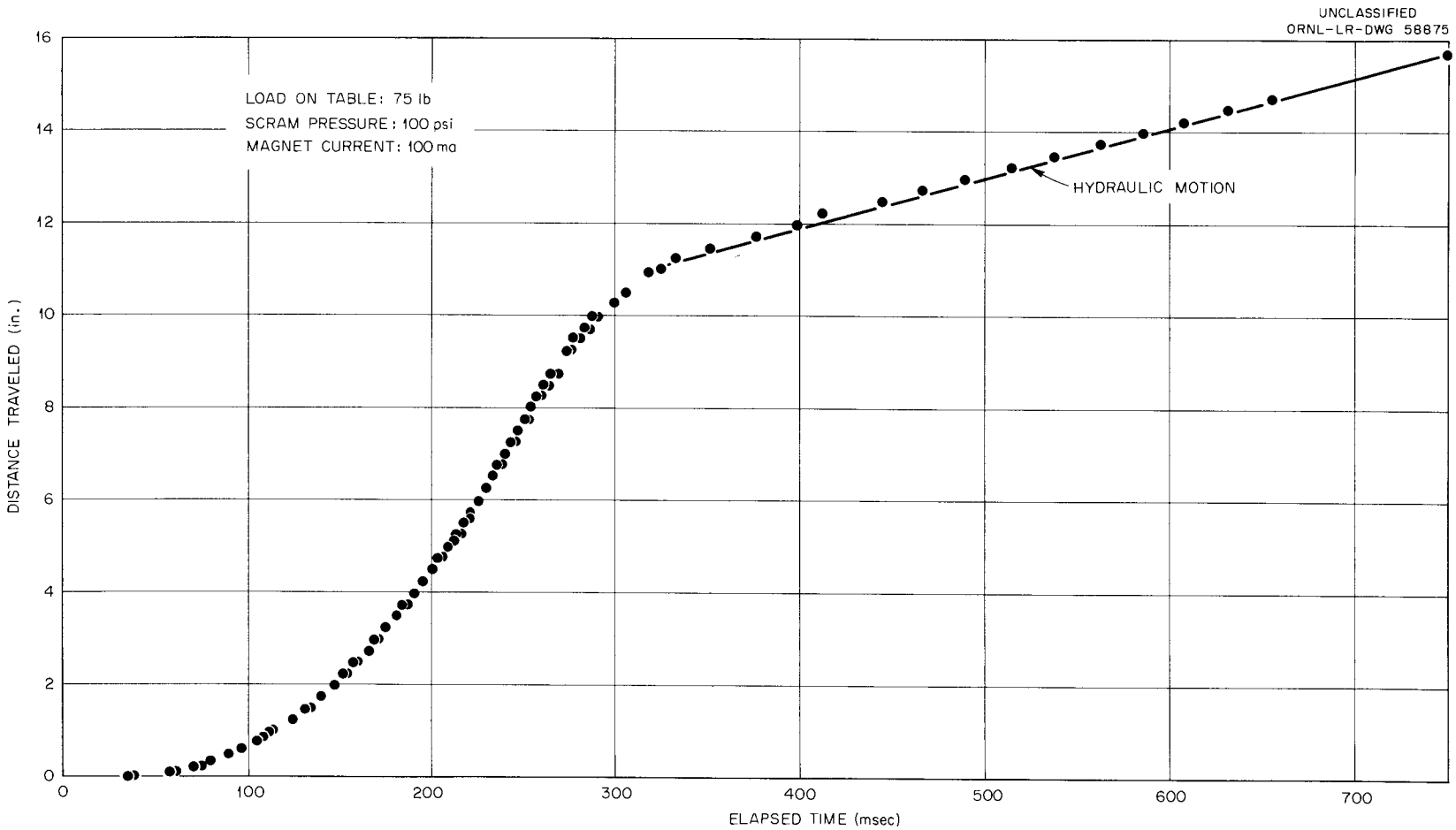

Fig. 2. Motion of the Movable Vertical Table after a Combined Hydraulic and Magnet Scram.

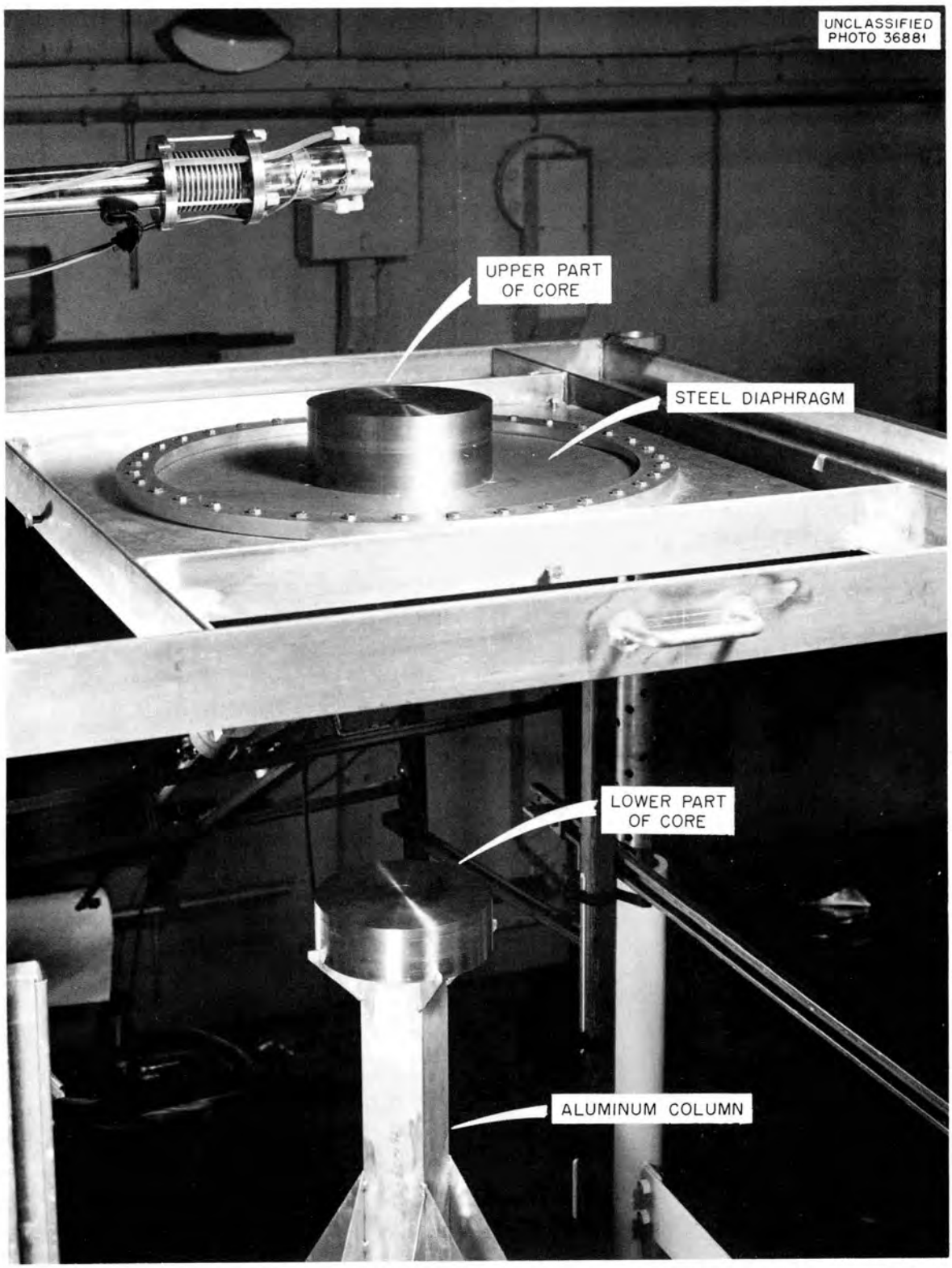

Fig. 3. Vertical Split-Table Assembly, Showing Steel Diaphragm Support for Upper Half of Core.

estimated, from measurements with Godiva I indoors and outdoors at Los Alamos Scientific Laboratory,[2] to increase the reactivity of the system by 15 cents. By measuring the reactivity change associated with an incremental change in height the critical height was corrected for the above effects. After these corrections are made, the critical heights are in error by less than 0.08 cm.

Table 1. Critical Parameters of an Unreflected, 20.32-cm-dia Uranium-Molybdenum Alloy Cylinder

Assembly	Critical Height (cm)		Critical Mass (kg of U-235)
	Experimental	Calculated	Experimental
Solid cylinder	14.78	13.86	68.6
Annular cylinder;			
5.08-cm-dia axial hole	19.74	18.22	85.9
5.08-cm-dia steel core	18.92	17.57	82.4

The radial fission density distribution at the midplane of each unreflected assembly was obtained from the fission product activity in cylindrical U-Mo pellets, 0.635-cm long and 0.927 cm in dia. Results of repeated measurements of the fission distributions indicate that the precision of these measurements is about 1%. The fission rate as a function of radial position for the unreflected systems is given in Figs. 4, 5, and 6.

The change in reactivity caused by the introduction of voids in the system was measured by comparing the reactivity of the assembly with a void to that of the assembly with the void filled with U-Mo alloy. The

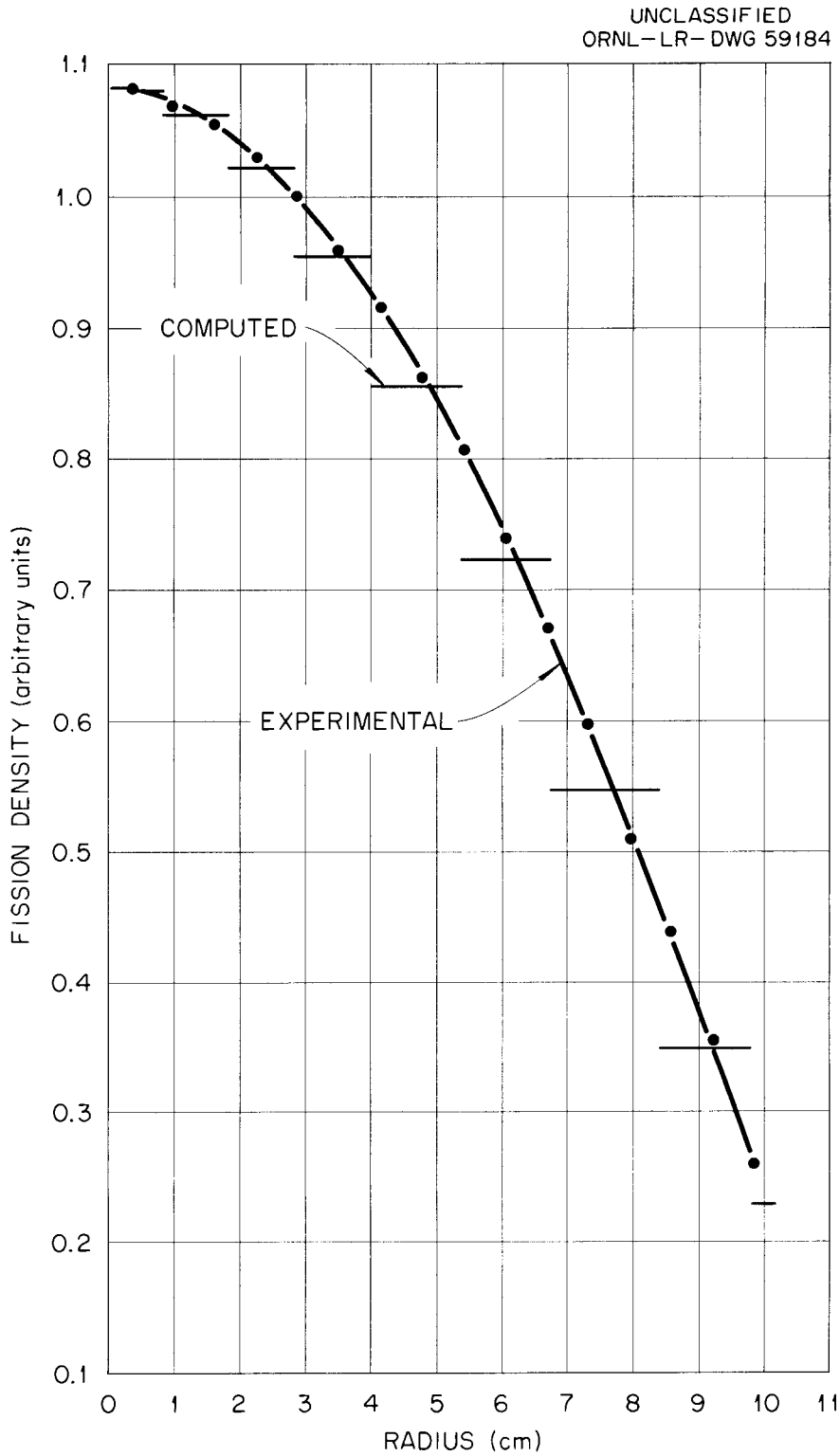

Fig. 4. Midplane Fission Density as a Function of Radius: Solid Core.

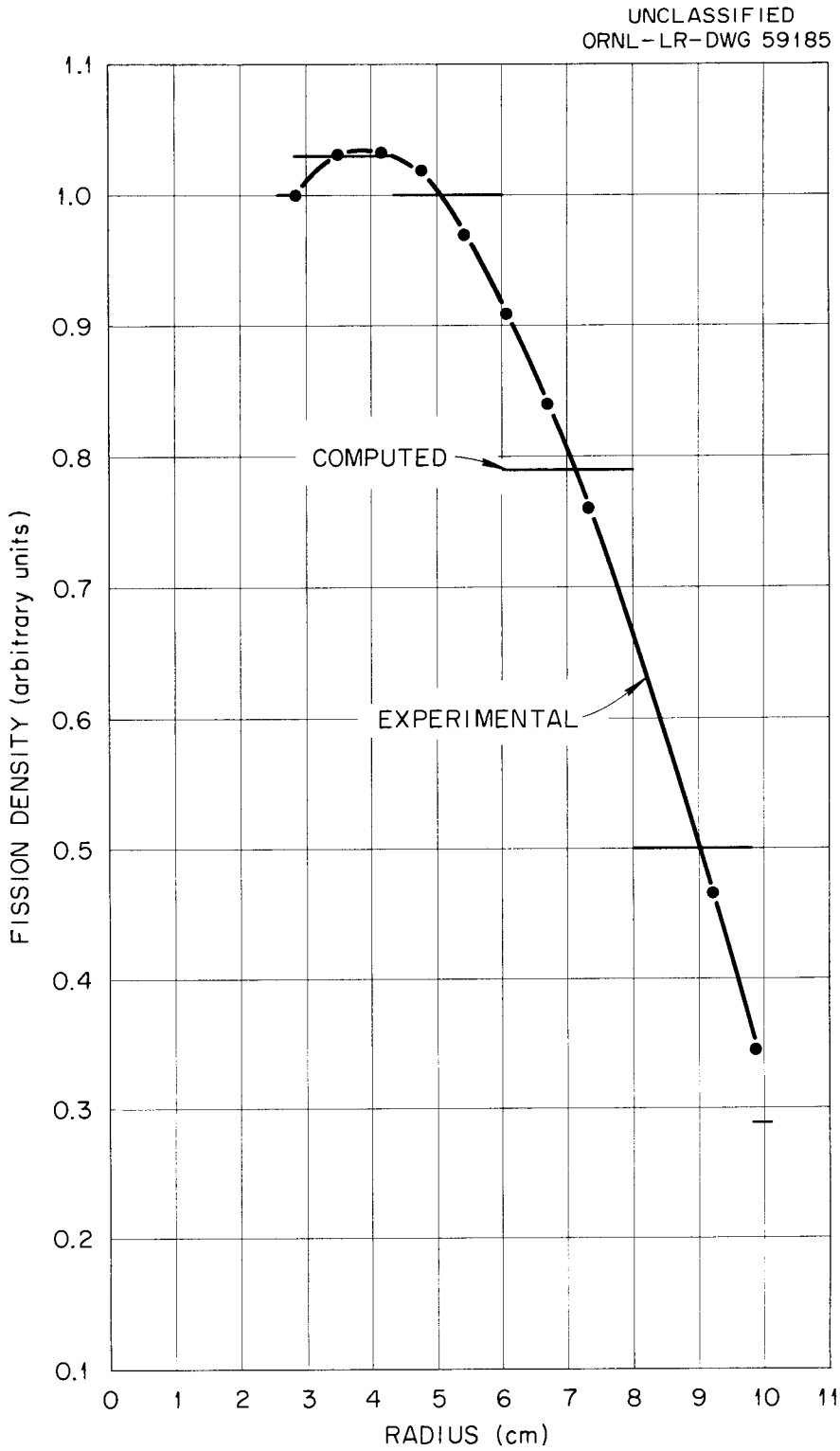

Fig. 5. Midplane Fission Density as a Function of Radius: Void Core Assembly.

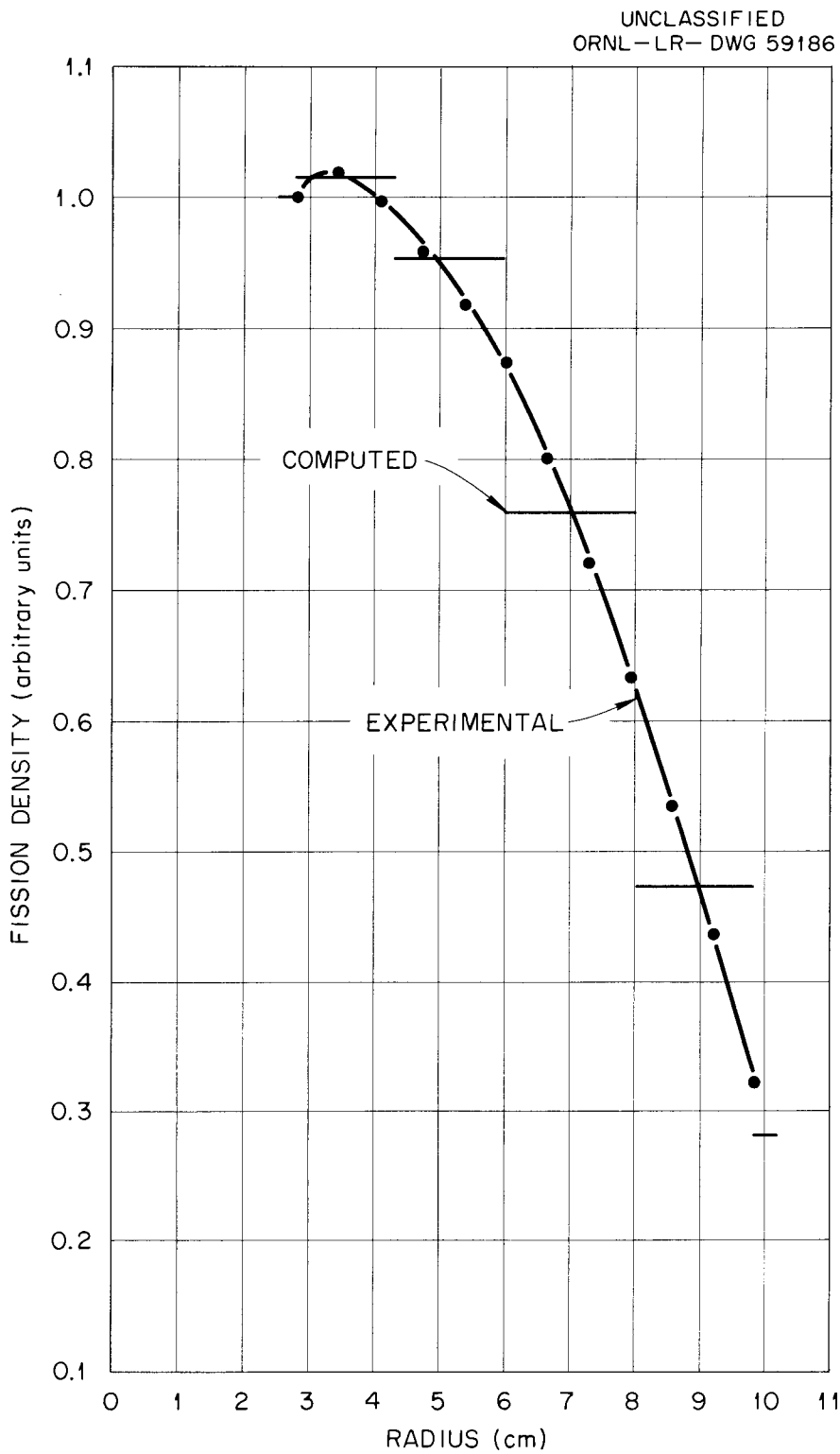

Fig. 6. Midplane Fission Density as a Function of Radius: Stainless Steel Core.

size of the cylindrical void was 0.927 cm in dia by 0.635 cm long. Void coefficients of reactivity at various positions in the unreflected systems were measured and are given in Table 2 and Fig. 7. Based on repeated measurements of the void coefficient, the error in these measurements is between 2 and 3%.

Table 2. Void Coefficient Measurements in 20.32-cm-dia Unreflected Uranium-Molybdenum Cylinders[a]

Radial Position in Void (cm)	Void Coefficient (cents/cm^3)	
	Measured	Computed (Perturbation Theory)
Midplane of U-Mo-SS Annular System with Steel Rod in Center		
2.86	4.62	4.58
3.49	4.72	4.76
4.13	4.80	
4.76	4.78	4.57
5.40	4.43	
6.61	3.75	3.57
7.94	3.26	
9.21	2.27	2.28
9.84	2.03	
U-Mo-SS System, 13.33 cm above Base		
2.86	3.71	3.6
3.49	3.86	3.7
Midplane of U-Mo Solid System		
0.00	7.40	7.9
Midplane of U-Mo Annular System with Void in Center		
2.86	4.15	4.2
4.13	4.54	4.55
9.84	1.81	1.65

a. Void is 0.927-cm-dia, 0.635-cm-long cylinder.

A series of experiments was performed to estimate the amount of hydrogenous reflector that could be safely brought near the reactor during assembly, with the safety device removed from the core. In these experiments the lower 12.70-cm section of the assembly, which was 20.32 cm in

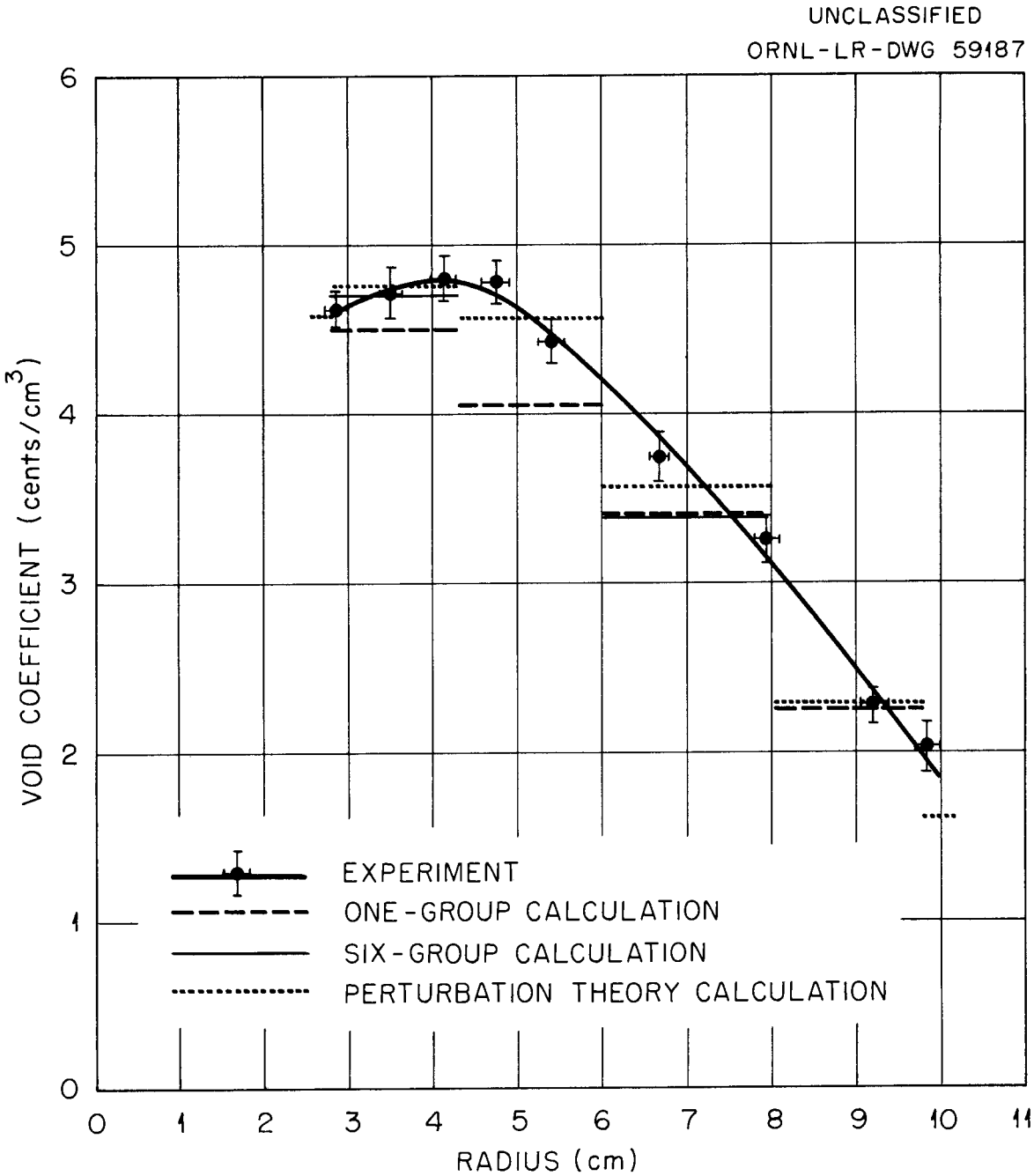

Fig. 7. Void Coefficients as a Function of Radius: 20.32-cm-dia Uranium-Molybdenum Cylinder with a 5.08-cm-dia Steel Core.

diameter, was constructed from disks with an 8.89-cm-dia central hole, while the remainder had a 5.08-cm-dia central hole. The critical heights, determined for nine different reflector conditions, are given in Table 3. The reflector material, Plexiglas, was placed around the assembly and within the central cavity, and, in four instances, 0.025-cm-thick cadmium sheets separated the assembly from the outer reflector.

Table 3. Critical Parameters for a 20.32-cm-dia Annular Cylinder* Under Various Conditions of Reflection

Reflector Condition	Critical Mass (kg of U^{235})	Critical Height (cm)
No Cadmium Between Assembly and Reflector		
2.54-cm-thick Plexiglas on all outer surfaces; void in center	67.5	17.25
2.54-cm-thick Plexiglas on all outer surfaces; Plexiglas in center	52.8	13.86
5.08-cm-thick Plexiglas on all sides; void in center	47.2	12.57
15.2-cm-thick Plexiglas on bottom only; Plexiglas in center	69.4	17.68
15.27-cm-thick Plexiglas on bottom; 2.54-cm-thick Plexiglas on top and on the lower 8.25-cm-section of lateral surface; void in center	75.3	19.05
0.025-cm-thick Cadmium Between Assembly and Outer Reflector		
2.54-cm-thick Plexiglas on all outer surfaces; void in center	70.7	17.98
2.54-cm-thick Plexiglas on all outer surfaces; Plexiglas in center	56.2	14.65
5.08-cm-thick Plexiglas on all outer surfaces; void in center	61.2	15.80
5.08-cm-thick Plexiglas on all outer surfaces; Plexiglas in center	49.8	13.18

* Lower 12.7 cm of assembly as 8.89-cm-dia central hole and remainder has a 5.08-cm-dia central hole.

IV. CALCULATIONS

The calculations were performed with a multigroup transport code, TDC,[3] for the IBM 7090 computer. It is a two-dimensional code in (r,z) cylindrical geometry and uses the S_n method of solution.[4] The S_4 approximation was used throughout.

The group structure and cross sections as listed by Mills[5] and tabulated in Appendix B were used. Iron cross sections were used for steel. Since the unreflected assemblies were all symmetric about the midplane, only ten radial and nine axial spatial intervals were used in the calculation. The spatial mesh is given for the steel-core, unreflected assembly in Table 4. All results are averages over the mesh cells.

Table 4. The Mesh Points: Calculation of the U-Mo Assembly with Steel Core

Radial Point	r(cm)	Axial Point	z(cm)
1	0.00	1	0.000
2	0.20	2	0.207
3	1.10	3	1.033
4	2.30	4	2.067
5	2.54	5	3.307
6	2.80	6	4.650
7	4.30	7	6.097
8	6.00	8	7.233
9	8.00	9	8.474
10	9.80	10	8.784
11	10.16		

3. B. Carlson, C. Lee, and J. Worlton, The DSN and TDC Neutron Transport Codes, LAMS-2346, February 12, 1960.

4. B. G. Carlson, G. I. Bell, Solution of the Transport Equation by the S_n Method, P/2386, PICG, 1958.

5. C. B. Mills, Neutron Cross Sections for Fast and Intermediate Nuclear Reactors, LAMS-2255, January 16, 1959.

A typical running time to convergence on a critical height was 12 min. on the IBM 7090.

The computed critical heights are given in Table 1. The fission density distributions at the midplane are shown in Figs. 4, 5, and 6.

Void coefficients of reactivity were computed by three methods. In the first, using TDC, material was simply removed from a mesh interval and the new configuration calculated. For the second, to save computing time, one-group constants were obtained by weighting the 6-group cross sections with the fluxes from the unperturbed assembly and performing one-group calculations on the perturbed systems, again with TDC. The one-group results are in good agreement with the multigroup results and with experiment, as shown in Fig. 7, and the computing time is reduced to 4 min. per case. The results are also summarized in Table 5.

Table 5. Calculated Void Coefficients for the Steel Core Assembly

		ρ(cents/cm^3) [a]	
Radial Interval (cm)	1 Group[b]	6 Group	Perturbation Theory
Axial Interval = 0. - 0.2 cm			
2.54-2.80		4.7	4.58
2.80-4.30			4.76
4.30-6.00			4.57
6.00-8.00			3.57
8.00-9.80			2.28
9.80-10.16			1.61
Axial Interval = 0.2 - 1.033 cm			
2.80-4.30	4.5		
4.30-6.00	4.1		
6.00-8.00	3.4	3.4	
8.00-9.80	2.5		

a. The effective fraction of delayed neutrons was assumed to be 0.0068.
b. k for the unperturbed assembly was 1.0056.

The third method, the use of perturbation theory, is both the most economical and most satisfying means of computing void or material replacement coefficients. The fractional change in the multiplication constant in first order perturbation theory is given by

$$\frac{\Delta k}{k} = \frac{1}{Ak} \int_{\underline{r}} d\underline{r} \int_0^\infty du \int_{\underline{\Omega}} d\underline{\Omega}\, \phi^+(\underline{r},u,\underline{\Omega})\, X(u) \int_0^\infty du' \int_{\underline{\Omega}} d\underline{\Omega}'\, \Delta[\nu\Sigma_f(\underline{r},u')]\, \phi(\underline{r},u',\underline{\Omega}')$$

$$+ \frac{1}{A}\Bigg\{ \int_{\underline{r}} d\underline{r} \int_0^\infty du \int_{\underline{\Omega}} d\underline{\Omega}\, \phi^+(\underline{r},u,\underline{\Omega})$$

$$\int_0^u du' \int_{\underline{\Omega}'} d\underline{\Omega}'\, \Delta[\Sigma_s(\underline{r},u')]\, \phi(\underline{r},u',\underline{\Omega}')\, K(u'\to u,\ \underline{\Omega}'\to\underline{\Omega}) \Bigg\}$$

$$- \frac{1}{A} \int_{\underline{r}} d\underline{r} \int_0^\infty du\, \Delta[\Sigma_T(\underline{r},u)] \int_{\underline{\Omega}} d\underline{\Omega}\, \phi^+(\underline{r},u,\underline{\Omega})\, \phi(\underline{r},u,\underline{\Omega}), \qquad (1)$$

where

$$A = \frac{1}{k} \int_{\underline{r}} d\underline{r} \int_0^\infty du \int_{\underline{\Omega}} d\underline{\Omega}\, \phi^+(\underline{r},u,\underline{\Omega})\, X(u) \int_0^\infty du'' \int_{\underline{\Omega}'} d\underline{\Omega}'\, \nu\Sigma_f(\underline{r},u')\, \phi(\underline{r},u',\underline{\Omega}').$$

Though it is tempting to say in the last term of Eq. 1:

$$\int_{\underline{\Omega}} d\underline{\Omega}\, \phi^+(\underline{r},u,\underline{\Omega})\, \phi(\underline{r},u,\underline{\Omega}) = \frac{1}{4\pi} \int_{\underline{\Omega}} d\underline{\Omega}\, \phi^+(\underline{r},u,\underline{\Omega}) \int_{\underline{\Omega}'} d\underline{\Omega}'\, \phi(\underline{r},u,\underline{\Omega}') \qquad (2)$$

such an assumption can lead to large errors where the flux is highly anisotropic, for example, toward the edges of the assembly. The angular fluxes and adjoints must be obtained and their product integrated numerically over solid angle. Void coefficients for the steel-core assembly computed with the assumption expressed in Eq. 2 are compared in Fig. 8 with those computed correctly.

The void coefficients computed by perturbation theory are compared to those computed by the first two methods and those obtained from experiment in Fig. 7 and in Table 2 for the steel core assembly at or near midplane. Only the 6-group void coefficient with radial range 2.8 - 4.3 cm was computed at midplane. All other group void coefficients were calculated one axial interval from midplane. Figure 9 gives the perturbation-theory void coefficients for all the axial mesh intervals. The lines indicating the radial mesh intervals have been omitted for clarity. Figure 10 compares experimental and perturbation-theory void coefficients for the void core assembly at midplane.

Perturbation theory also provides the best means of calculating neutron lifetimes in fast assemblies, since the lifetimes are so short. The lifetime is equal to the change in the multiplication constant resulting from the addition throughout the assembly of an absorber with cross section numerically equal to 1/v (Ref. 6). The lifetimes were computed to be 9.31×10^{-9} sec, 9.00×10^{-9} sec, and 8.19×10^{-9} sec for the steel core, void core, and solid assemblies, respectively. The lifetime in Godiva I was measured to be 6.23×10^{-9} sec (Ref. 2).

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6. A. F. Henry, Nuclear Sci. and Eng. 3, 52 (1958).

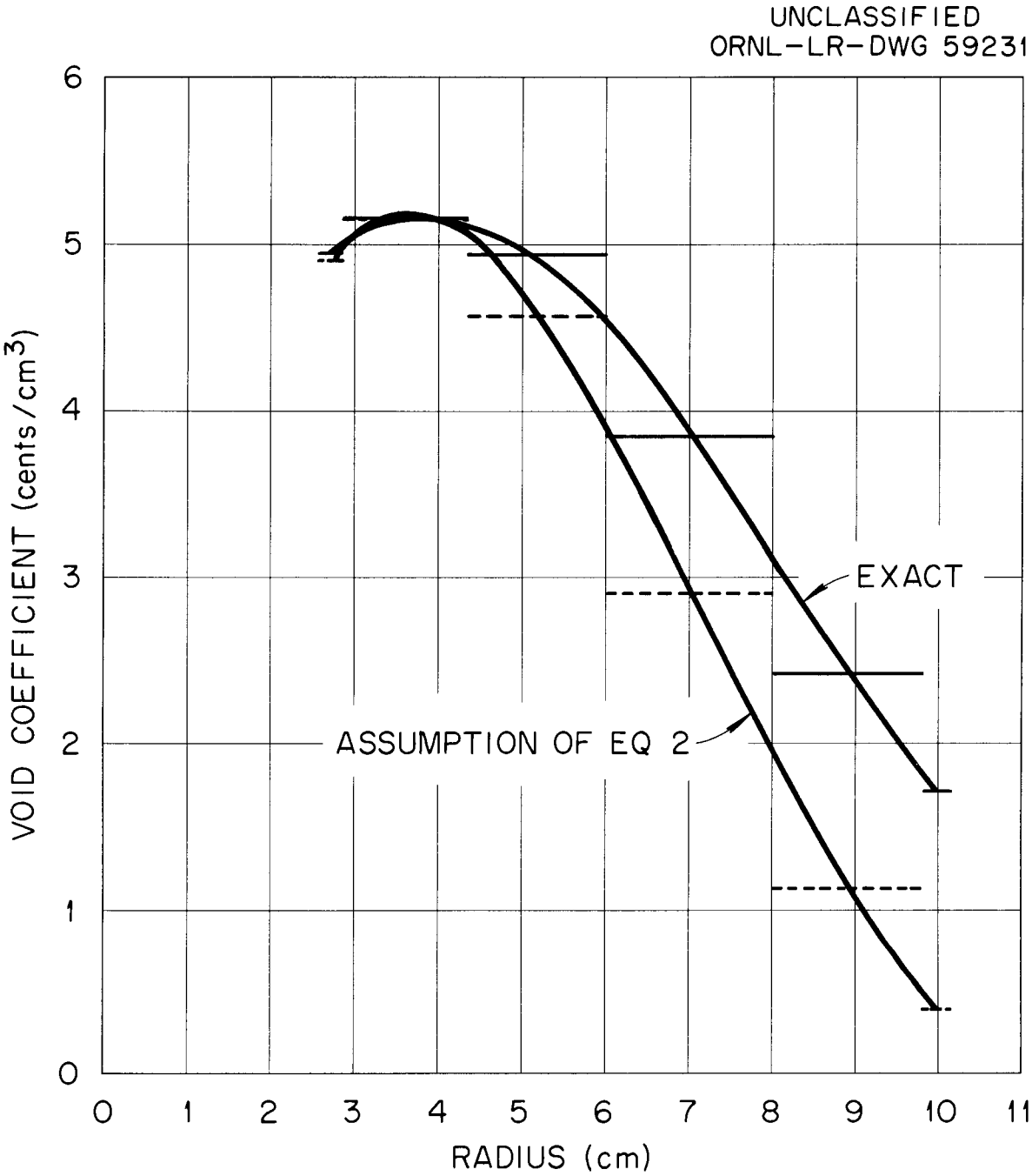

Fig. 8. Void Coefficients for the Steel-Core Assembly; Comparison of Calculation Methods.

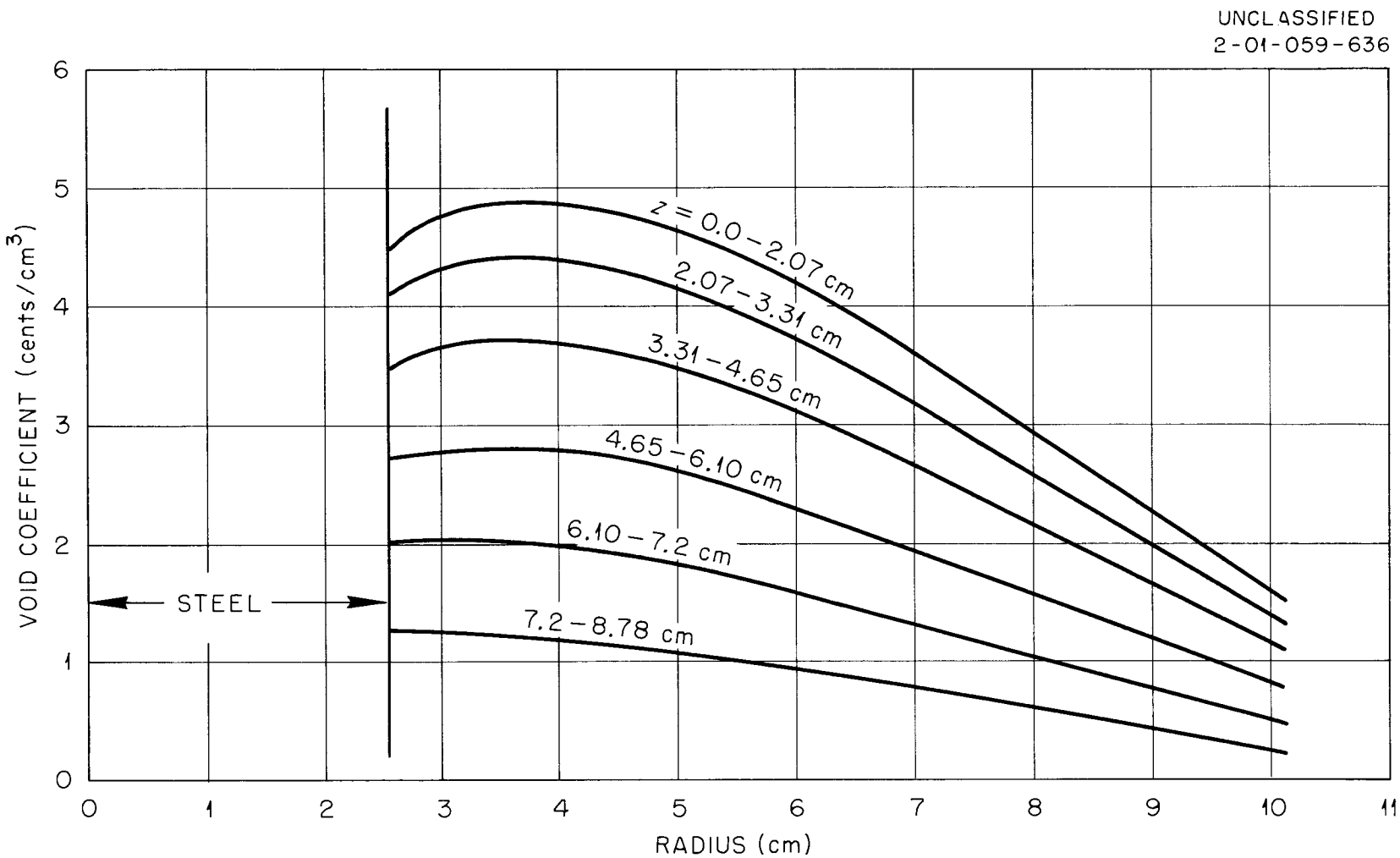

Fig. 9. Perturbation Theory Void Coefficents of Reactivity as a Function of Radius, for the Steel Core Assembly.

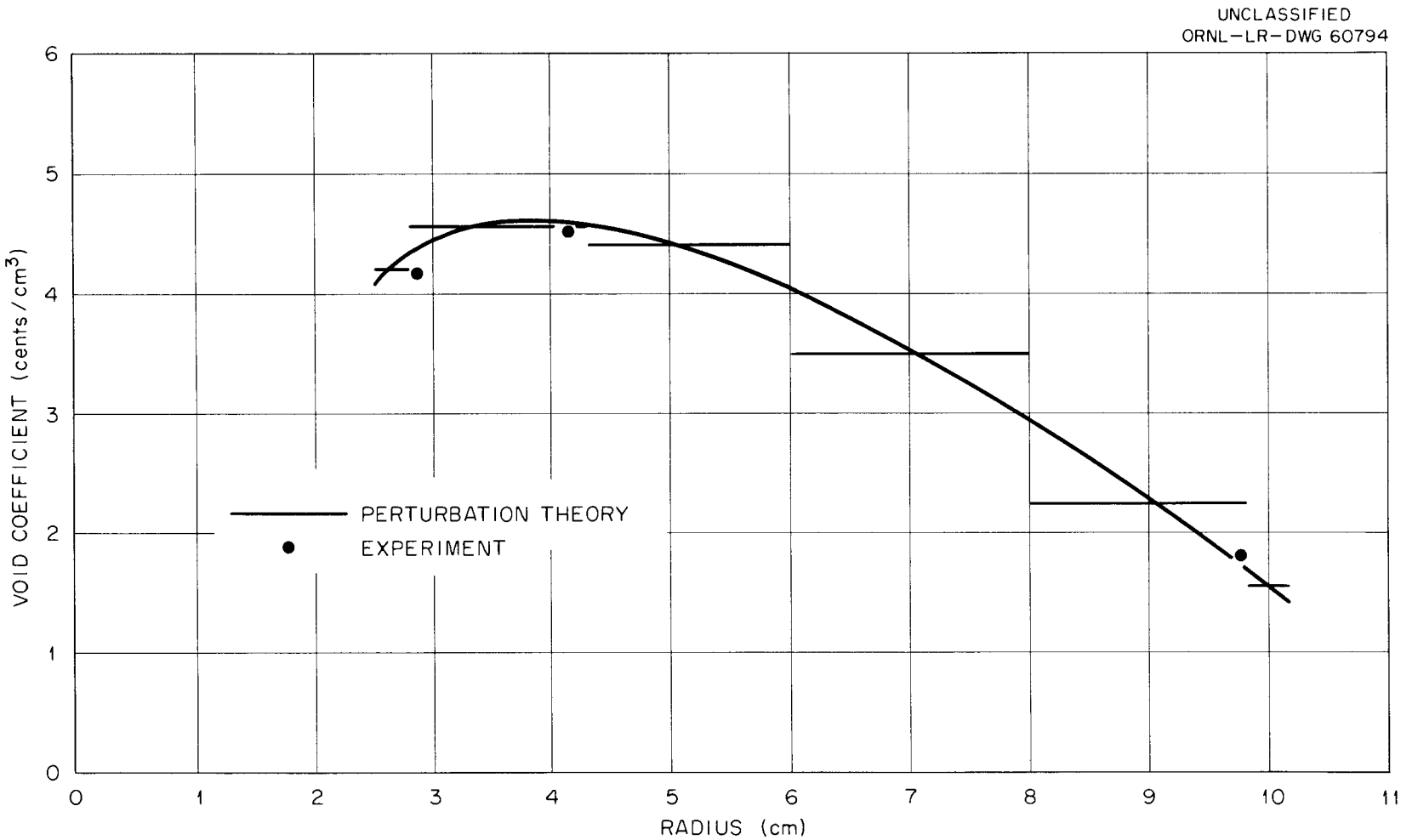

Fig. 10. Void Coefficient as a Function of Radius: Void Core at Midplane.

Godiva I was composed of unalloyed uranium in which neutron lifetimes should be somewhat shorter than in the U-Mo assemblies.

V. RESULTS AND COMPARISONS

Measured and calculated critical heights of the unreflected assemblies are compared in Table 1. The calculations predict critical heights which are 6% to 8% less than those experimentally determined.

From the measured reactivity change resulting from a variation in height of each assembly, the calculated multiplication constants of the critical configurations are about 2% too large. Although the cross sections which were used give excellent results for fast, unalloyed -uranium assemblies,[5] the neutron spectrum in U-Mo is softer than in pure uranium, and group constants, especially $\nu\sigma_f$, characteristic of a somewhat lower neutron energy should ideally be used. The error in the measured number of neutrons per fission in the energy range of the calculations is $\pm$ 2.5% (Ref. 7). This error, the errors in cross sections, and the uncertainties in obtaining group constants could easily account for the differences between the calculated and experimental values of the critical parameters.

Satisfactory agreement between measured and calculated midplane fission-density distributions in the unreflected assemblies is shown in Figs. 4 to 6 in which the fission densities have been normalized to unity at 2.86 cm. The introduction of the steel core reduces the maximum-to-minimum radial fission-density ratio from 4.3 to 3.1.

Measured and calculated void coefficients for the unreflected assemblies are compared in Figs. 7 and 10. Since most of the group coefficients

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7. Reactor Physics Constants (ed. by R. Avery) ANL-5800, July, 1958.

for the steel core assembly were calculated one axial interval from mid-plane, they should be lower than the experimental results. The coefficients computed by perturbation theory are in good agreement with both those from experiment and those from group calculations.

VI. CONCLUSIONS

The S_4 transport theory approximation and the cross sections suitable for calculations of fast uranium assemblies provide a satisfactory method of computing the critical mass of unreflected uranium alloyed with small amounts of other heavy materials. The corresponding calculated fission distributions and material-replacement reactivity coefficients may be expected to be good. It is further concluded that both the forward and adjoint angular fluxes must be used in perturbation theory calculations.

VII. ACKNOWLEDGEMENTS

The authors wish to acknowledge the help of various members of the staff of the Oak Ridge National Laboratory Critical Facility in the performance of the experiments. They are especially grateful to G. E. Whitesides for carrying out the machine calculations.

APPENDIX A. PROPERTIES OF MATERIALS

Table A-1. Impurity Content of U-Mo Alloy

Impurities	Abundance (ppm)
Si	60 - 300
Fe	10 - 300
C	150 - 300
Alkali metals	<7
Gases: O, N, H	60 - 400
Zr	<20
Maximum total	580 - 830

Table A-2. Physical Properties of U-Mo Alloy

Wt % Molybdenum	10
Tensile strength	133,000 psi
Yield strength	127,000 psi
Modulus of elasticity	12.5×10^6
Elongation	15%
Density	17.08 g/cm^3

APPENDIX B. CROSS SECTIONS USED IN THE CALCULATION OF THE UNREFLECTED ASSEMBLIES

The cross sections used were those as tabulated by Mills[5] except those of molybdenum. Molybdenum cross sections were obtained from Howerton's compilation[8] by weighting with the fission spectrum and using the evaporation model in the computation of inelastic transfer. Table B-1 gives the group structure. Table B-2 lists the cross sections, where

$$\sigma_{tr_T} = \sigma_a + (1 - \bar{\mu})\sigma_s + \sum_j \sigma_x(i \rightarrow j) ,$$

$$\sigma_{tr_s} = (1 - \bar{\mu})\sigma_s ;$$

$\sigma_x(i \rightarrow j)$ = microscopic cross section for the transfer from group i to group j.

Table B-1. The Group Structure

Group	Upper Energy Limit (Mev)	Lower Energy Limit (Mev)	Lethargy Width
1	10	3	1.204
2	3	1.4	0.762
3	1.4	0.9	0.442
4	0.9	0.4	0.811
5	0.4	0.1	1.386
6	0.1	0.017	1.772

8. R. J. Howerton, Semi-Emperical Neutron Cross Sections, UCRL-5351 (1958).

Table B-2. Cross Sections

Group	$\nu\sigma_f$	σ_{tr_T}	σ_{tr_s}	$\sigma_x(i \rightarrow i + k)$				
				k = 1	k = 2	k = 3	k = 4	k = 5
		U^{235}						
1	3.557	4.25	1.2	0.27	0.37	0.65	0.44	0.06
2	3.196	4.50	1.77	0.24	0.67	0.45	0.07	
3	3.087	4.65	2.30	0.55	0.4	0.07		
4	2.988	5.20	3.42	0.35	0.08			
5	3.518	7.90	6.16	0.08				
6	6.125	12.4	9.06					
		U^{238}						
1	1.725	4.0	1.254	0.33	0.46	0.79	0.53	0.07
2	1.213	4.4	1.825	0.35	0.96	0.64	0.09	
3	0.108	4.5	2.906	0.80	0.55	0.10		
4	0	5.25	4.53	0.50	0.08			
5	0	8.20	7.96	0.08				
6	0	11.8	11.4					
		Fe						
1	0	1.713	0.50	1.21	0	0	0	0
2	0	2.322	1.5	0.477	0.322	0.023	0	
3	0	2.610	1.899	0.335	0.346	0.03		
4	0	2.764	2.641	0.123	0			
5	0	2.816	2.744	0.071				
6	0	4.377	4.377					
		Mo						
1	0	3.94	1.91	0.64	0.49	0.60	0.26	0.032
2	0	4.77	3.15	0.44	0.73	0.40	0.047	
3	0	6.57	5.46	0.61	0.44	0.052		
4	0	7.94	7.64	0.26	0.037			
5	0	9.0	8.86	0.14				
6	0	8.0	8.0					

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