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Analysis of Heavy-Water-Moderated Cluster-Type Fuel Lattices by Cluster Physics Code MESSIAH

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This paper presents a comprehensive analysis performed by a new cluster analysis code 'MESSIAH' on reactor physics constants measured in the critical facility for a pressure-tube-type, heavy-water-moderated reactor. The MESSIAH code system utilizes the method of the collision probability to solve the neutron transport equation. The effective space dependent cross sections are calculated in the thermal and resonance energy range before the eigenvalue calculation for the whole energy range. With use of these cross sections, the multi-group, space dependent transport equation is solved, and the flux distribution, spectrum and k_{eff} are obtained to the input bucking. In the above three steps the method of the collision probability is used consistently and extensively. The treatment of leakage neutrons from lattices in MESSIAH is also confirmed by an independent method using a Monte Carlo calculation. The calculated reactor physics constants, especially the micro-parameters and the activation traverse of Dy, agreed fairly well with the experiment. The diffusion calculation with use of the group constants calculated by MESSIAH predicts the reactivity of 0% void core excellently (<0.12%). However, for a 100% void core, the calculated reactivity was slightly lower than the experiment (~0.74%), which was attributed to over prediction of the diffusion constants.

KEYWORDS: heavy water reactors, collision probability method, cluster-type fuel assembly, cluster physics code, lattice parameters, void reactivity, criticality analysis, neutron leakage, advanced thermal reactor, computer codes

I. INTRODUCTION

The heterogeneity for the clustered fuel of heavy water moderated reactors is a mixture of the following two components or superposition of two concepts; (1) a fuel pin and its surrounding coolant and (2) clustered fuel and coolant confined within a pressure and calandria tubes, and heavy water moderator.

The other characteristic of the cluster-type fuel assembly is the large area of its unit cell. The lattice pitch is usually greater than 20 cm, which is equivalent to several mean free paths for thermal neutrons in heavy water. As a result of this large unit cell, criticality can be achieved with very few assemblies for a clean core. Therefore, if the core is loaded with many fuel assemblies, the heavy water level has to be kept low to increase neutron leakage along the vertical direction. The neutron physics constants measured in this pan-cake shaped core are usually affected by the anisotropic neutron leakage.

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In the present paper, we will analyze the reactor physics constants measured in the DCA core⁽⁷⁾ with a new code system 'MESSIAH'. Deuterium Critical Assembly (DCA) is a critical experiment facility for 'FUGEN'⁽⁸⁾, which is a heavy-water-moderated, boiling-light-water-cooled, pressure-tube-type power reactor developed in Japan. The MESSIAH is developed by taking into account the characteristics of a unit cell of a cluster-type fuel lattice, namely, (1) large complicated unit cell geometry with strong heterogeneity and (2) anisotropic neutron leakage from the core.

Since the appearance of the THERMOS code⁽⁹⁾, the method of solving the Boltzmann equation for the neutron transport problem has been centered on the collision probability method. There are other promising procedures such as the response matrix method⁽¹⁰⁾, which is, however, rarely used in a production phase code.

The typical cell analysis codes that have been used for developing FUGEN are METHUSELAH-II⁽¹¹⁾, CLUSTER⁽¹²⁾ and WIMS-D⁽¹³⁾. All of these codes utilize the collision probability method more or less, in accordance with the needs required for each code. The collision probability method is believed to be one of the few reliable analytical methods to solve the integral transport equation of neutrons in the complicated unit cell geometry, and also is one of the few procedures proven by many applications to power reactors as well as critical assemblies. Therefore, we selected the collision probability method as the basic numerical technique in the MESSIAH code system.

However in view of the application to the DCA core, existing codes seemed unsatisfactory in the use of the collision probability method. That is; (1) the collision probability method is used approximately to describe the neutron distribution in a unit cell⁽¹⁴⁾, (2) the thermal neutrons are treated in one group⁽¹⁵⁾ or two overlapping group model⁽¹⁶⁾, (3) the effective resonance cross section is calculated by the equivalent theorem with Dancoff-Ginsburg correction⁽¹⁷⁾, and (4) the space and energy variables are calculated separately, which results in unsatisfactory evaluation of the neutron leakage effect on the flux distribution and spectrum. The objective in developing the new code system 'MESSIAH' is to overcome the above problems by extensive and consistent use of the collision probability method.

Although the collision probability method is a well established procedure, one of the important disadvantages is that it requires long computing times. Even with the use of present computers, only the calculation of the two-dimensional collision probability is feasible. Therefore, we are obliged to depend on some approximation to treat the axial leakage of neutrons at least. As the critical water level of the DCA core is low compared with its radius, the experimental data might be affected by this effect. To check the approximation used in MESSIAH, and to obtain the magnitude of this leakage effect, we simulated the fuel assembly of the DCA core with the Monte Carlo code KENO-IV⁽¹⁸⁾. An analysis by the Monte Carlo method can be regarded as, in a sense, a kind of an independent experiment which enables us to control the neutron leakage artificially.

The other feature of the MESSIAH code system is that the new scattering kernel for heavy water is incorporated in the nuclear library. In a thermal neutron reactor the major reactor physics property is determined by the moderator and the thermalization of neutrons is essentially governed by its scattering kernel. The scattering kernel for heavy water is calculated in the incoherent scattering approximation⁽¹⁹⁾⁽²⁰⁾, which agreed well with the scattering law for heavy water.

II. DESCRIPTION OF MESSIAH

1. Outline

The MESSIAH code system has been developed to comprehend the lattice physics of a pressure-tube-type fuel assembly by calculating reaction rate distributions for the lattice, an eigenvalue for a given buckling, and average cell constants for use in overall reactor calculations. The conceptual flow of the calculation is shown in Fig. 1. In MESSIAH, the neutron transport equation is solved three times. The collision probability method is consistently applied to solve each equation. The first transport equation is that for thermal neutrons with a given epithermal source distribution. A few group cross section set is calculated with the use of the fine group (50 groups) neutron flux. The second equation is concerned with the resonance. The flux for each resolved resonance level is calculated for the super fine energy mesh. The effective resonance cross section for each resonance nucleus is generated following the GAM-I⁽¹³⁾ type fine group structure. Finally the eigenvalue equation is solved for the whole energy range using the cross section library corrected by the above thermal and resonance cross sections.

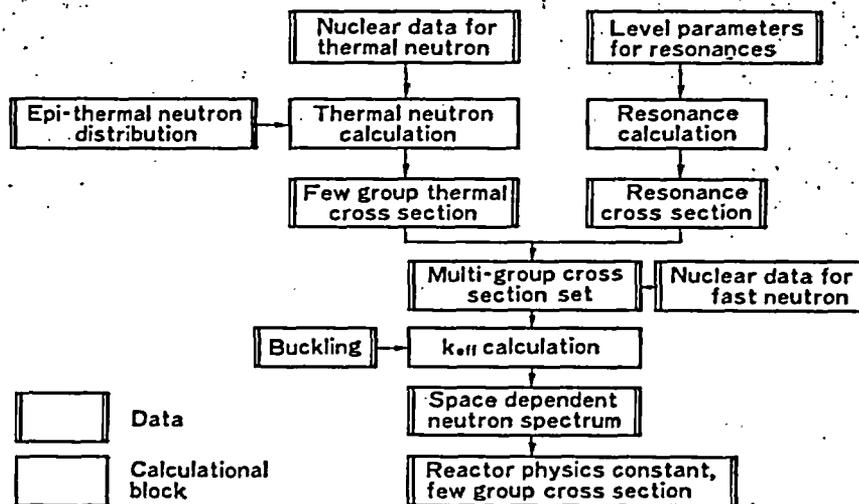


Fig. 1 Conceptual calculation flow of MESSIAH

One of the difficult problems we are forced to resolve in applying the collision probability method to an actual, finite system is the evaluation of the neutron leakage from the system being analyzed. Essentially the evaluation of leakage neutrons in MESSIAH is performed in the diffusion approximation.

2. Basic Nuclear Library

The basic nuclear library for MESSIAH is composed of three parts, namely the fast cross sections, thermal cross sections and resonance parameters. The fast cross section library was generated from ENDF/B-III⁽¹⁴⁾ with use of the SUPERTOG code⁽¹⁴⁾. We used GAM-I type group structure (67 groups) for the fast neutron library. The resonance level parameters given in ENDF/B-III were transferred to the library for the resonance calculation. We adopted 50 groups for thermal neutrons of energy less than 1.98 eV. The reaction cross sections for thermal neutrons were also generated from ENDF/B-III with use of the FLANGE-2⁽¹⁵⁾ code.

The scattering kernel of a hydrogen atom in light water is calculated by the frequency

distribution or the spectral density proposed by Haywood⁽¹⁴⁾. The theoretical model used for calculating the heavy water scattering kernel is the incoherent scattering approximation to the kinetic model of the heavy water molecule by Kadotani & Iijima⁽¹⁰⁾⁽¹¹⁾. The actual calculation of the kernels were performed with the GASKET⁽¹²⁾-FANGE⁽¹³⁾ code combination. The total cross section of heavy water is compared with the experimental data⁽¹⁹⁾ in Fig. 2. The dip in the experimental cross section around 10^{-3} eV is due to the coherent scattering effect, which was neglected in the present calculation.

3. Treatment of Neutron Leakage

The treatment of neutron leakage from the lattice being analyzed in MESSIAH is essentially a simple diffusion approximation. That is, the total and absorption cross sections are corrected by adding DB^2 . The MESSIAH code system is provided with two methods for calculation of the diffusion coefficient. The simple one is defined as $1/(3\Sigma_{tr})$, where Σ_{tr} is the flux-volume average transport cross section for a unit cell. The second is more sophisticated definition of the diffusion coefficient based on Benoist's model⁽²⁰⁾,

$$D = \frac{\iint_{\text{cell}} \frac{P(r \rightarrow r')}{3\Sigma_{tr}(r')} \phi(r) dr' dr}{\int_{\text{cell}} \phi(r) dr}, \quad (1)$$

where $\Sigma_{tr}(r)$ is the transport cross section at a position r , $\phi(r)$ the neutron flux at r , and $P(r \rightarrow r')$ the collision probability that a neutron appearing isotropically at r will make its first collision at r' . Since we need to know in this procedure the neutron flux prior to the calculation of D , the whole calculation becomes iterative.

4. Collision Probability⁽²¹⁾

The one-speed form of the equation to be solved in the collision probability method with isotropic source and scattering is

$$\Sigma_i^t \phi_i V_i = \sum_{j=1}^N P_{j \rightarrow i} V_j (\Sigma_j^t \phi_j + s_j), \quad i=1, 2, \dots, N, \quad (2)$$

where ϕ_i is the flat flux and V_i the volume of i -th region, Σ_i^t and Σ_i^s the total and scattering cross section of i -th region, $P_{j \rightarrow i}$ the collision probability defined as above, the source term s_j the external source for a source problem or the fission neutrons divided by the eigenvalue k_{eff} for an eigenvalue problem and N the total number of divided regions. The extension to the multi-group formula is straightforward. For the cylindrical geometry $P_{j \rightarrow i}$ is calculated by the double integral of Bickley function⁽²²⁾. In MESSIAH the same scheme of this integration used as CLUP⁽²³⁾ and Eq. (2) is solved with adopting PIJF code⁽²³⁾.

Since the assumption of isotropic scattering is essential for the collision probability method, we took the anisotropic scattering process into account by replacing the total cross section with the transport cross section. The self scattering cross section is modified artificially to maintain the balance of the cross section.

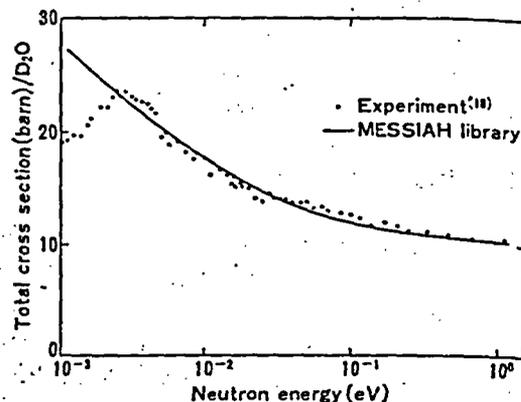


Fig. 2 Comparison of total cross section for heavy water molecule

5. Resonance Calculation

The energy and space dependent resonance flux is calculated for each resolved resonance level separately, neglecting the effect from higher energy levels and overlapping of resonance levels. As far as this assumption is valid, the neutron balance equation can be solved with much finer energy mesh to express each resonance profile, and with the boundary condition of the spatially flat $1/E$ flux at the highest energy mesh. The collision probability which is necessary to calculate the resonance flux is evaluated by interpolation of the pre-calculated table for different cross sections. The scattering source in Eq. (2) will be generated by slowing down neutrons for the resonance calculation. The integral to calculate this source term is performed in either NR (narrow resonance) or WR (wide resonance) approximation. At the user's option, this integral can be treated rigorously. The resonance calculation module for MESSIAH is called RICM-M, which is an updated version of the RICM code⁽²⁰⁾.

Since, in the final eigenvalue calculation in MESSIAH, the fuel rods in a cluster are treated heterogeneously, we should calculate the resonance cross section. Therefore, the usual procedure is used to generate group-wise cross sections for each resonance level.

6. Procedure and Condition for MESSIAH Calculation

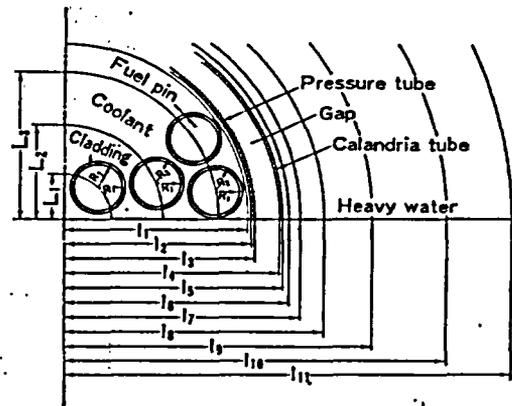
(1) Thermal Cross Sections

The nuclear library for the thermal neutron cross sections has 50 groups up to 1.98 eV. The fine-group cross sections were reduced to three groups, namely 1.98~0.6 eV, 0.6~0.45 eV and 0.45~ 2.5×10^{-4} eV, by the THERMOS type calculation. The source distribution above 1.98 eV was assumed to be the measured epi-Cd activation distribution of Au-foil.

The modeling of a unit cell is shown in Fig. 3. In this model, the fuel pins were not subdivided into annular rings.

(2) Treatment of Resonance Cross Section

The resolved levels of ^{235}U and ^{238}U were treated rigorously, and the cross sections for the unresolved energy region were evaluated assuming infinite dilution. The lower boundary energies for this region were 4 keV and 83 eV for ^{235}U and ^{238}U , respectively. The geometrical regions were greatly simplified due to limitations of RICM-M module. We distinguished three rings for the fuel pins, cladding, two regions for the coolant; one homogenized region consisting of the calandria tube and moderator.



Region	Material	Outer radius (cm)	
		22.5 cm pitch	25.0 cm pitch
1(R ₁)	Fuel	0.74	0.74
2(R ₂)	Cladding	0.8265	0.8365
3(R ₃)	Fuel	0.74	0.74
4(R ₄)	Cladding	0.8265	0.8365
5(R ₅)	Fuel	0.74	0.74
6(R ₆)	Cladding	0.8365	0.8365
7(L ₁)	Coolant	0.8125	0.8125
8(L ₂)	Coolant	3.0	3.0
9(L ₃)	Coolant	4.7575	4.7575
10(L ₄)	Coolant†	5.72	5.72
11(L ₅)	Coolant†	5.84	5.84
12(L ₆)	Pressure tube	6.04	6.04
13(L ₇)	Gap	6.625	6.625
14(L ₈)	Calandria tube	6.825	6.825
15(L ₉)	Heavy water	6.9836	7.0770
16(L ₁₀)	Heavy water	7.3008	7.4150
17(L ₁₁)	Heavy water	7.9353	8.2020
18(L ₁₂)	Heavy water	8.8870	9.3800
19(L ₁₃)	Heavy water	13.791	11.244
20(L ₁₄)	Heavy water	12.694	14.105

† Hanger wire (Al) is diluted homogeneously into coolant of this region.

Fig. 3 Calculation model of DCA cluster for MESSIAH

(3) Group Constants for Bare Fuel Part

For the calculation of k_{eff} , the DCA core was modeled as a multi-layer structure (see Fig. 1 and Table 2 of Ref. (1)). The cross section for the structural material was evaluated, assuming the fission plus $1/E$ spectrum for fast neutrons and the Maxwellian spectrum at room temperature for thermal neutrons. Concerning the bare fuel part (layer) where no coolant and moderator, but only fuel pins exist, however, the precise group constants should be calculated because more than 1% of the neutron multiplication is made in this part. We used only the WIMS-D code for this calculation, which was selected to obtain a clear comparison of the group constants for the core region calculated by both MESSIAH and WIMS-D codes. But, as far as the diffusion coefficients are concerned, the calculated values from this code were questionable. Therefore, we calculated the diffusion coefficient with MESSIAH in Benoist's approximation. Here, we used the transport cross section calculated by WIMS-D.

(4) Flux and k_{eff} Calculation

The effective multiplication factor and the associated fluxes were calculated with the updated cross section library, which consists of the fast cross section (62 groups) corrected by the resonance shielding effect and the thermal cross sections (3 groups). The experimental total buckling was used for the leakage calculation.

III. ANALYSIS OF EXPERIMENT

1. DCA Core and Clustered Fuel Assembly

The DCA fuel cluster⁽¹⁾ is composed of 28 UO_2 fuel pins of 1.2% ^{235}U enrichment with Al cladding, and the core is made of a square 22.5 cm or 25.0 cm pitch lattice of clusters. The structure of DCA is an Al tank of 150 cm radius. However, there are a few regions of structural material along the axial direction. The most important region is the dry fuel pin above coolant and moderator

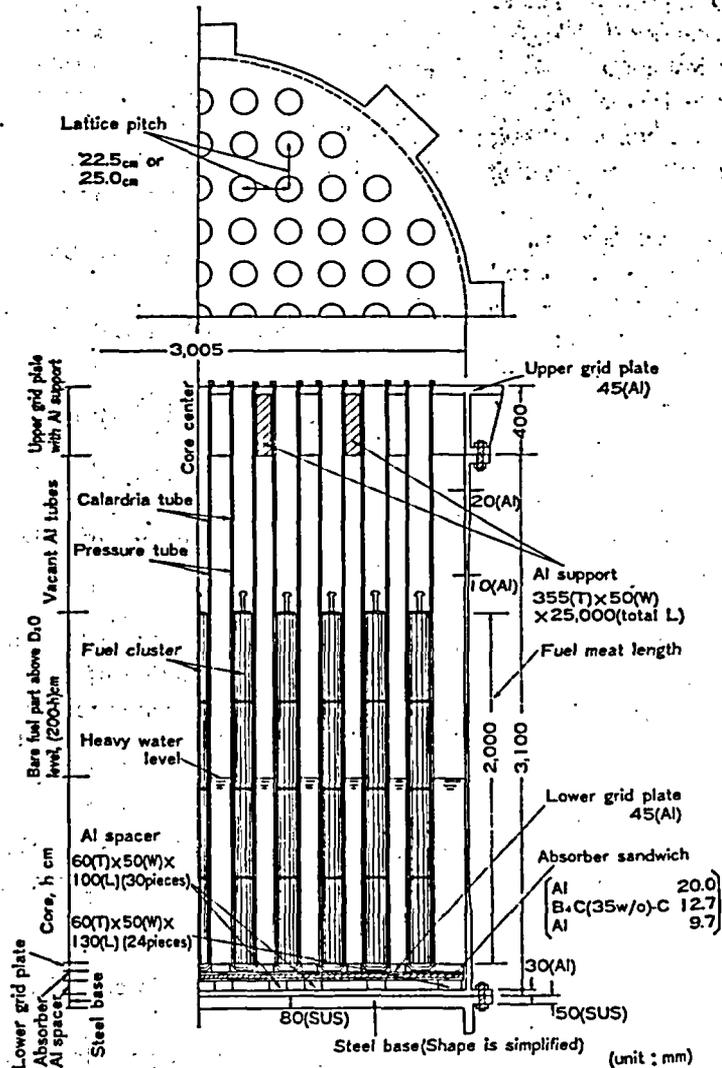


Fig. 4 DCA core configuration (25 cm pitch lattice, schematic)

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level (bare fuel part), for the region contains fissile material. We simplified other regions like the upper and lower grid plates into a multi-layer model for criticality analysis (see Fig. 4).

The effective void fraction of the experimental coolant to simulate actual boiling and the loss of coolant condition was changed from 0 to 100% (complete loss of coolant) with two intermediate fractions of 30 and 70%. These two conditions were simulated with D₂O-H₂O-H₃BO₃ mixtures specified in Table 1. Here, adjustment of the moderation ratio and the thermal absorption cross section was made for obtaining those of the actual voided coolant.

Table 1 Composition of experimental coolant mixture

Coolant void fraction (%)	0	30	70	100
Material (%)				
H ₂ O	100	63.17	18.07	0
D ₂ O	0	36.82	81.91	0
H ₃ BO ₃	0	0.0092	0.0215	0

2. Reactor Physics Parameters Analyzed

(1) Micro-parameters

The micro-parameters ρ^{25} , δ^{25} and δ^{28} defined by Eqs. (3)~(5) are analyzed.

$$\rho^{25} = \int_{E_{Cd}}^{\infty} \sigma_{f25}(E)\phi(E)dE / \int_0^{E_{Cd}} \sigma_{f25}(E)\phi(E)dE, \tag{3}$$

$$\delta^{25} = \int_0^{\infty} N^{25}\sigma_{f25}(E)\phi(E)dE / \int_0^{\infty} N^{25}\sigma_{f25}(E)\phi(E)dE, \tag{4}$$

$$\delta^{28} = \int_{E_{Cd}}^{\infty} \sigma_{f28}(E)\phi(E)dE / \int_0^{E_{Cd}} \sigma_{f28}(E), \tag{5}$$

where we used the conventional nomenclature, and 25 and 28 designates ²³⁵U and ²³⁸U, respectively. We can relate these parameters to the gross neutron spectrum; ρ^{25} to resonance neutrons, δ^{25} to lower edge of neutrons of epi-Cd energy region and δ^{28} to fast neutrons above the fission threshold of ²³⁸U.

(2) Thermal Neutron Distribution

Since micro-parameters were measured only in the fuel pins (in each fuel ring) within the pressure tube, we analyzed the distribution of Dy activation in a unit cell including the pressure and calandria tubes and heavy water region. As the activation in the epi-Cd energy region was subtracted by using Cd-ratio measured, this traverse is almost the same as the thermal neutron distribution.

(3) Criticality

We analyzed the critical water level at 22.0±4°C. Here, coolant and moderator level were kept at the same height. The purity of heavy water was 99.4 mol% and impurity was assumed to be light water.

(4) Void Reactivity

The coolant void coefficient of reactivity is one of the most important constant for a FUGEN type power reactor, because this coefficient is apt to have a positive value, which is unfavorable from not only the safety aspect but also the operational stability. For estimation of this coolant void coefficient, the reactivity introduced by the change of coolant void fraction was measured by the pulsed neutron source method⁽²⁵⁾.

IV. RESULTS AND DISCUSSION

An analysis of 22.5 and 25.0 cm pitch lattice with 0, 30, 70 and 100% void clusters has been performed and the results have been compared with the experiment⁽²⁾.

1. ρ^{23}

In Fig. 5 the calculated micro-parameter ρ^{23} is compared with the experiment. We compared them with calculation by WIMS-D⁽³⁾ also in this discussion. The agreement of calculation with the experimental data other than 100% void cores is fairly good except for 70% void, 22.5 cm pitch and 30% void, 25.0 cm pitch lattices. These discrepancies may be ascribed to random errors in the experiment.

The anomalous distribution of ρ^{23} is seen for 100% void lattice especially in 22.5 cm pitch lattice. This has remained as a long-pending problem which has not been explained analytically. The phenomenon is also unexplained by MESSIAH. However, we propose here the following interpretation. The sub-Cd activation within fuel pins were measured by ^{235}U and ^{238}U foils which should agree with each other, because the (n, γ) cross section of ^{235}U and the fission cross section of ^{235}U have $1/v$ type energy dependence, except for the small deviation of ^{235}U fission cross section towards the Cd-cutoff energy.

Figure 6⁽²⁶⁾⁽²⁷⁾ shows that agreement of the measured sub-Cd activation of ^{235}U and ^{238}U foils are ascertained for 0, 30 and 70% void cores, but not for 100%

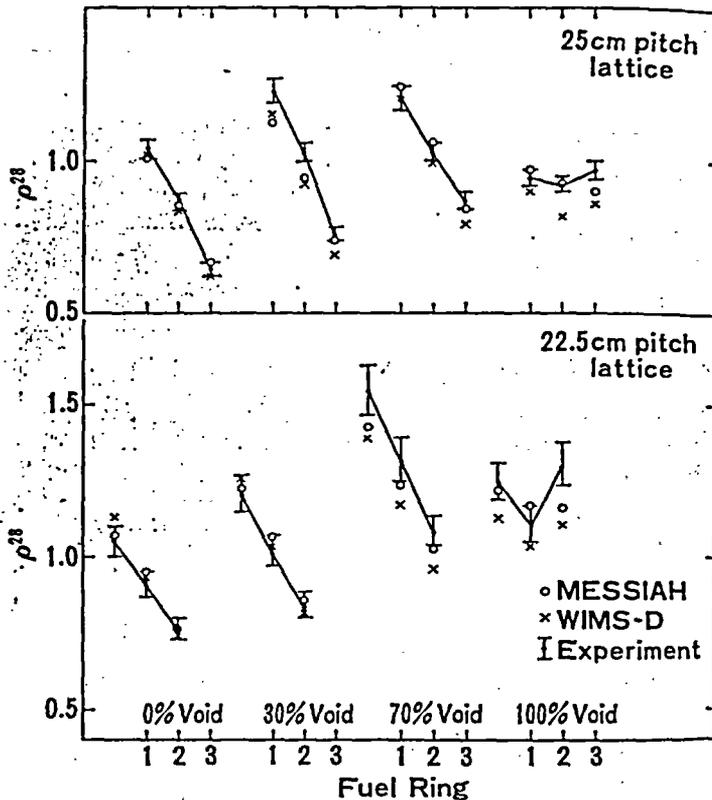


Fig. 5 Comparison of ρ^{23} with experiment (22.5 and 25.0 cm pitch lattices)

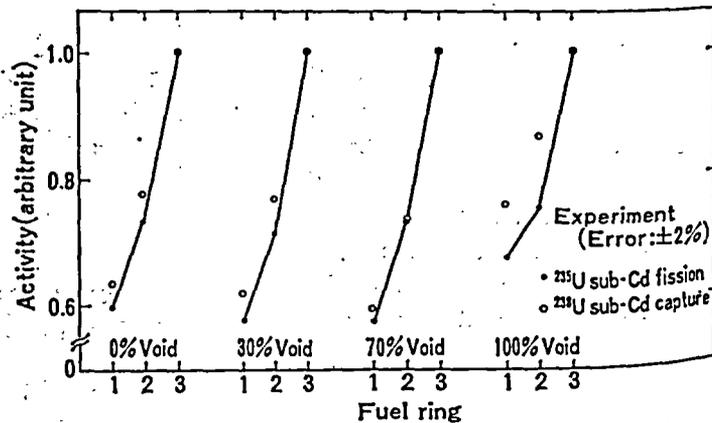


Fig. 6 Distribution of sub-Cd activity of ^{235}U and ^{238}U foils (normalized to 1.0 at the 3rd ring)

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void core. On the other hand, the calculated results of these sub-Cd activity agreed with each other even for the 100% void core. Therefore, we suspect that an experimental error in the measurement of the ^{235}U Cd ratio exists for this lattice. That is, the Cd covered activity of ^{235}U was measured smaller than the actual one (see APPENDIX).

If the sub-Cd activation of ^{235}U at the third ring is increased to the level measured by ^{235}U for 100% void core, the ρ^{235} of this particular ring will be decreased in amount of about 10% and the distribution becomes flatter. This tendency is also seen in 25.0 cm pitch lattice, though lesser in quantity, which suggests that the experiment in the ρ^{235} measurement bears some systematic error.

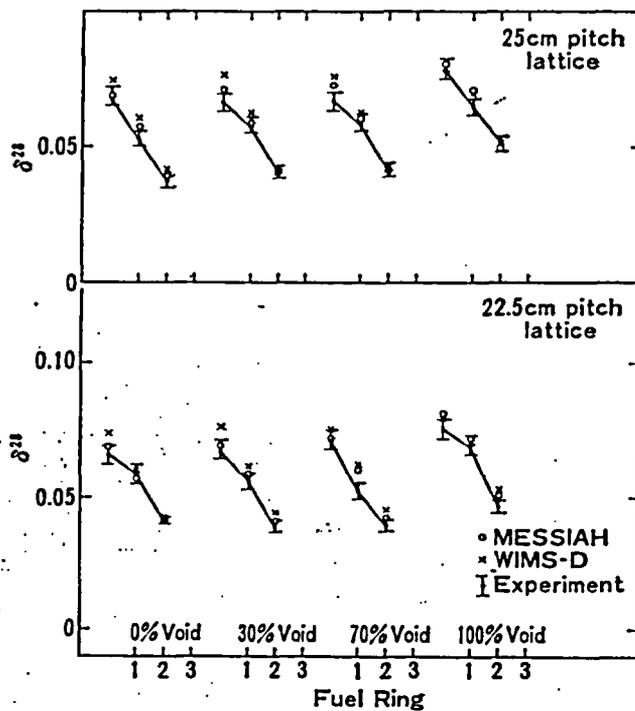
2. δ^{235}

The micro-parameters δ^{235} are shown in Fig. 7(a) with the experiment. The agreement of the δ^{235} calculated by MESSIAH with the experiment is satisfactory, which indicates that the behavior of fast neutrons is properly treated in MESSIAH.

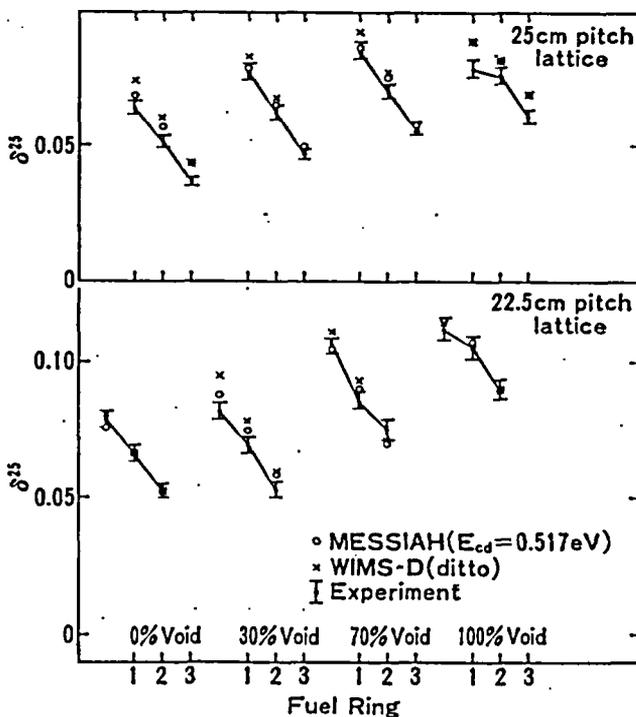
3. δ^{238}

In Fig. 7(b) the comparison of δ^{238} with the experiment is shown.

The agreement of δ^{238} with the experiment is fair, though we can see some general tendency that the calculated δ^{238} is higher than the experiment. Originally, the δ^{238} by MESSIAH as well as by the present reference calculation code WIMS-D was calculated for Cd cut-off energy $E_{cd}=0.45\text{ eV}$, since this cut-off energy was adopted in the previous calculation using METHUSELAH-II⁽⁵⁾ for comparison with the same



(a) δ^{235}



(b) δ^{238}

Fig. 7(a),(b) Comparison of δ^{235} and δ^{238} with experiment (22.5 and 25.0 cm pitch lattices)

experimental data⁽⁴⁾. However, systematic deviation (over prediction) was observed in the detailed calculation by both MESSIAH and WIMS-D. Therefore, we scrutinized the dependence of the values δ^{25} on Cd cut-off energy as shown in Fig. 8 by changing $E_{Cd}=0.45, 0.5$ and 0.625 eV, artificially. As Fig. 8 shows that the δ^{25} is strongly dependent on E_{Cd} , we obtained δ^{25} for $E_{Cd}=0.517$ eV by the interpolation for comparison with the experiment. Here, we selected $E_{Cd}=0.517$ eV for ^{235}U activation with the Cd cover of 0.5 mm thickness⁽²⁵⁾.

The accuracy of δ^{25} is strictly determined by the correctness of E_{Cd} . Therefore, a detailed analysis of E_{Cd} will clarify the ambiguity that still exists in the present comparison of the experiment and the analysis.

4. Thermal Neutron Distribution

The activation traverse of Dy-foil is compared with the experiment for two typical cases in Figs. 9(a) and (b). The analytical results are normalized to the experiment at the outside of a calandria tube where the unit cell is divided into moderator and other regions. The activity traverse is measured along the line to the nearest neighbor unit cell (0° direction) and the next nearest neighbor unit cell (45° direction) in the square lattice.

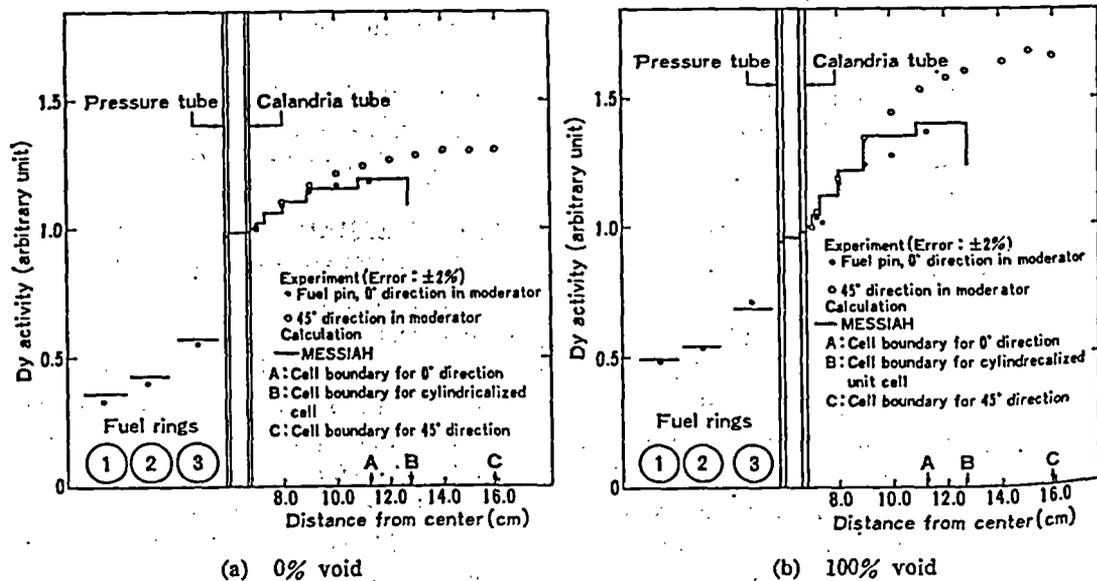


Fig. 9(a),(b) Comparison of sub-Cd activity of Dy-foil with experiment (22.5 cm pitch lattice, 0 and 100% void)

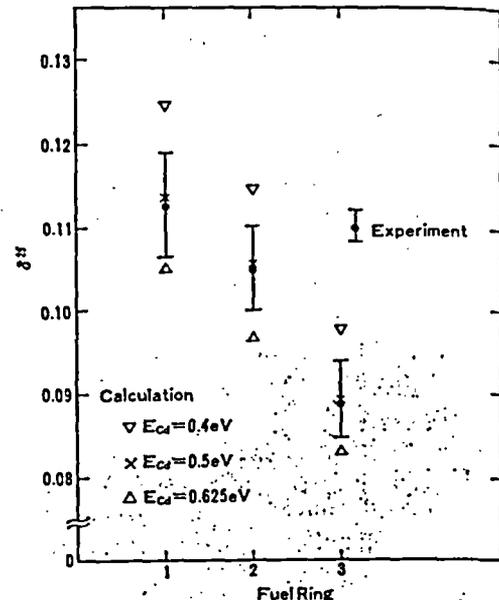


Fig. 8 Dependence of δ^{25} on Cd cut-off energy calculated for 22.5 cm pitch lattice, 100% void core with WIMS-D

The agreement between the experiment and the analysis is fair for the 0 and 100% void cores of 22.5 cm lattice pitch. But the flux depression within a pressure tube is smaller than the experiment. This tendency is seen for the sub-Cd activation distribution measured by other foils like ²³⁵U. This is probably caused by the mesh volume which is a little too coarse for the coolant regions and fuel pins. Especially this is caused by the space meshes for the fuel pins of which division into semi-circle regions was made only at the pin center by using coaxial circle lines having its center at the cluster center. However, this has not been ascertained by the calculation.

5. Criticality

The eigenvalue k_{eff} was calculated for a given critical height with CITATION code⁽²⁾, in the approximation of the diffusion theory, R-Z geometry and 4 group model. The calculated k_{eff} which should be unity, is shown in Table 2.

Table 2 Calculated k_{eff} for experimental critical water level

Lattice pitch (cm)	Void (%)	Experimental critical water level (cm)	Effective multiplication factor	
			MESSIAH	WIMS-D
22.5	0	96.41	0.99979	0.99860
	30	96.85	0.99769	0.99779
	70	102.02	0.99366	1.00190
	100	111.48	0.99260	1.00784
25.0	0	105.42	1.00124	1.00053
	30	102.58	0.99917	0.99864
	70	102.21	0.99634	1.00414
	100	104.81	0.99712	1.01146

The difference from unity of the calculated k_{eff} is very small for 0% void lattice of both 22.5 and 25.0 cm pitches. However, as void percentage becomes larger, this difference becomes larger. The maximum difference of ($k_{eff}-1$) is 0.0074 for 22.5 cm pitch lattice at 100% void and 0.00366 for 25.0 cm pitch lattice at 70% void. Hitherto, the calculated k_{eff} for 100% void core were greater than the experiment by about 1%⁽³⁾. This tendency is also seen for WIMS-D results.

The reason for this tendency was ascribed to the under prediction of ρ^{28} for 100% void core⁽⁴⁾. The small ρ^{28} means the small resonance absorption by ²³⁸U and this results in the over prediction of k_{eff} . This interpretation is applicable for the present WIMS-D results (see Fig. 5).

The calculated ρ^{28} is much improved in the MESSIAH compared with the one with WIMS-D. However, the calculated k_{eff} with the MESSIAH group constants becomes lower than unity as the coolant void increases. This indicates that the interpretation of the former analysis is inadequate for the present case.

Let us assume here that the reactor physics parameters in a unit cell are calculated accurately. This is equivalent to assuming that the reaction cross sections for a homogenized cell are calculated properly. And this assumption is supported for the DCA core by the comparison of ρ^{28} , δ^{28} and δ^{25} . Therefore, the only possible cause for the under prediction of k_{eff} is an error in the diffusion constant. This is likely because the deviation of k_{eff} from unity increases towards 100% void core, which is the same indication that the leakage from the core becomes more important.

The diffusion coefficient is calculated by the Benoist's method in MESSIAH. In this method the dependence of the diffusion coefficients on the direction is treated in principle.

However, the present diffusion calculation is performed with use of the isotropic diffusion coefficients, resulting in some possible error. To evaluate the effect of anisotropic leakage in the diffusion approximation, we compared k_{eff} for 22.5 cm pitch lattice, 100% void core with the isotropic and anisotropic diffusion coefficient. The difference of k_{eff} in both approximations was found less than $\sim 0.2\%$, which cannot explain the present disagreement. Since the anisotropy in the diffusion coefficients is greatly enhanced in the case of 100% void core, this difference must be reduced for other smaller void cores. Therefore, we can attribute the disagreement of the reactivity to the error of diffusion coefficient itself, if the above discussion is applicable to the present comparison.

6. Void Reactivity

The calculated void reactivity in dollar unit is compared with the experiment in Fig. 10, and is defined as

$$\rho(v\% \text{ void}) = \left(\frac{1}{k_{eff}^v} - \frac{1}{k_{eff}^0} \right) / \beta_{eff} \quad (6)$$

where k_{eff}^v is k_{eff} calculated by the group constants of $v\%$ void core for the critical water level of 0% void core, and β_{eff} is the effective delayed neutron fraction, 0.72%⁽²⁰⁾.

The void reactivity calculated by MESSIAH is closer to the experiment than that by WIMS-D. The over prediction of the reactivity by WIMS-D corresponds to the fact that k_{eff} by WIMS-D is greater than unity for larger void core, for k_{eff} of 0% void cores is almost unity for both MESSIAH and WIMS-D as shown in Table 2. The void reactivity by MESSIAH was calculated slightly smaller (more negative) than the experiment, which corresponds to the fact that k_{eff} by MESSIAH is smaller than unity for larger void cores.

V. CONFIRMATION OF CALCULATIONAL ADEQUACY BY MONTE CARLO METHOD

It is recognized that the effect of the anisotropic neutron leakage from the DCA core is not sufficiently estimated by MESSIAH because it treats this effect in the diffusion approximation. We performed a Monte Carlo calculation with KENO-IV to establish the validity of the method used in the MESSIAH code system by checking this leakage effect.

Although the basic method used to solve the neutron transport equation in KENO is completely different from that in MESSIAH, we utilized a 4 group cross section set calculated by MESSIAH. However, we can assume that Monte Carlo approach can evaluate exactly neutron leakage as far as geometrical effect are concerned such as the core shape. The calculation was performed for a unit cell with the reflective boundary condition for the radial boundary, while the vacuum boundary condition was applied to the axial end

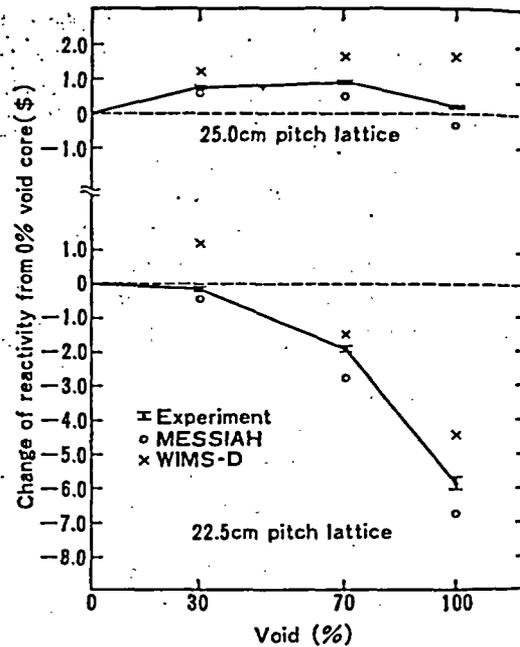


Fig. 10 Comparison of void reactivity with experiment

points. The radial neutron leakage was estimated by adding DB_0^2 to the absorption cross section, where B_0^2 is the radial buckling of the DCA core ($2.4 \times 10^{-4} \text{ cm}^{-2}$). The calculation was done within the P_0 (isotropic scattering) approximation. The bare fuel part and the upper and lower structure of the core were neglected for simplicity. The length of the fuel rods was changed to simulate the variation of the axial neutron leakage caused by the critical water levels.

The neutron flux distribution in the central region of the 20 cm length was monitored by the track length estimators and the flux was converted into the form of ρ^{28} . The results are compared with the MESSIAH calculation in Fig. 11. The values for ρ^{28} are independent of the change of axial leakage and agreed well with each other. The calculated ρ^{28} shows the tendency to be greater than that of the MESSIAH calculation, especially for inner rings. This is probably due to the limited number of neutrons tracked in the Monte Carlo calculations.

To estimate the leakage treatment in MESSIAH, the simulated calculation was done for the core with no leakage but DB_0^2 loss, where B_0^2 is the total geometrical buckling. The results has again agreed well with that for the axial neutron leakage core. Therefore, the treatment of leakage of neutrons in MESSIAH is considered to be justified in the worst case of the DCA core of 22.5 cm pitch lattice and 100% void by the independent procedure, in which the leakage effect is expected to be most enhanced.

VI. CONCLUSION

The MESSIAH code system that utilizes the collision probability method extensively was developed and used to analyze the reactor physics properties of a pressure-tube-type, heavy-water reactor, measured in the DCA core.

Almost all of the experimental data are predicted successfully by this code system. The discrepancies we encountered in the analysis are, (1) ρ^{28} in the 100% void lattice, and (2) under prediction of k_{eff} for larger void cores.

The ρ^{28} anomaly was not explained and the discrepancy with the analysis still exists in the present analysis. However, we proposed an interpretation of this disagreement, that this was caused by the experimental condition which is characteristic to the DCA core, (see APPENDIX also).

We established the adequacy of the diffusion approximation to treat neutron leakage in MESSIAH through a Monte Carlo calculation.

Concerning the deviation of k_{eff} from unity, we concluded that the diffusion coefficient

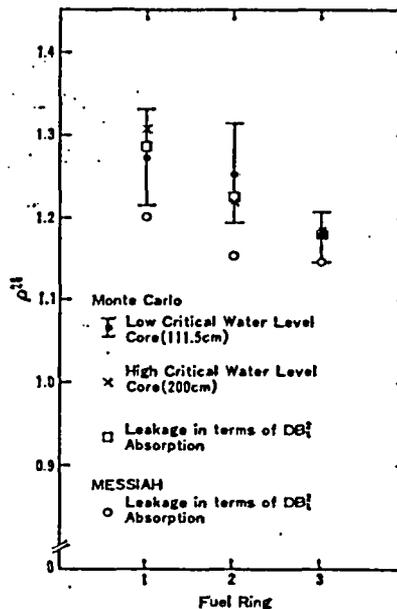


Fig. 11 Monte Carlo calculation of ρ^{28} dependence on change of neutron leakage from lattice (22.5 cm pitch lattice, 100% void, statistical errors for \times and \square are smaller than indicated one for \circ)

was the possible error. Some further study seems necessary to improve the accuracy of the diffusion coefficients in a system with strong leakage and heterogeneity.

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[APPENDIX]

Re-measurement of ρ^{28} in DCA Core with High Critical Water Level

In order to check the effect of the anisotropic neutron leakage, the reactor physics parameters were measured in a new core whose shape was altered from the original one by making the critical water level much higher. That is, the number of loaded fuel assemblies was reduced to increase radial leakage of neutrons. The reactor physics parameters were measured for this altered-high core of the same 22.5 cm lattice pitch. However, every parameter agreed well with the one which was measured in the core with the original low critical water level except for ρ^{28} of 100% void core (see Fig. A1)⁽³¹⁾.

The anomaly observed in Fig. 5 disappeared and the agreement with the MESSIAH calculation has been greatly improved. By this phenomenon, however, it cannot necessarily be understood that the axial neutron leakage directly influenced the distribution of ρ^{28} in the 100% void core with low critical water level because the anomaly of ρ^{28} observed in the core might be caused accidentally by the experimental conditions.

The angular dependence of neutron flux in the pressure-tube-type core at low critical water level condition, like the DCA core, is strongly peaked in the axial direction of the tubes. And this peaking of the angular flux will be increased towards the upper or lower end of the DCA core which has no axial reflector. This effect is, from the view point of void in a unit cell, stressed at the outside of a third layer of fuel rings in the 100% void lattice. The experimental configuration of ρ^{28} measurement, which is shown in Fig. A2 (a)⁽³¹⁾, shows that a small gap exists along the cladding in the Cd-ring; The Cd-box consists of a ring for the circumferential cover and two disks for the cross-sectional cover. The leakage of thermal neutrons through this gap is increased greatly when we use UO_2 -buttons. Because the solid angle of the gap spanned by the Cd-ring and disk is effectively increased, when it is seen from the foil side. As the peaking of neutron flux coincides with the direction of this gap towards the end of the core, the Cd covered activity is increased appreciably by the leaking thermal neutrons.

Since the existence of UO_2 -buttons affects the measured δ^{28} by the order of 10%, this effect was corrected with use of a correction factor measured in the DCA core. The equivalent correction factor was not measured for ρ^{28} because the factor was found small for almost the same lattice performed with the SUMITOMO critical facility⁽³²⁾. However, this circumstance may not be applicable to the DCA core for the following reasons:

- (1) The SUMITOMO core has a thick axial reflector. Therefore, the angular flux along the axial direction does not have a peak even in the vicinity of the third ring.
- (2) As it is shown in Fig. A2 (b)⁽³²⁾, the structure of the Cd-box, consisting of a vessel (a ring and a disk in one body) and a lid (disk), is different from the one used

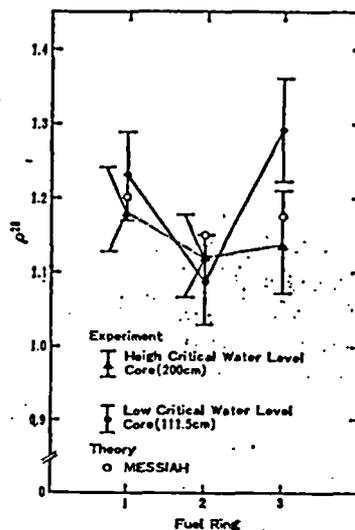


Fig. A1 Measured ρ^{28} dependence on change of neutron leakage from lattice (22.5 cm pitch lattice, 100% void core)

in the DCA experiment. By this Cd-box, irradiated foils were completely shielded from thermal neutron leakage.

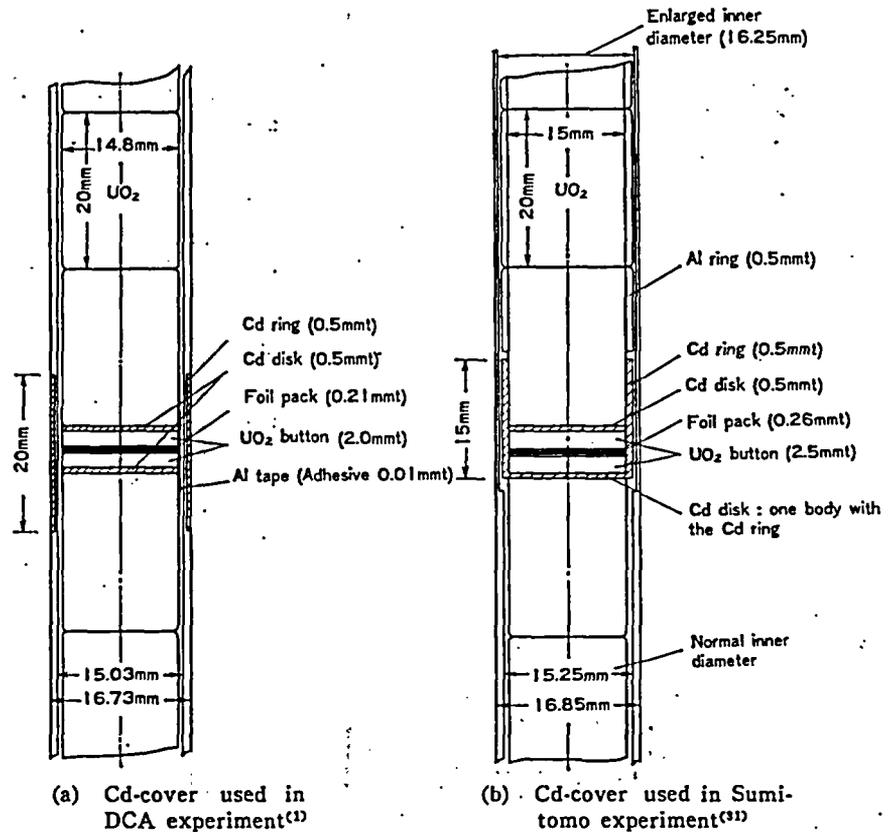


Fig. A2 Difference in shape of Cd-boxes

Consequently the change in the value of ρ^{28} at the third ring between the high and low cores does not necessarily mean a change in the neutron spectrum between the two kinds of core shapes.

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