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MEASUREMENT OF VOID REACTIVITY

September, 1972

Power Reactor and Nuclear Fuel
Development Corporation, Tokyo, Japan

Measurement of Void Reactivity

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A series of measurements on void reactivity was performed by critical and pulsed neutron methods for the lattice of 1.2% enriched UO_2 , 28 rod cluster, 22.5 cm pitch, at Deutrium Critical Assembly (DCA) in O-arai engineering center of PNC.

Experimental results were analyzed using NOAH-2 (METHUSELAH-2) and group diffusion codes. The experimental values agree with the calculations within 0.5 dollars.

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1. Introduction

The Japanese advanced thermal reactor "FUGEN" is a heavy water moderated, boiling light water cooling, pressure tube type. From the standpoint of nuclear safety it is essential for this kind of a reactor to estimate the coolant void reactivity, especially the effect of the loss of coolant accident on a reactor.

For this purpose a series of measurements on void reactivity were performed in Deuterium Critical Assembly (DCA) by changing the void ratio of coolant water. It is intended to establish the method of void reactivity calculation by analysing experimental data.

Measurements were performed on the following cores: (1) the clean core which has a uniform void ratio, (2) partially voided core to simulate the loss of coolant accident and (3) partially voided core with inserted solid absorbers. Void reactivity was measured by the use of the critical and pulsed neutron methods. In the critical method, void reactivity was obtained from data of difference of critical heights of moderator between the void and standard cores, and of the reactivity coefficient of moderator level determined by the positive period method.

Pulsed neutron technique was also applied to the clean cores having a uniform void ratio. Void reactivity was obtained by measuring the prompt neutron decay constant of sub-critical system as a function of the axial buckling for each void ratio. The result obtained by the pulsed neutron method agreed with data by the critical method within experimental error except in the case of heavy water used as coolant.

Analysis on the experimental results was carried out in a standard pattern for reactor design. Group constants were obtained from NOAH-2 (METHUSELAH-2) and the few group diffusion calculation was performed.

By comparison of calculated void reactivity with the experimental one, calculational models were checked. A new input model for NOAH-2 was proposed in the present paper. That is, by the correct choice of thermal neutron spectrum which is used to average the thermal group constant dependently on void ratio, material bucklings were adjusted to agree with experimental data. The void reactivity, however, was over-estimated by the present input model to NOAH-2 code.

In the following sections experiment and analysis will be described.

2. Description of DCA

Reactor vessel of DCA is cylindrical, 3 m in diameter and 3.5 m in height, and is made of aluminum. The number of fuel channel is 121 in the case of 22.5 cm square pitch (Fig. 1). Each fuel channel is composed of a calandria tube, a pressure tube and a cluster fuel element (Fig. 2). The calandria and the pressure tubes are both cylindrical, and their dimensions are 13.65 cm and 11.68 cm in outer diameter respectively, and both 2 mm in thickness. Gap between the calandria and the pressure tube is filled with air. The cluster fuel element consists of 28 rods having an outer diameter of 16.65 mm. The rods are arranged in three concentric layers as shown in Fig. 2. The fuel consists of 1.2 % enriched UO_2 pellets, 14.8 mm in diameter and is clad with aluminum tube. Moderator is heavy water (D_2O) of 99.50 mol/o.

The "coolants" used in the experiment were H_2O (0% void), air (100% void), mixture solution (H_2O , D_2O and H_3BO_4) and D_2O . The mixture solutions were prepared to simulate neutron cross sections of actually voided coolant (Table 1). The thermal absorption cross section (Σ_a 2200) and the slowing-down power ($\xi \Sigma_s$) for each mixture solution are nearly equal to those of actually voided coolant of 30% and 70% respectively. But $\xi \Sigma_s$ on D_2O corresponds to 86.7% void coolant. Comparison of nuclear constants between the simulated and the actual void coolant is made in Table 2.

In Fig. 3, arrangement of DCA is shown.

3. Experimental Method

3.1. Core having uniform void ratio

3.1.1. Critical method

Critical moderator height (H_0) for the clean core which uses light water as the coolant (0% void core) and critical moderator height (H_v) for the coolant of V% void ratio (V% void core) were measured experimentally. The difference in these critical heights is reduced to reactivity. To invert the amount of the critical height to reactivity, the reactivity coefficient of moderator level for V% void core ($(\frac{\partial \rho}{\partial h})_v$) is used. Then, void reactivity (ρ_v) which corresponds to the change of 0% to V% coolant void ratio is expressed as

$$\rho_v = \int_{H_0}^{H_v} \left(\frac{\partial \rho}{\partial h} \right)_v dh. \quad (\#) \quad (1)$$

The reactivity coefficient of moderator level as a function of moderator height was measured by the positive period method at various critical moderator heights by inserting solid absorbers into the core. The results were fitted by the least square method to

$$\left(\frac{\partial \rho}{\partial h} \right) = \frac{C}{(h + \delta_z)^3} \quad (\$/cm) \quad (2)$$

and the constant C was determined. Where h is the height of moderator, δ_z is the sum of the upper and lower extrapolated lengths along the axial direction. The extrapolated length δ_z was determined from the neutron flux distribution along the axis measured by the activation of Cu wire.

The reactivity measured by the positive period method may be influenced by the photo-delayed neutron of (γ, n) reaction in heavy water moderator. This effect depends on the core history. To eliminate this effect each measurement was carried out at two-hour intervals, and in addition measurement of the positive period was not started until the reactor power levels up two decades.

The reactivity coefficient of moderator level measured at the core with solid absorber ($(\frac{\partial \rho}{\partial h})^a$) was reduced to the coefficient for the clean core ($(\frac{\partial \rho}{\partial h})^c$) by using the equation based on modified one group theory,

$$\left(\frac{\partial \rho}{\partial h} \right)^c = \left(\frac{\partial \rho}{\partial h} \right)^a \left\{ 1 + \frac{C}{2} \left(\frac{1}{(h_c + \delta_z)^2} - \frac{1}{(h_a + \delta_z)^2} \right) \right\} \quad (3)$$

where h_c and h_a are critical heights for the clean core and that with absorbers respectively.

In the measurement of critical moderator height, the difference between the moderator and the coolant levels should be accounted. This difference may become an error in its measurement. Therefore the measured data of critical height were corrected to such critical heights that there were no differences between the moderator and coolant. For this purpose another experiment was made on reactivity effect caused by additional coolant level

3.1.2. Pulsed neutron method

The decay constants (α) of 14 MeV neutrons generated by $D(T, \alpha)n$ reaction for sub-critical cores were measured by changing the moderator height. The decay constants were fitted by the least square method to the equation

$$\alpha = \alpha_c + A (B_z^2 - B_{zc}^2) + B (B_z^2 - B_{zc}^2)^2 \quad (4)$$

where B_z^2 and B_{zc}^2 were the axial buckling for the sub-critical and critical core respectively. The constants α_c , A and B were determined for each void ratio core. The constant α_c is the decay constant for the critical core.

The decay constant (α) of the sub-critical core of $V\%$ void ratio having the reactivity ρ is expressed as follow;

$$\alpha = \frac{k_{eff} \beta_{eff}}{\ell} (1 - \rho) \quad (5)$$

where k_{eff} is the effective multiplication factor and β_{eff} is the effective delayed neutron fraction and ℓ is the prompt neutron life time for $V\%$ void core.

Equation (5) can be written

$$\rho = 1 - \frac{\alpha}{\alpha_c} \quad (6)$$

by making the approximation

$$\frac{k_{eff} \cdot \beta_{eff}}{\ell} = \frac{\beta_{eff}}{\ell} = \alpha_c \quad (7)$$

When coolant void ratio changes from 0% to V%, the corresponding reactivity change is estimated by

$$\rho_V = 1 - \frac{\alpha(V, H_0)}{\alpha_c(V, H_V)} \quad (8)$$

where $\alpha(V, H_0)$ is the decay constant of V% core having the moderator height H_0 and determined from Eq. (4), and $\alpha_c(V, H_V)$ is the decay constant of V% void critical core.

The pulsed neutron was generated by a 200 Kv Cockcroft-Walton type accelerator. The pulse width was set to 1.1 msec. Value of repetition rate was chosen appropriately out of the range between 10 Hz. to 0.005 Hz. depending on sub-criticality of the experimental core to minimize the build-up of delayed neutrons. In the measurement BF_3 counter was used as a neutron detector.

3.2. Partially voided core

To study the reactivity effect of the loss of coolant in "FUGEN" which has two independent coolant loops, the reactivity effect of DCA core was measured by replacing the coolant with air in the part of one, two or three quarters of the DCA core (Fig. 4). Due to such asymmetric core configuration, it is difficult to measure the extrapolation length (δ_z) by the flux distribution. Therefore C and δ_z in Eq. (2) were obtained parametrically by the least square fitting.

In addition to the above experiments a series of experiments were performed on the core of all 100% void channel, and on the core with solid absorbers inserted (Fig. 4). The solid absorbers used have the same geometry as the control rod which is to be used for "FUGEN" and is shown in Fig. 5. Four or nine absorbers were inserted into the heavy water region surrounded by four fuel channels (Fig. 4).

4. Experimental Result

4.1. Core having the uniform void ratio

Experimental results of the void reactivity are shown in Table 3. In this table results obtained both from the critical and the pulsed neutron methods are shown. These results agree well between the two methods within the experimental error except 86.7% void core which uses heavy water as coolant.

Experimental errors of the critical method are:

- (1) Error of reactivity coefficient of moderator level,
- (2) Error of critical height of moderator.

The main part of the error due to reactivity coefficient comes from the error in detection of small change of the moderator level. The value of this level change was 4~5 mm in the present experiment, and this change was measured with an accuracy of 0.1 mm, therefore the error was 2~2.5%. Another error is due to the measurement of the period, and it was estimated as 0.5%. Then the total error of reactivity coefficient was 2.5~3%.

On the other hand experimental error in the measurement of critical moderator height is caused by change of temperature during the experiment and by accuracy of level meter. All the experiments were performed at room temperature ($\sim 20^{\circ}\text{C}$), and indication of the level meter was calibrated up to 1 mm in accuracy. Then the error in the critical height was about 0.1%, and in fact the measured data of critical height were reproduced within 0.1%.

The measured value of critical moderator height is affected by the difference between the moderator and coolant level. The true critical moderator height was such that the moderator level was equal to the coolant level, and was obtained to correct the measured critical moderator height by measuring the value of the above effect of level difference. This correction was at most about 1%, and the error in this correction factor for the level difference was neglected. While the influence on the reactivity coefficient of moderator level due to the above level difference was neglected, since every measurement of reactivity coefficient was performed at the following conditions: the coolant level was higher than the moderator level, but the difference between these two levels was limited within 10 cm.

To obtain the void reactivity from the reactivity coefficient of moderator level using Eq. (1), the reactivity coefficient was measured as a function of the moderator height. These results were fitted to Eq. (2) to find the constant C by the least square method. The error of this constant becomes maximum 5% in the case of 86.7% void core, and 2~3% in other void cores.

Since the void reactivity in DCA was negative, the functional formula of reactivity coefficient, Eq. (2), which was determined based on data of $\frac{\partial \rho}{\partial h}$ measured at various moderator heights above H_v , was integrated in a range below H_v . Then, there is a limitation for the lower boundary in this integration to obtain a negative reactivity within an allowable error, because the value of $\frac{\partial \rho}{\partial h}$ at the much lower moderator height than H_v deviates from the value estimated by using the experimental formula of reactivity coefficient. According to the theoretical calculation this deviation was small in the present experiment, and was neglected.

Experimental errors of the pulsed neutron technique are:

- (1) Error of the prompt neutron decay constant,
- (2) Error of the axial buckling.

The error of the prompt neutron decay constant is due to the effect of higher harmonics, the statistical error of counting and build up of photo-delayed neutron. To obtain the decay constant, the measured neutron density (N) as a function of time was fitted to

$$N = N_0 \exp(-\alpha t) \quad (9)$$

Then the effect of higher harmonics on decay constant was eliminated and it was found that the decay constant became constant within 2% by changing the starting time of fitting. The statistical error of counting was kept within 0.1%. On the other hand the effect of build up of photo-delayed neutron could be neglected by selecting the repetition rate of the pulsed neutron source. Therefore maximum the error of the decay constant was 2%.

The decay constant of the critical core (α_c) was evaluated with the use of measured decay constants and axial bucklings by fitting them to Eq. (4). The fitting error of α_c was estimated as 1 to 4%, and the worst value of 4% corresponded to the case of 30% void core.

The axial buckling of the critical core B_{zC}^2 was obtained by measuring the neutron flux distribution with Cu wire. The measurements have been performed both in the center of the fuel channel and in the heavy water moderator. However, between data of these two measurements, different bucklings can be obtained by the least square fitting of the function

$$\phi_z = \phi_z^0 \cos B_{zC} (Z - C) \quad (10)$$

where ϕ_z is the neutron flux distribution. Therefore the fitted value ϕ_z^0 , B_{zC} and C evaluated at different positions are weighted by the following procedure,

$$B_{zC}^2 = \frac{\sum_i \int_{V_i} \frac{\partial^2 \phi(Z)}{\partial Z^2} \Delta_i dZ}{\sum_i \int_{V_i} \phi(Z) \Delta_i dZ}$$

$$= \frac{\sum_i \int_{-H/2}^{H/2} \phi_{zi}^0 B_{zci}^2 \Delta_i \cos B_{zci} (Z - C_i) dZ}{\sum_i \int_{-H/2}^{H/2} \phi_{zi}^0 \Delta_i \cos B_{zci} (Z - C_i) dZ} \quad (11)$$

where B_{zci}^2 is the critical buckling, i is the index for fuel channel or moderator region, Δ_i is the area of region i and H is the region of the least square fitting. The error of buckling evaluated by the above method was 2.3% for 0% void core and less than 2% for other void cores. The error of the pulsed neutron method in Table 3 contains all of the above errors.

In the pulsed neutron method it is necessary to make some corrections for increase of higher harmonics effect and for the approximation in Eq. (7), as the sub-criticality of core becomes large. Therefore the pulsed neutron technique is not appropriate to measure relatively large sub-critical reactivity. Generally speaking the maximum sub-critical reactivity which can be measured by the pulsed neutron technique may be -10\$. Since the maximum negative reactivity measured in the present experiment is -7.6\$, it is feasible to believe that the present experimental data is reliable.

4.2. Partially voided core

In Table 4, the reactivity effects measured by the critical method are shown. In these cases coolant of fuel channels was changed to air

(100%) in the part of one, two or three quarters of the DCA core. In addition to the partially voided core the results are shown for cases when the change to 100% void is made for full channels in DCA core. In Fig. 6 the relation between void reactivity and number of voided channels is plotted.

The results on the cores which contain solid absorbers are listed in Table 4 also. The relation between void reactivity and number of channels is not different obviously from the results on clean core (Fig. 6) Moreover the amount of void reactivity for the core with solid absorbers is not changed, when compared with that of the clean core, though geometry of the critical core becomes larger. It seems that decrease of negative void reactivity due to increase of core size is suppressed by increase of negative reactivity worth (1.5 times) of absorbers.

5. Method of Analysis

Analysis on the experiment was performed by adopting the traditional method. Four group constants were prepared by NOAH-2, and diffusion calculation was performed by FOG-3 and PDQ-5 codes.

Void reactivity was obtained by performing critical calculation by FOG-3 and PDQ-5 codes with group constants of each void core calculated by NOAH-2 code. Void reactivity is defined as

$$\begin{aligned} \rho_v &= \frac{1}{\beta_{\text{eff}}} \left(\frac{k_{\text{eff}}^v - 1}{k_{\text{eff}}^v} - \frac{k_{\text{eff}}^0 - 1}{k_{\text{eff}}^0} \right) \\ &= \frac{k_{\text{eff}}^0 - k_{\text{eff}}^v}{\beta_{\text{eff}} k_{\text{eff}}^v \cdot k_{\text{eff}}^0}, \quad (\#) \end{aligned} \quad (12)$$

where k_{eff}^v and k_{eff}^0 are eigen values of V% and 0% void core respectively, and β_{eff} is effective delayed neutron fraction.

5.1. Core having uniform void ratio

Though calculational models adopted in NOAH-2 code are established, an input model is searched to reproduce experimental data. Group constants of cores having different void ratio were checked by the use of experimental material bucklings, since this parameter which directly determines reactivity can be regarded as one of the essential parameter in reactor physics.

The input model for NOAH-2 is determined as follows. The spectrum which is used to average the thermal group constant is chosen as a parameter and optimized to give the best calculated axial buckling. Two kinds of spectra are prepared: one is the hard spectrum which is used in the fuel region, the other is the soft spectrum which is used in the moderator region. We succeeded to give the best axial buckling by changing the region of assignment of spectra.

The fundamental idea of the present model is that in 0% void core neutrons are effectively thermalized by water inside of the pressure tube and on the other hand, 100% void core, thermalization of neutron is not enough even in the moderator close to the calandria tube. Therefore we assumed from the above physical consideration that in the case of 0% void core the soft spectrum emerges inside of the pressure tube and in 100% void core the hard spectrum emerges outside of the calandria tube.

In Fig. 7 the parametric search of 0% void core is shown. If we choose the boundary of the soft and hard spectrum at 1.4 mm inside of the pressure tube, the best fit for calculation and experiment for the axial critical buckling is obtained. In the case of 100% void core the agreement between calculation and experiment improved when we draw the boundary line of the soft and hard spectrum at 15 mm outside of the calandria tube, though it is hard to predict the experimental value of the axial buckling with good agreement by using NOAH-2 and FOG-3 for 100% void core. In the case of 70% void core good agreement was obtained by choosing the boundary at the surface of the calandria tube. The input model to perform systematic calculation for other void cores is shown in Fig. 8.

The axial bucklings obtained by the buckling search option of FOG-3 code with the use of group constants prepared by the above mentioned input model are compared with experiment in Table 5. Agreement between calculation and experiment can be said to be appreciably good. But calculated values for 30% and 86.7% void cores differ slightly from experimental values; the main reason seems that these simulated void fractions are regarded as actual void fractions in the calculation.

Eigen value for each void core in Eq. (12) was obtained by the calculation of FOG-3 code with the use of four group constants prepared by the above mentioned input model for NOAH-2. In this calculation, eigen value of V% void core, k_{eff}^V , was obtained for the core having the same moderator height of 0% void critical core. But this does not mean that the axial buckling of 0% void core is to be used for this calculation. The reason is that the extrapolation length depends on the void fraction of the coolant. If we neglect this effect, void reactivity calculated becomes large up to maximum 30%. Therefore the axial buckling to be used for the void reactivity calculation is estimated as follows,

$$B_{ZV}^2 = \left(\frac{\pi}{H_0 + \delta_Z} \right)^2 \quad (13)$$

$$\delta_Z = \frac{B_{ZV}^{\text{cal}}}{\pi} - H_0 \quad (14)$$

where B_{ZV}^2 is the axial buckling for void reactivity calculation, H_0 is the critical height of moderator of 0% void core, and $B_{ZV}^{\text{cal}2}$ is the calculated axial critical buckling.

Experimental values of void reactivity are measured in β unit, so it is necessary to evaluate β_{eff} in Eq. (12). For the calculation of β_{eff} PERT code is utilized by using the direct and adjoint four group neutron fluxes obtained from FOG-3. The values of β_{eff} due to U^{235} and U^{238} fission were calculated. In addition to these the contribution of photo-delayed neutron from (γ, n) reaction of heavy water was estimated to be 0.02% ⁽¹⁾. The total β_{eff} value was 0.72% and the dependence of β_{eff} on the change of void ratio was confirmed to be negligible by the calculation.

In Table 3 the comparison of void reactivity is shown. Though calculated values with the use of B_{2v}^2 generally overestimate the void reactivity agreement between theory and experiment can be said to be good.

5.2. Partially voided core

The four group constants of 0% and 100% void unit cell of clean core have been calculated by NOAH-2 code. And after the first and the second group constants for the fast neutron energy range was degenerated into one group, the three group constants for each cell were prepared. In the NOAH-2 calculation for group constants, the boundary between the soft and the hard spectrum regions was taken at the surface of the calandria tube for 0% void cell, and at 15 mm outside of the calandria tube for 100% void cell.

For the core with solid absorbers, the group constants were calculated by the following method. As the solid absorber was inserted into the center position of moderator region surrounding four fuel channels, the region composed of these four unit cells was taken as one cell. The group constants of this cell with a solid absorber were obtained by using the absorption area method proposed by Deutsch⁽²⁾. To obtain the quantity of neutron absorbed by the absorber, the following conditions are assumed: (1) the fast neutron is not absorbed by the absorber, (2) the space distribution of slowing-down neutron from fast to epithermal energy is flat, and (3) the extent that is affected by the absorber is about a diffusion length of materials around the absorber.

Moreover the logarithmic derivative condition for neutron flux is used at the surface of the absorber. In this condition the linear extrapolation length was estimated by using the formula of Davison et al (3), (4) for black absorber, and Reyston's correction formula⁽⁵⁾ for gray absorber.

Using these group constants obtained the above mentioned procedure, and the critical calculation to analyse the experimental data was performed. This calculation was two dimensional (X,Y) diffusion calculation in PDQ-5 code. In this calculation, the axial buckling for 0% void core used the measured value of 0% void clean core. But in determining bucklings for both uniformly and partially voided cores of 100%, the measured value of the extrapolation length of 100% void clean core was used, because the bucklings for partially voided cores could not be measured.

The calculated results of the void reactivity for each experimental condition are shown in Table 4 being compared with the experimental data obtained by the critical method. On the uniform 100% void core, the calculated result is different from the previous one shown in Table 3. This is probably due to the use of the experimental value of extrapolation length.

6. Concluding Remarks

The two different experimental methods, the critical and the pulsed neutron method, have been applied to the measurement of the void reactivity in DCA core. The results obtained by both methods coincided with each other within the experimental error except in the case of heavy water used as coolant. The difference between the measured values obtained by the two methods becomes large for the core of large negative reactivity, and it is necessary to clarify this cause. In DCA core, the void reactivity is negative. This negative value may be due to the negative effect of the large neutron leakage in such a small core as DCA.

Based on the design code, NOAH-2, the experimental data have been analysed to decide the calculational model. It was confirmed that the experimental values could be reproduced well in the calculation by considering that the boundary between the soft and the hard spectrum regions in NOAH-2 code, and the extrapolation length vary depending on the coolant void ratio. But it is not sufficient to reproduce the experimental data of the critical buckling and the void reactivity systematically. Especially, it is insufficient for 100% void core, because the physical character of 100% void core differs qualitatively with other void cores in which the mixed solution or heavy water is used as coolant. Therefore other physical phenomena, such as anisotropy of diffusion coefficient, should be considered for 100% void core.

Table.1 Composition of Simulated Coolant

Void Fraction in Simulated Coolant (%)	Weight Percent (W%)		
	Heavy Water(D ₂ O)	Light Water(H ₂ O)	Boric Acid(H ₃ BO ₄)
30	36.817	63.174	0.0092
70	81.91	18.068	0.0215
86.7	99.55	0.45	0.

Table.2 Comparison of Neutron Cross Section

Ratio of Macroscopic Cross Section Void Fraction in Simulated Coolant (%)	30	70	86.7
$(\frac{\Sigma_s^S}{\Sigma_s^A})$ Thermal (2200 m/s)	1.0000	0.9998	0.0376
$(\frac{\Sigma_s^S}{\Sigma_s^A})$ Thermal (2200 m/s)	1.0008	1.0042	1.0112
$(\frac{\Sigma_{tr}^S}{\Sigma_{tr}^A})$ Thermal (2200 m/s)	1.0203	1.1103	1.3108
$(\frac{\xi \Sigma_s^S}{\xi \Sigma_s^A})$ Epi-Thermal	0.9998	0.9994	1.0091
$(\frac{\Sigma_{tr}^S}{\Sigma_{tr}^A})$ Epi-Thermal	1.1634	1.8896	2.3514

Σ^S : Cross Section of Simulated Coolant

Σ^A : Cross Section of Actually Voided Coolant

Table.3 Void Reactivity of Uniformly Voided Core

Change of Void Fraction in Coolant (%)	Experimental Value (\$) **		Calculated Value (\$) ***
	Critical Method	Pulse Neutron Method	
0 → 30	-0.382 ± 0.046	-0.38 ± 0.07	-0.535
0 → 70	-2.15 ± 0.07	-2.06 ± 0.11	-2.548
0 → 86.7	-3.08 ± 0.09	-2.79 ± 0.06	-2.954
0 → 100	-7.33 ± 0.16	-7.60 ± 0.24	-7.724

* The void fraction is The value of the simulated coolant

** The moderator (D₂O) height is 96.20 cm

*** $\beta_{eff} = 0.720 \%$

Table.4 Void Reactivity of Partially Voided Core
(Critical Method)

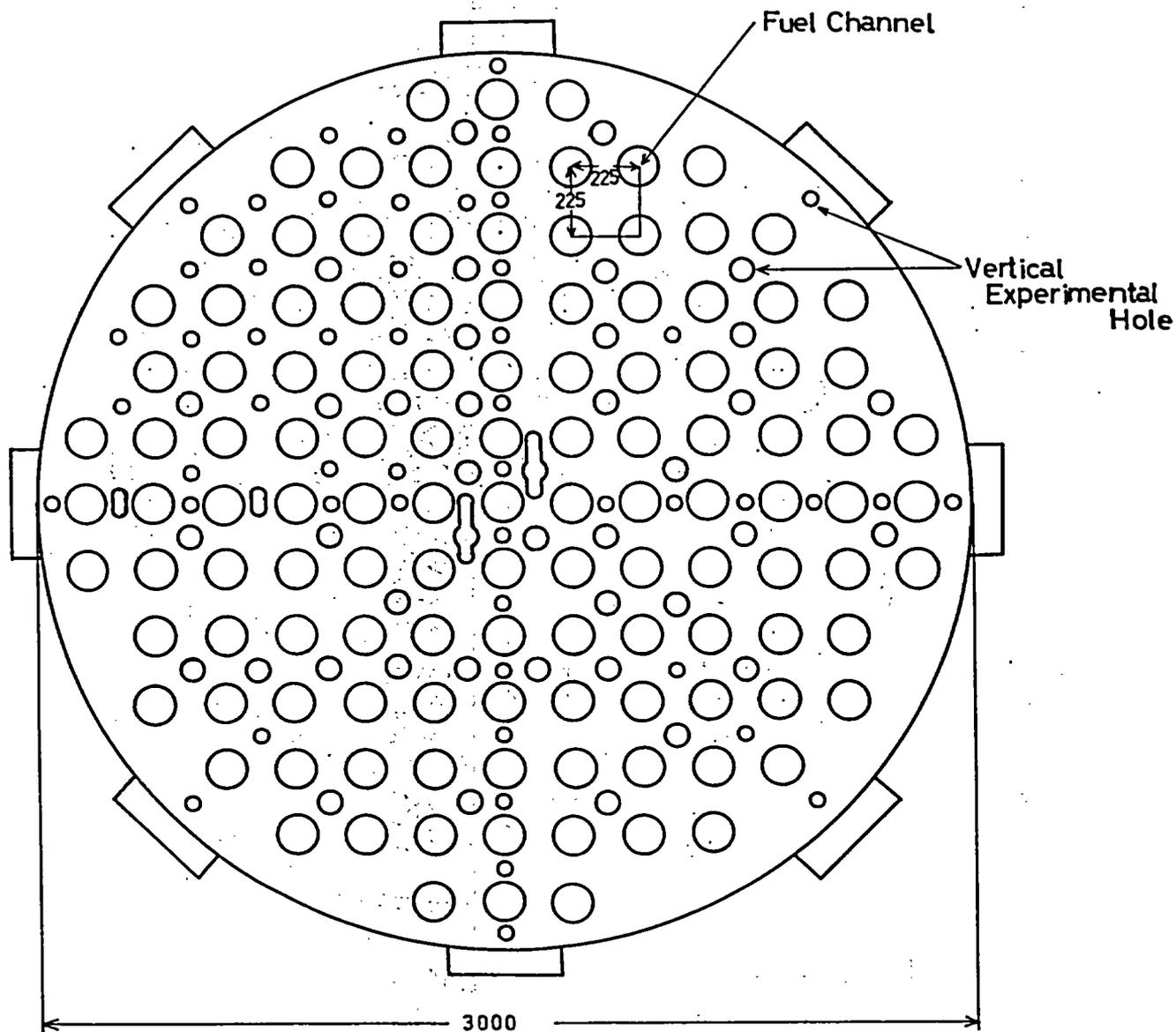
Number of Solid Absorber	Number of Void Channel	Void Reactivity (\$)*	
		Experimental Value	Calculated Value**
0	30 (1/4 Core)	- 0.93 ± 0.09	- 0.962
	60 (2/4 Core)	- 2.15 ± 0.10	- 2.11
	90 (3/4 Core)	- 3.88 ± 0.15	- 3.95
	121 (Full Core)	- 7.33 ± 0.16	- 7.26
4	30 (1/4 Core)	- 0.79 ± 0.06	- 0.868
	60 (2/4 Core)	- 1.86 ± 0.06	- 1.90
	90 (3/4 Core)	- 3.49 ± 0.12	- 3.58
	121 (Full Core)	- 6.57 ± 0.11	- 6.83
9	30 (1/4 Core)	- 0.78 ± 0.04	- 0.850
	60 (2/4 Core)	- 1.94 ± 0.06	- 2.09
	90 (3/4 Core)	- 3.65 ± 0.11	- 3.82
	121 (Full Core)	- 6.23 ± 0.08	- 6.30

* The coolant void changed from 0%(H₂O) to 100%(Air)

** $\beta_{eff} = 0.720 \%$

Table.5 Axial Critical Buckling

Void Fraction of Simulated Coolant(%)	Axial Buckling (cm ⁻²)	
	Experimental Value	Calculated Value
0	9.18 ± 0.21 × 10 ⁻⁴	9.18 × 10 ⁻⁴
30	8.75 ± 0.20 × 10 ⁻⁴	8.93 × 10 ⁻⁴
70	7.84 ± 0.17 × 10 ⁻⁴	7.84 × 10 ⁻⁴
86.7	7.85 ± 0.11 × 10 ⁻⁴	7.73 × 10 ⁻⁴
100	6.36 ± 0.12 × 10 ⁻⁴	6.65 × 10 ⁻⁴



Total Number of Fuel Channel : 121

Fig.1 DCA Core Configuration

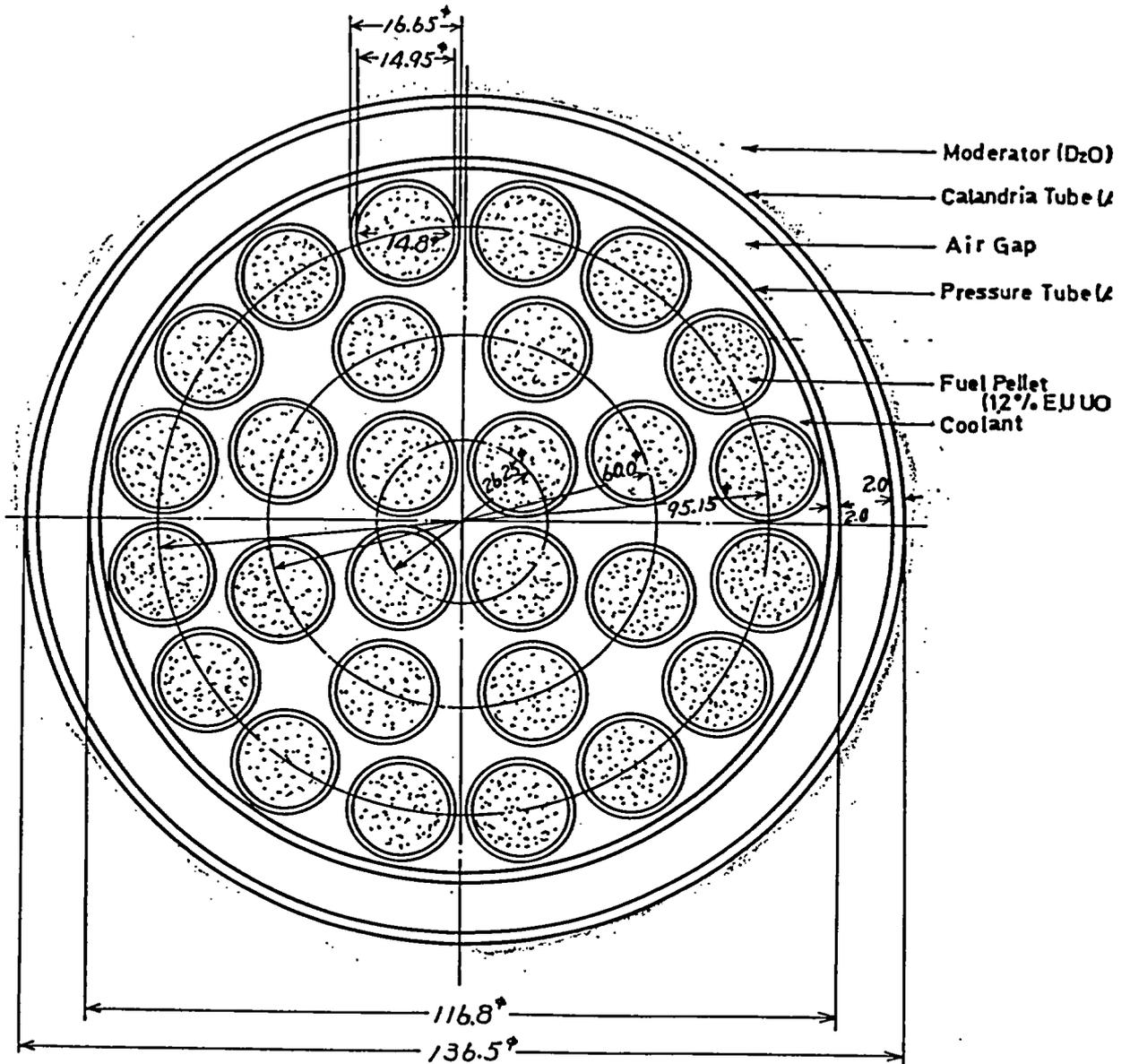


Fig.2 Cluster Fuel Channel of DCA

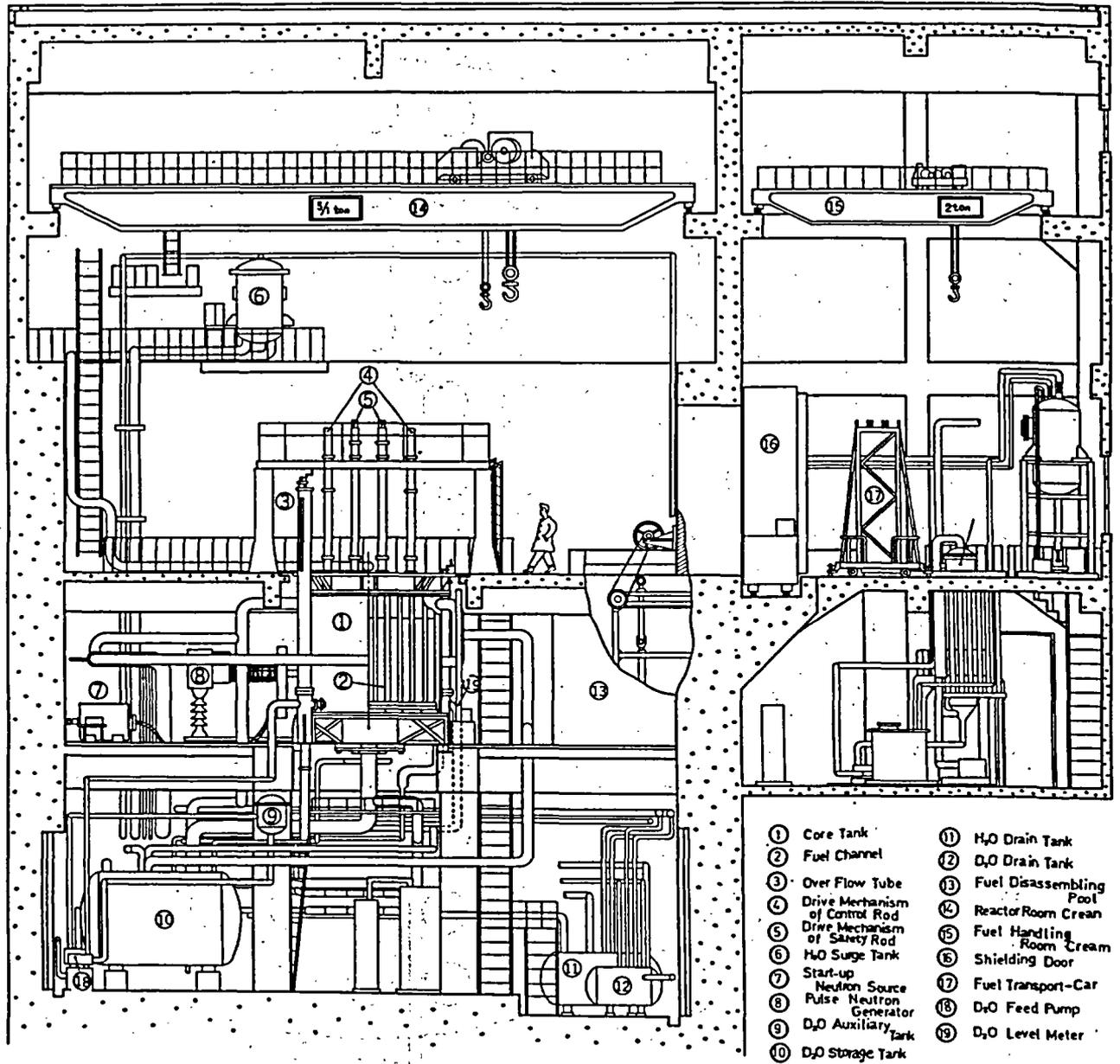
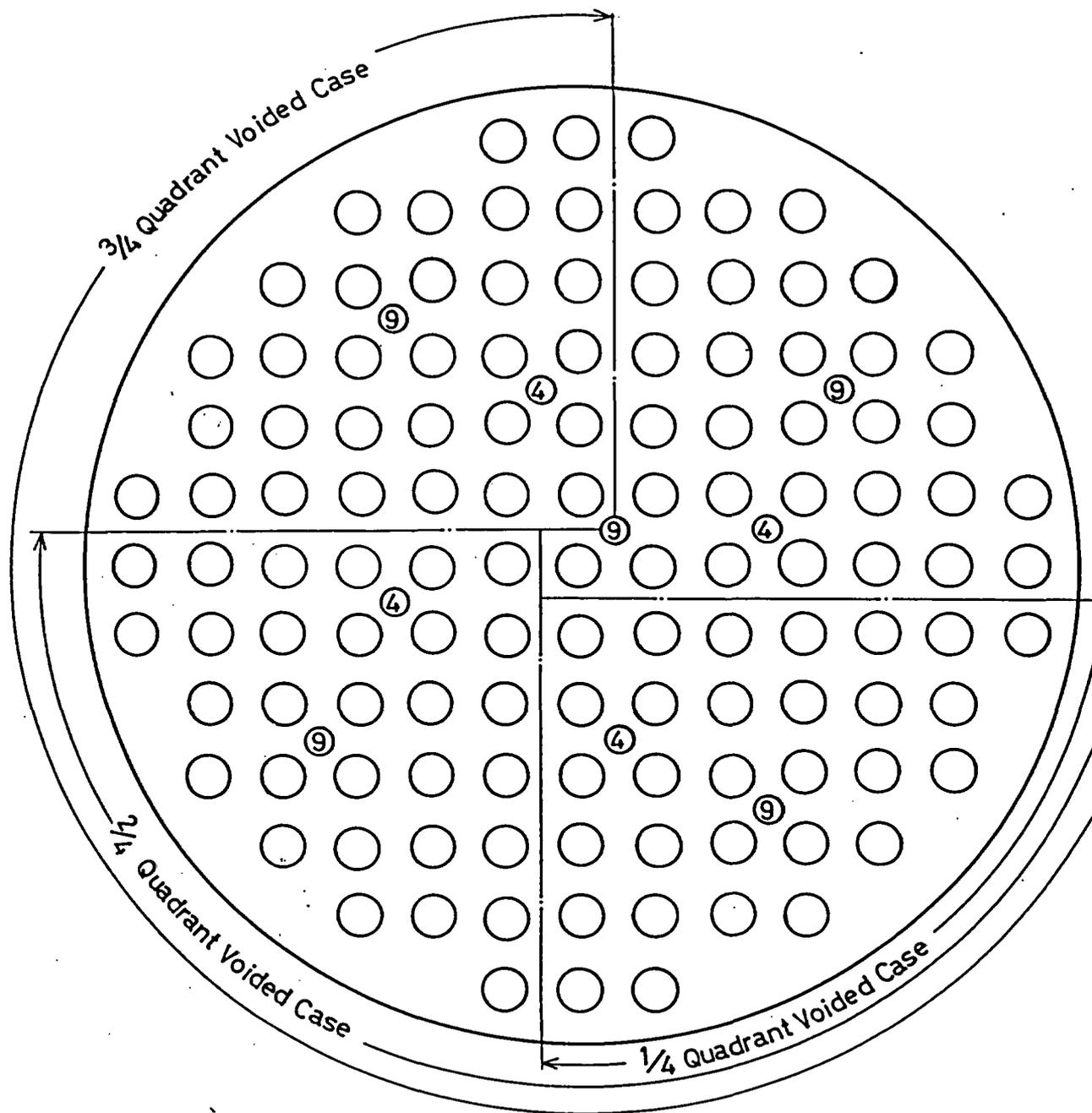
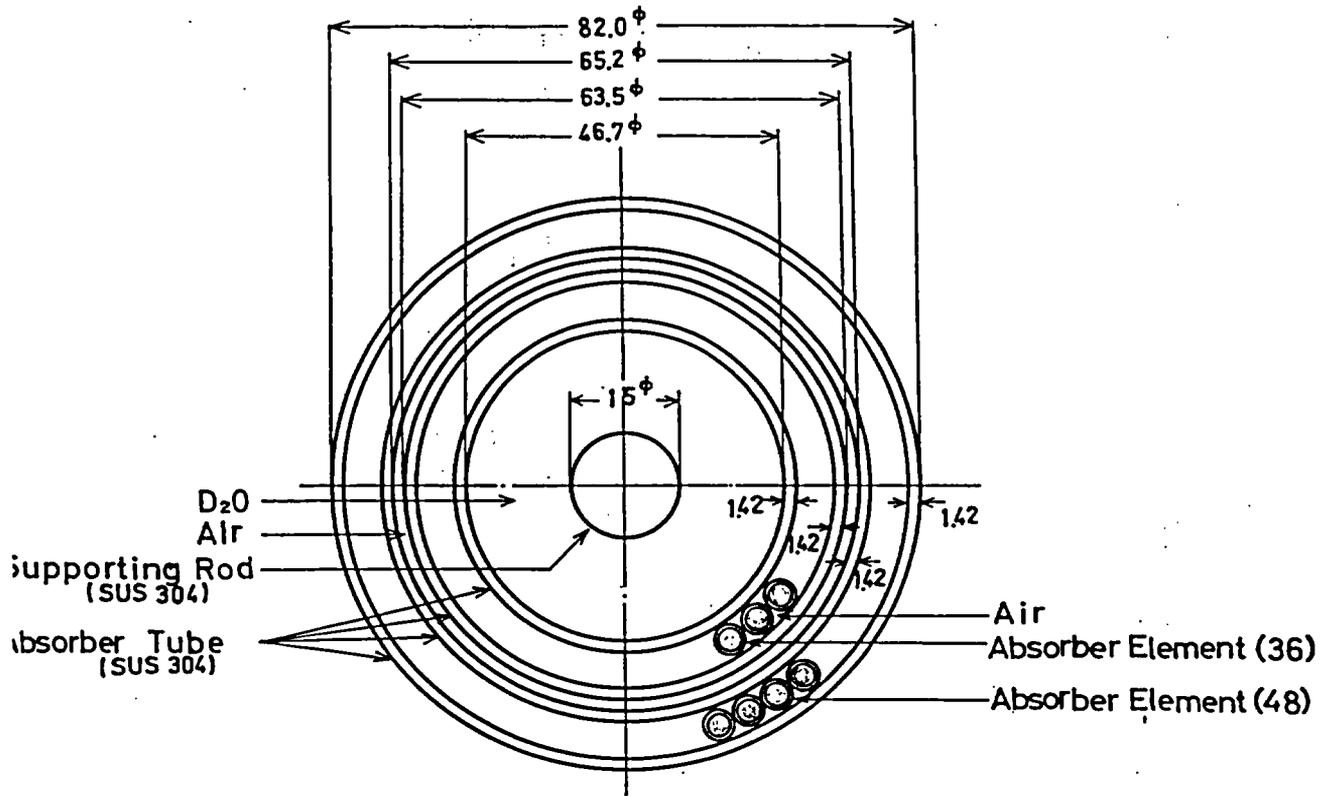


Fig.3 DCA



- 1.2% Enrich UO₂ Fuel Channel
- ④ Solid Absorber 4 Rods Inserted Position
- ⑨ Solid Absorber 9 Rods Inserted Position

Fig. 4 Configuration of Partially Voided Core



Absorber Element

Absorber	B ₄ C	34.8 ± 0.5g
Effective Length		1975 mm
Density		70% TD

Clad (SUS 304)

Diameter	3.58mm I.D.
Thickness	0.59mm
Length	1983mm

Fig. 5 Solid Absorber (Control Rod)

FUSIN

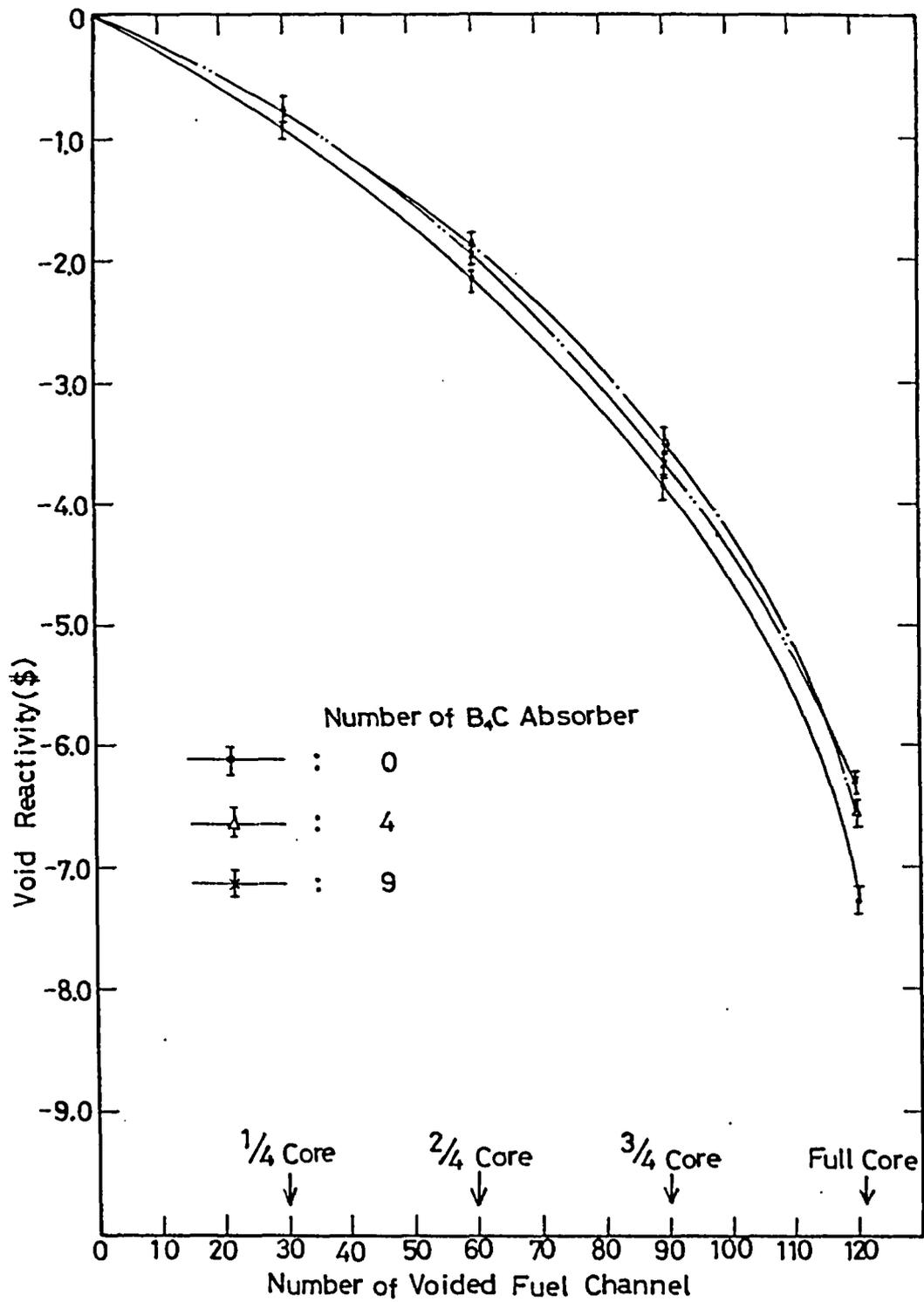


Fig.6 Void Reactivity of Partially Voided Core

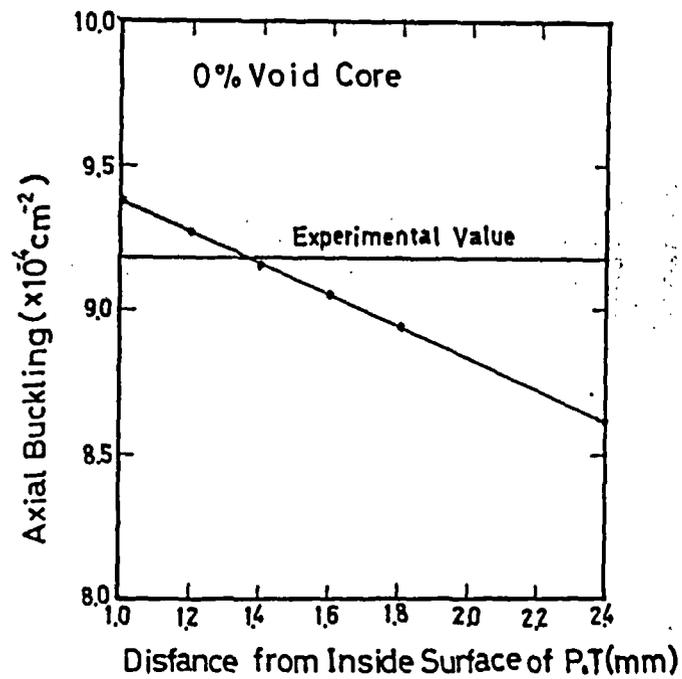


Fig.7 Axial Buckling vs Water Gap
(Determination of Soft Spectrum Region)

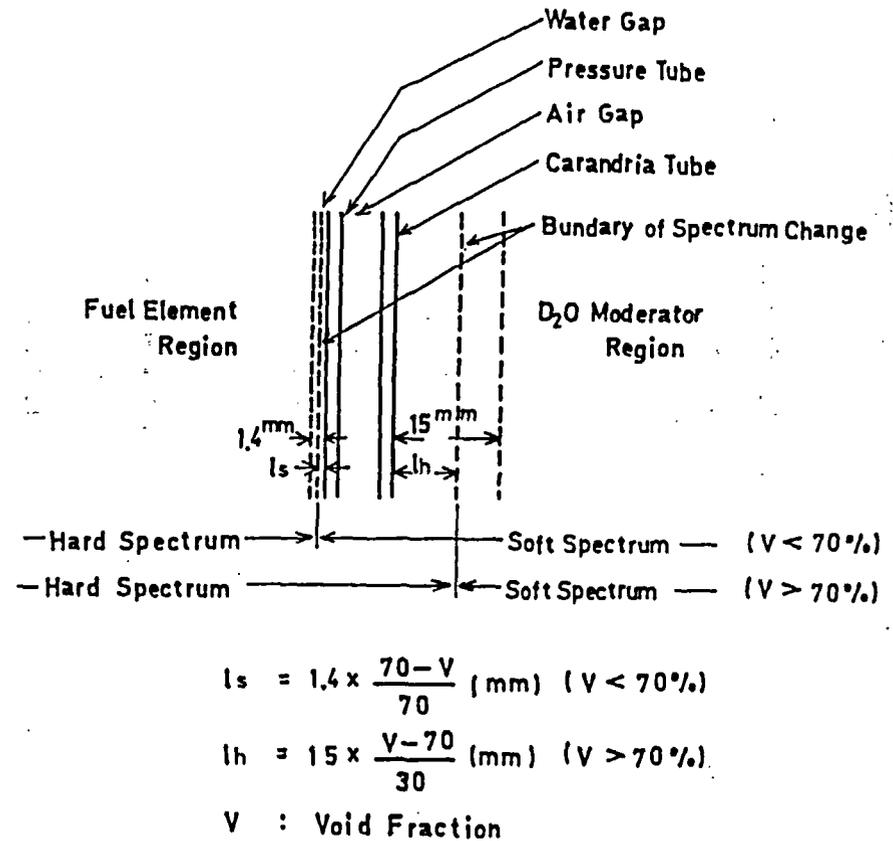


Fig.8 Set of Region for Thermal Neutron Spectrum

A c k n o w l e d g e m e n t

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R e f e r e n c e s

- (1) Y. Kanedo, J. Nucl. Sci. Technol. 8, 32 (1971)
- (2) R.W. Deutsch, Nucl. Sci. Eng. 5, 150 (1959)
- (3) B. Davison, MT-135 (1945)
- (4) W.P. Seidal, B. Davison and S.A. Kushneriuk, MT-207 (1946)
- (5) R.J. Royston, AERE T/R 2211 (1957)