REDUCED ENRICHMENT NEUTRONIC STUDY ON HIGH POWER RESEARCH REACTOR

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ABSTRACT

The FG2DB two dimensional two group diffusion/burnup code and the CELL few group cell parameter code with 69 groups neutron cross section database have been used for reduced enrichment neutronic study on high power research reactor with LEU fuel elements of uranium density 3.6 - 7.2 g/cm^3 and cladding thickness 0.38 - 0.56 mm. Results show the equivalence of fuel meat uranium density on the thickness changing. Control rod worth and other operation safety parameters of core are acceptable. Parts of the results have been fitted to linear or quadratic expression for easy application. The minimum critical value, excess reactivity, cycle length, fast and thermal fluxes and integrated fluxes, etc. are given. Analysis of these parameters shows that with the U-235 content increased by 20% or more, the LEU core main physical characteristics are similar to those of the HEU core; reduced enrichment almost has no influence on the fast neutron flux; the decreasing rate of the thermal neutron fluxes proportional approximately to the increasing rate of the U-235 content in fuel element; because of the cycle length prolonging, common radioisotope production and fuel element irradiation testing are apparently not influenced.

1. INTRODUCTION

Since 1970's, reduced enrichment for research and test reactors has become a worldwide tendency. To promote the conversion from the use of highly enriched uranium (HEU) to the use of low enriched uranium (LEU) fuels, a guidebook has been issued by the International Atomic Energy Agency. In this guidebook, typical conversion calculations for two generic examples are given, one of which for a 2 MW reactor and the other for a 10 MW reactor. In these examples, the uranium density of the fuel meat used are rather low since the high density fuel material has not been developed at that time. So the results and conclusions may be of limited use to high power (>10 MW) and high uranium density fuel research reactors.

High power research reactors are of great significance on national nuclear project. Up till now, a limited number of reduced enrichment calculations on high power research reactor core conversion have been published. It is believed that many high power research reactors, including the High Flux Engineering Test Reactor of China, the multi-tubular thin wall concentric tube type fuel assembly are used. For this reason, it was selected for the neutronic study of the core conversion example of high power research reactors. Following the experience of conversion practice, the high density U_3Si_2-Al dispersion fuel material has been chosen as the meat of the LEU fuel, and the U-235 enrichment-20%. In this paper, high power core physical characteristics will be studied for several uranium density and thickness of the LEU fuel meat.

2. CELL PARAMETERS OF LEU FUEL ASSEMBLY

The LEU fuel assembly model used in calculation is similar to the HEU fuel assembly of HFETR [2]. They have same shape and main dimensions. Their differences are: firstly, instead of $UA1_x$-Al alloy type in HEU fuel assembly, fuel meat material in LEU is U_3Si_2-Al dispersion type; U-235 enrichment is 20% instead of 90%; Secondly, the uranium density and thickness are changing in LEU fuel assembly.

Eight cases of LEU fuel assembly have been calculated, cell homogenized nuclear densities of the meat nuclides, parameters of each case, and calculational results are given in Table 1. The HEU fuel assembly parameters are also given.

The few group cell parameter code CELL, based on WIMS/D4 with 69 groups database, was used to calculate cell macroscopic parameters of LEU fuel assemblies. Results are listed in table 2. In Table 2, N(H/5) is the nuclei ratio of hydrogen and U-235, other symbols have usual meaning.

3. LEU CORE PARAMETERS

3.1 Core Parameters Calculation

The reference core configuration in calculation is similar to the first cycle core configuration of HFETR, but some simplifications have been done. The arrangement of the core is given in Figure 1.

There are 29 fuel assemblies, 57 beryllium blocks and 10 isotope targets. Absorber of the control rods is Ag-In-Cd alloy, and follower is beryllium rod.

Diffusion calculation has been done by the two dimensional two group diffusion/burnup code FG2DB, but using the input macroscopic parameters obtained by the CELL code. For verifying the correctness of the macroscopic parameters, a comparative calculation has been done with macroscopic parameters obtained by the CELL code or the FG2DB code itself, the resulting keff and total control rod worth are listed in Table 3.

From Table 3 it can be seen that two keff values are very close, but total rod worth differ a certain degree. However, experiment performed earlier shows that the CELL code value is nearer to the measured value, and the FG2DB code value is rather low. For this reason, it was confirmed that the macroscopic cross sections obtained by the CELL code are acceptable for the core diffusion calculation.

For verifying the FG2DB code in LEU core calculation, a comparative calculation of a special LEU-S case using the CITATION code or the FG2DB code has been performed**.** Fuel assembly of the LEU-S case has a U-235 content of 320 g, the core arrangement is the same as in this study, but four control rods in center region have fuel followers. The difference between keff's about 0.2%, and neutron flux - about 6%.

Results of the diffusion calculation of the LEU core are listed in Table 4. It can be seen that in all LEU core cases, core parameters were determined by the U-235 content value only, and no apparent other relation with the uranium density of meat thickness value has been found. For this reason, in further core neutronic study, when the uranium density and meat thickness are in the given range of this report, we need only to consider one of these two parameters. In further work, the uranium density was chosen as the variable of the core physical parameter.

3.2 Curve Fitting Results and Minimum Critical Fuel Number

From data in Tables 1 and 2, the curve fitting results of the LEU core parameters could be obtained. Results are listed in Table 5 and shown in Figures 2 and 3.

Results of the minimum critical fuel number are listed in Table 6. In the calculation, the axial water reflector saving λ_h = 9 cm; the radial reflector values saving for beryllium, λ_r^{Be} = 12.7 cm, and for water $\lambda_r^{H_2O}$ = 7.8 cm, these values were obtained from the clean HEU critical experiment, and used here approximately. The HEU minimum critical fuel number was also given.

4. BURNUP CALCULATION OF LEU CORE

4.1 Method and Main Results of Burnup Calculation

The FG2DB code was used in burnup calculation. Except the fast group microscopic cross sections of fuel and control rod cells were calculated by the CELL code, other fast and all thermal microscopic cross sections were obtained by the FG2DB itself[5]. The resonance escape probabilities were recalculated, while the thermal neutron self-shielding factors were remain unchanged. To check the applicability of the method, the initial k_{eff} values have been compared for macro- or micro- cross section input of several core case. Results are listed in Table 7. From Table 7 it can be seen that two sets of input data have got quite good compliance.

The cycle length of the core is an important parameter in burnup calculation and it is difficult to get exact value. For reliability consideration, cycle lengths of LEU-S case have been calculated by the FG2DB or CITATION codes. The former is about 12% higher than the latter, k_{eff} burnup curves from burnup calculation are shown in Figures 4 and 5. From these Figures it can be seen that k_{eff}'s decrease very rapid at the beginning time, then decrease linearly after reaching equilibrium xenon. Reactivity loss rate R_T could be calculated according to the following relationship in linear range of *keff* curve:

$$R_T = 2\,\frac{k_{eff}(o) - \Delta k_{Xe} - 1}{T} \qquad (3)$$

In equation (3), R_T -reactivity loss rate, Δk/100MWd; T-core cycle length, day; $\Delta k_{\chi e}$- equilibrium xenon poison reactivity; factor 2 appears when the operational power is 50 MW. In Table 8, T and R_T were obtained from bumup calculation, $\Delta k_{\chi e}$ and resonance escape probability were calculated using methods given in reference [6]. From Table 8 it can be seen that the cycle length T of the LEU cores with the fuel assembly U-235 content m are linearly related, see Figure 6. Fitting results get:

$$T = 0.2842m - 60.6 \qquad (4)$$

with bias less than 2%.

4.2 Fast and Thermal Flux and Fluence in Core

The burnup calculation gives core fast and thermal flux distribution when the control rods withdraw step by step and the reactor operational power is 50 MW. The average fast and thermal fluxes of all core cases at BOC and EOC are listed in Table 9. From Table 9 it can be seen that compared with HEU core, all cases of LEU core have following tendency: the average fast fluxes are almost the same, but the average thermal fluxes decrease about 18% - 45%, except in case LEU-D1.

In Figures 7 and 8, the fast and thermal flux distributions along the core K axis and 1,2AB connecting line axis (AB axis) of LEU-D2 core case are given. In Figures 9 and 10, the fast and thermal flux ratio distributions along the core K axis and AB axis of LEU-D2 and HEU core cases are given. From Figures 9 and 10 it can be seen that the thermal flux ratios appear peaks at the 1,2AB beryllium follower position along the AB axis, and the fast flux ratios appear small peaks only. Along the K axis, at in-core target position, similar situation with smaller amplitude have been found. Furthermore, in fuel region, the fast flux ratio is about 1.0, but the thermal flux ratio-0.8 approximately; in outer part of the reflector, the fast and thermal flux ratios are both about 1.1. For other cases, similar tendency have been found, but the values are somewhat different. Detail results are given in reference [7].

The thermal fluence in K9 and K13 targets of fuel region, and in G7, G11, G14, K6, K16, N8, P11, P15 targets of reflector region for each case were calculated. The total fluence of targets in fuel and reflector region, and the total fluence ratio, respectively, of the LEU and HEU cores are listed in Table 10.

From Table 10 it can be seen that comparing with HEU core, the target thermal fluence in LEU-D1 core is rather low, in LEU-D2 core is near, and in LEU-D4 core is much higher. These results show that the isotope production would not decrease or rather would increase when the U-235 content in LEU fuel assembly is 20% or more higher than in HEU fuel assembly.

5. CONCLUSION

This paper is a systematic reduced enrichment neutronic study on one of the most popular type high power research reactor, i.e., multi-tubular thin wall concentric tube type fuel assembly, light water moderator and coolant, beryllium reflector, and triangle arrangement of lattice cell type research reactor. In the study, the fuel meat uranium density range is taken to be 3.6 g/cm^3-7.2 g/cm^3, which is the uranium density reached and will be reach in near future. The cladding thickness range is taken to be 0.38 - 0.56 mm, which is in the safety limit and reached by mature technique. For these reason, all cases under study is thought to be realistic possibility.

The FG2DB two dimensional two group diffusion/burnup code, which has been used successfully in our Institute, and the CELL lattice cell few group parameter code with 69 group database, have been used for neutronic study. The CELL code gives the input parameters of macro- or micro- cross section. The acceptance of the calculational method has been verified not only in HEU core validation calculation, but also in comparison of the LEU core results obtained by the CITATION code.

The calculational results show, for thin wall fuel assembly, if the fuel tube and original fuel assembly configuration and dimension are kept unchanged, to change the fuel meat uranium density of the cladding thickness for getting different U-235 content of one fuel assembly, is equivalent physically. These two technological routines are both applicable.

Results also show, in the range of the LEU core parameters, all core cases are possible physically. Operation and safety closely related parameters, such as: control rod worth, maximum allowable power, core power non-uniform coefficient, etc., are within the acceptable limits. Other results, including initial excess reactivity, cycle length, reactivity loss rate, minimum critical mass, fast and thermal flux and fluence, are also obtained by neutronic calculation.

Parts of the results are fitted to linear or quadratic expressions with quite satisfying accuracy. They could be used in initial case selection or core parameter evaluation. In application of these expressions, attention should be paid on the applicable range of them.

Calculational results also show that for core neutronic designs practically meaningful in engineering, U-235 content in LEU fuel assembly should be greater than in related HEU fuel assembly. If in both assemblies there are same U-235 content, the cycle length in LEU core would be much less than in HEU core. When U-235 content is increased by 20% or more, core main characteristics would be quite near, making it applicable in practice. When U-235 content is too high, the high thermal neutron flux, which is one of the main superiority index of the high power research reactor, would be decreased too much, and the difficulty in technology would be increased, practical significance would be lost finally.

Calculational results also show that after reduced enrichment, core fast neutron flux remains almost unchanged; material tests, which need high fast flux, are not influenced. Since the thermal flux decreases proportionally to the increase of U-235 content in fuel assembly, unfavourable influence would be appeared in beam port application, high specific isotope and transuranic isotope production. However, since the cycle length would be prolonged, for common isotope production and fuel element irradiation testing, influences would be compensated in a great degree.

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Table 1 Cell homogenized nuclear density of the meat nuclides

Core case*	LEU-D1	LEU-D2	LEU-D3	LEU-D4	LEU-T1	LEU-T3	LEU-T4	HEU
Fuel meat thickness d, mm	0.50	0.50	0.50	0.50	0.38	0.62	0.74	0.50
Cladding thickness, mm	0.50	0.50	0.50	0.50	0.56	0.44	0.38	0.50
Uranium density of fuel ρ v • g/cm^3	3.6	4.8	6.0	7.2	4.8	4.8	4.8	0.86
U-235 content in fuel m , g	251.2	334.0	418.9	500.2	253.8	414.2	494.4	270
Relative content of uranum-235	0.930	1.237	1.551	1.852	0.940	1.534	1.831	1.000
N_{U5} x 10^{20}cm^{-3}	1.813	2.418	3.022	3.627	1.837	2.997	3.578	1.947
N_{U8} x 10^{20}cm^{-3}	7.161	9.548	1.193	1.432	7.349	11.991	14.312	0.2163
N_{A1} x 10^{20}cm^{-3+}	212.5	203.6	192.7	179.4	196.6	210.6	217.6	229.8
N_{Si} x 10^{20}cm^{-3}	6.15	8.20	10.25	12.30	6.25	10.20	12.17	0

*Case LEU-T2 dismissed since it is identical to case LEU-D2

+ Cladding Al included in N_{A1}

Table 2 Two group macroscopic parameters of the fuel cell

Core Case	LEU-D1	LEU-D2	LEU-D3	LEU-D4	LEU-T1	LEU-T3	LEU-T4	HEU
D_1, cm	1.373	1.368	1.367	1.366	1.375	1.362	1.357	1.374
Σ_{a1}, cm^{-1}	0.00549	0.00658	0.00761	0.00856	0.00553	0.00757	0.00850	0.00326
Σ_{s1}, cm^{-1}	0.0238	0.0229	0.0221	0.0214	0.0237	0.0221	0.0214	0.0246
$(V\Sigma_f)_1$, cm^{-1}	0.00422	0.00546	0.00667	0.00782	0.00426	0.00661	0.00773	0.00434
D_2, cm	0.238	0.239	0.240	0.241	0.238	0.240	0.241	0.2397
Σ_{a2}, cm^{-1}	0.0864	0.1043	0.1207	0.1356	0.0869	0.1202	0.1351	0.0898
$(V\Sigma_f)_2$, cm^{-1}	0.1491	0.1864	0.2204	0.2513	0.1502	0.2195	0.2502	0.1584
K∞	1.549	1.576	1.587	1.589	1.550	1.586	1.588	1.714
B_M^2, cm^{-2}	0.01054	0.01131	0.00172	0.01192	0.01055	0.01175	0.01203	0.01307
M^2, cm^2	52.07	50.93	50.30	49.40	52.12	49.86	48.89	54.64
N(H/5)	221.5	166.3	132.8	110.7	218.8	134.1	112.3	198

Table 3 Results of the verifying calculation with different macroscopic parameters

Core Parameter	Code		Bias
	CELL	FG2DB	
k_{eff}	1.1498	1.1480	0.2%
Total rod worth *, β_{eff}	37.9	33.6	12.8%

*Measured value is 36 β_{eff}

Table 4 Main results of the LEU core diffusion calculation

Core case	LEU-D1	LEU-D2	LEU-D3	LEU-D4	LEU-T1	LEU-T3	LEU-T4	HEU
Δk_{eff}	0.0829	0.1229	0.1464	0.1613	0.0842	0.1461	0.1608	0.1786
Maximum power, MW	74	72	71	70	74	71	70	74
Power non-uniform factor k_ν	1.941	1.983	2.021	2.057	1.942	2.021	2.056	1.940
Maximum fast flux, $\times 10^4 n\nu$	20.80	24.27	27.33	30.04	20.90	27.26	29.95	21.00
Averaged fast flux, $\times 10^4 n\nu$	10.34	12.21	13.88	15.37	10.40	13.83	15.30	10.41
Maximum thermal flux, $\times 10^4 n\nu$	6.14	6.17	6.20	6.23	6.14	6.20	6.23	6.14
Averaged thermal flux, $\times 10^4 n\nu$	3.04	2.92	2.82	2.73	3.03	2.82	2.73	3.04
1,2AB safety rod worth, β_{eff}	15.76	14.16	13.15	12.44	15.70	13.16	12.45	14.47
1,2ZB automatic rod worth, β_{eff}	1.47	1.35	1.28	1.22	1.46	1.27	1.22	1.30
1-14SB shim rod worth, β_{eff}	23.73	22.03	20.97	20.24	23.70	20.97	20.21	21.96
Core subcriticality (safety rods out), β_{eff}	-13.48	-8.25	-5.04	-2.89	-13.32	-5.05	-2.91	3.06
Total rod worth, β_{eff}	40.97	37.55	35.40	33.91	40.87	35.41	33.89	37.73

*Power and neutron flux are normalized according to the maximum heat flux density
$g = 3.56 \times 10^6$ w/m^2(all rods out).

Table 5 Curve fitting results of the LEU core parameters

Core parameter	Variable	Curve fitting expression	Maximum relative bias
$K\infty$	m, g	$K\infty = 1.399 + 8.1913 \bullet 10^{-4} m - 8.8181 \bullet 10^{-7} m^2$ (1)	0.1%
	ρ_v,g/cm^3	$K\infty = 1.3982 + 0.05743\rho\upsilon - 0.0043\rho_\upsilon^2$	0.06%
	N(H/5)	$K\infty = 1.403 + 44.3\frac{1}{N(H/5)} - 26.4\frac{1}{N^2(H/5)}$	0.1%
M^2	m,g	$M^2 = 56.955 - 0.02291m + 1.45 \bullet 10^{-5} m^2$ (2)	0.6%

Table 6 Minimum critical number of fuel assemblies

Core Case	LEU-D1	LEU-D2	LEU-D3	LEU-D4	HEU
Radial reflector - Water	24.0	21.5	20.3	19.7	17.0
Radial reflector - Beryllium	11.7	10.0	9.2	8.8	7.0

Table 7 Initial keff using macro- or microscopic cross sections as input parameters

Core Case	LEU-D1	LEU-D2	LEU-D4	HEU
Initial keff (macro-input)	1.0829	1.1229	1.1613	1.1785
Initial keff (micro-input)	1.0768	1.1204	1.1625	1.1638
Relative bias	0.6%	0.2%	0.1%	1.2%

Table 8 Main results of the LEU core burnup calculation

Core Case	LEU-D1	LEU-D2	LEU-D4	HEU
Cycle length T ; d	10.7	33.7	81.6	30.6
Resonance escape probability P	0.938	0.928	0.912	0.993
Equilibrium xenon Δ kxe	0.0387	0.0402	0.0411	0.0397
Reactivity loss rate R_T, Δk/100MWd%	0.712	0.476	0.297	0.811

Table 9 Average fast and thermal neutron flux in core

Core Case	LEU-D1	LEU-D2	LEU-D4	HEU
BOC fast flux	7.92	7.76	7.51	7.67
BOC thermal flux	2.16	1.65	1.12	2.02
EOC fast flux	8.10	8.14	8.11	8.12
EOC thermal flux	2.29	1.91	1.41	2.56

Table 10 Integrated thermal neutron flux in targets

Core Case	LEU-D1	LEU-D2	LEU-D4	HEU
Target in fuel region, $\times 10^{20}$ nvt	3.42	8.95	17.48	9.38
Target in reflector, $\times 10^{20}$ nvt	4.86	13.07	27.69	12.54
Relative ratio in fuel region	0.364	0.954	1.863	1.00
Relative ratio in reflector	0.387	1.042	2.208	1.00

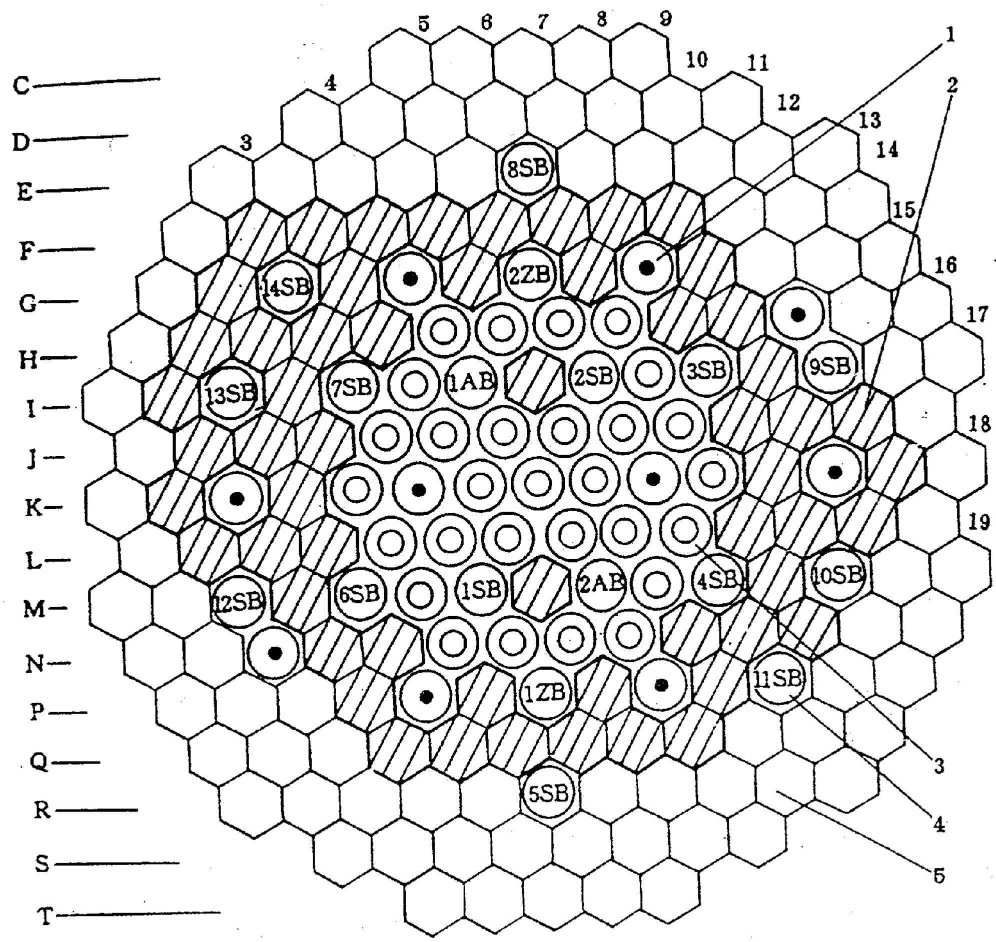


Fig • 1 LEU core arrangement

1 — isotope target, 2 — beryllium block, 3 — LEU fuel assembly, 4 — cont rod(AB,ZB,SB—safety, automatic control, shim rod respectively), 5—aluminum block

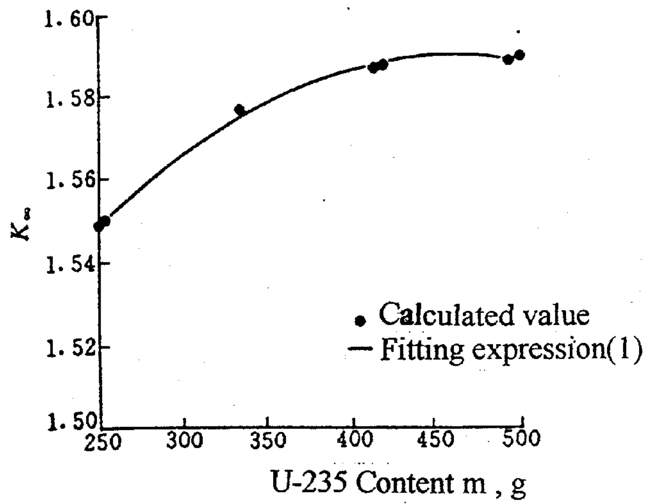


Fig. 2 K_∞ as a function of fuel element U-235 content

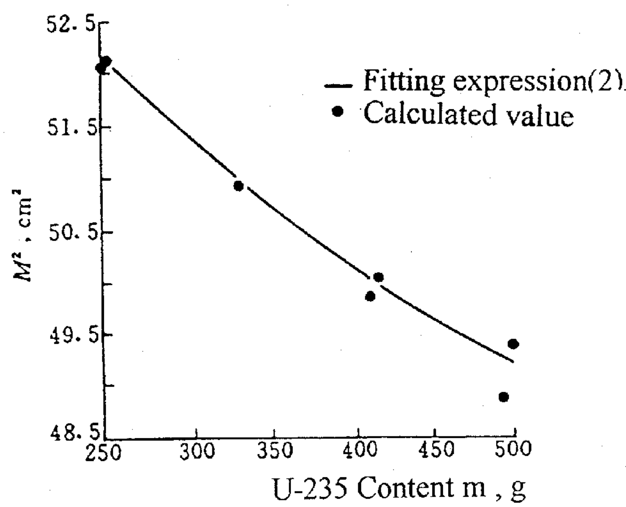


Fig. 3 M^2 as a function of fuel element U-235 content

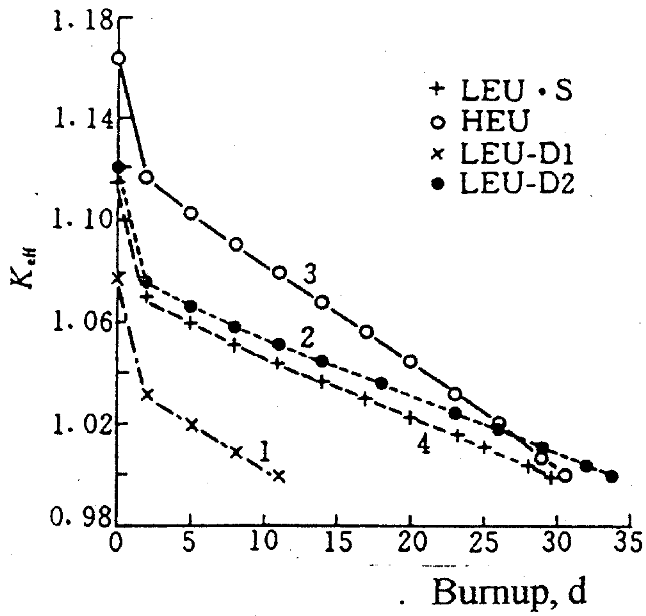


Fig. 4 K_{eff}-Burnup curves of some cores

1 LEU-D1 2 LEU-D2 3 HEU 4 LEU-S

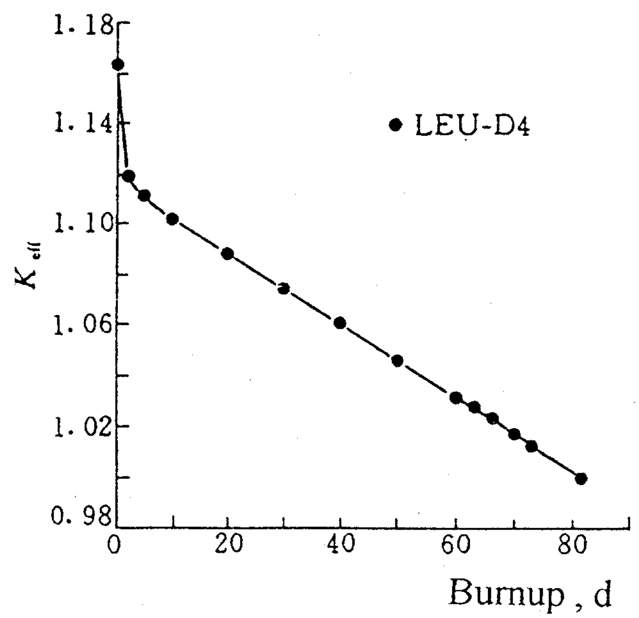
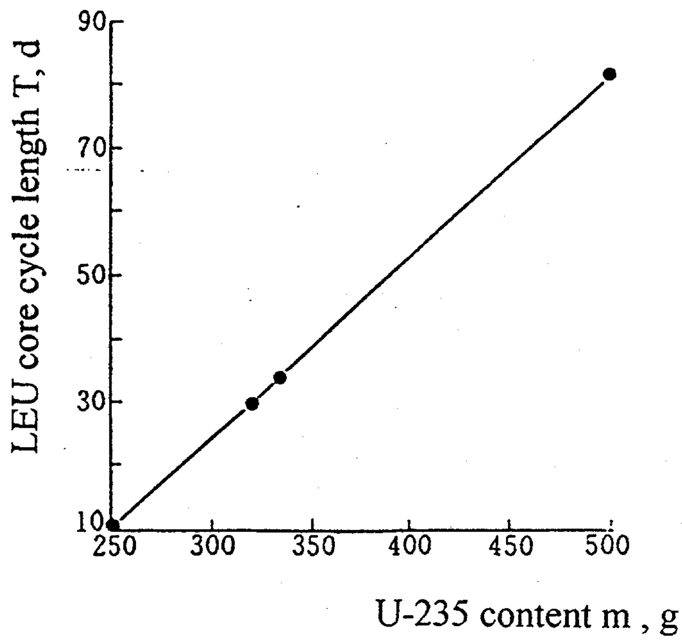


Fig. 5 K_{eff}-Burnup curve of LEU-D4 core



- Fig. 6 Relationship of LEU core cycle length with fuel element U-235 content

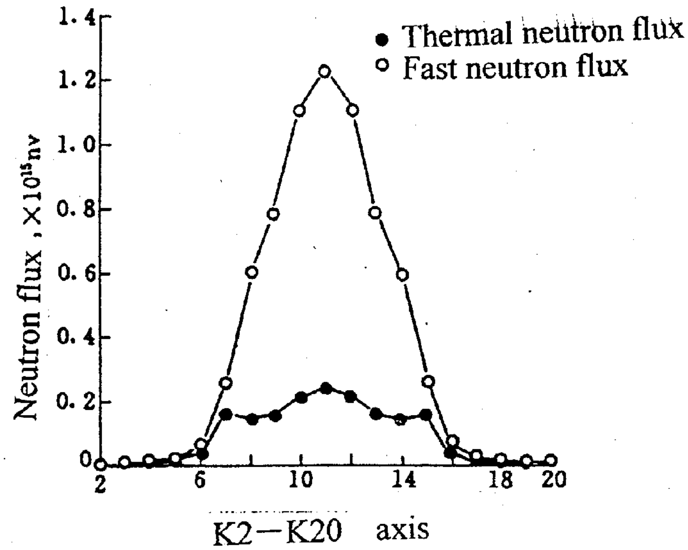


Fig. 7 Fast and thermal neutron flux distribution along *K* axis in LEU-D2 core

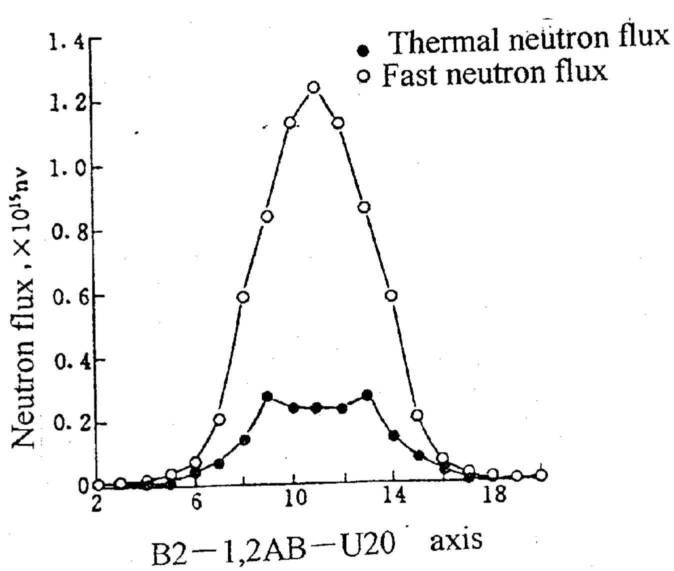


Fig. 8 Fast and thermal neutron flux distribution along *AB* axis in LEU-D2 core

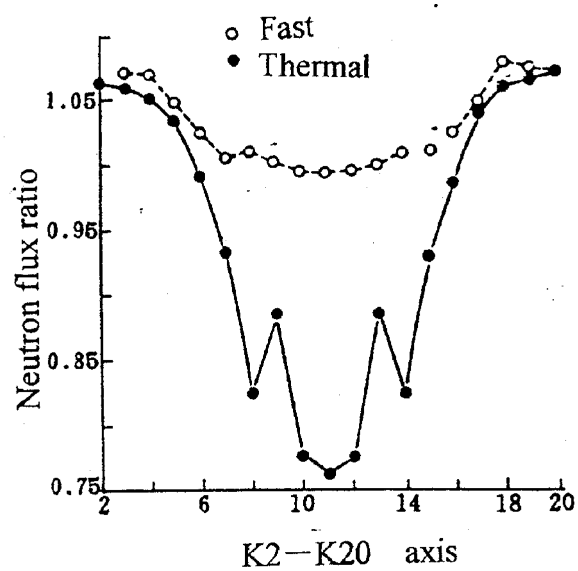


Fig. 9 LEU-D2 and HEU core neutron flux ratios along *K* axis

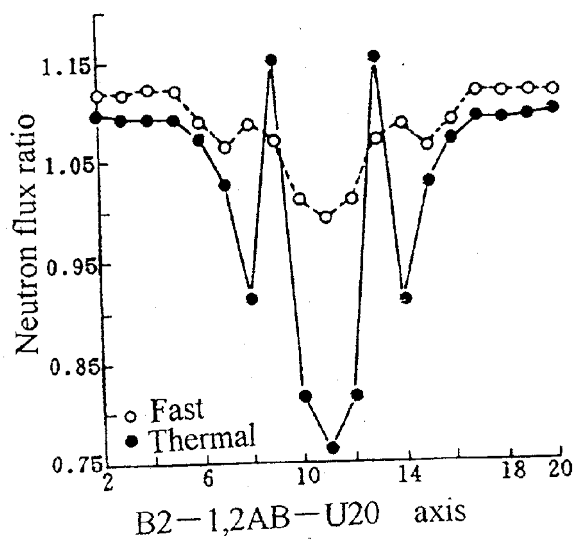


Fig. 10 LEU-D2 and HEU core neutron flux ratios along *AB* axis