Measurement and analysis of fission rates in a spherical mockup of uranium and polyethylene*

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Abstract: Measurements of the reaction rate distribution were carried out using two kinds of Plate Micro Fission Chamber (PMFC). The first is a depleted uranium chamber and the second an enriched uranium chamber. The material in the depleted uranium chamber is strictly the same as the material in the uranium assembly. With the equation solution to conduct the isotope contribution correction, the fission rate of ^{238}U and ^{235}U were obtained from the fission rate of depleted uranium and enriched uranium. Then, the fission count of ^{238}U and ^{235}U in an individual uranium shell was obtained. In this work, MCNP5 and continuous energy cross sections ENDF/BV.0 were used for the analysis of fission rate distribution and fission count. The calculated results were compared with the experimental ones. The calculation of fission rate of DU and EU were found to agree with the measured ones within 10% except at the positions in polyethylene region and the two positions near the outer surface. Because the fission chamber was not considered in the calculation of the fission counts of ^{238}U and ^{235}U, the calculated results did not agree well with the experimental ones.

Key words: fission rate, depleted uranium, polyethylene, neutronics analysis

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1 Introduction

In the new blanket concept of fusion-fission hybrid, depleted uranium was considered to be a candidate for neutron breeding and energy generation, and water for heat transmission. The volume ratio of about 2.0 for uranium to hydrogen was selected. It is considered that such structure is the best one for energy production and fissile breeding, and also for heat transmission [1–3]. For the feasibility of energy breeding and neutron multiplication in this blanket concept, the neutron transport and fission generation should be accurately predicted [4]. To do this, the method, code and data are of fundamental importance in the concept research as in the design of fission plants and fusion devices [5]. Therefore, a series of fission rate experiments on the assemblies having similar configuration to the hybrid blanket were planned. Now, a spherical assembly of depleted uranium and polyethylene in which similar ratio of uranium to hydrogen was selected has been setup to simulate the structure in blanket concept design and the fission rate experiment on the assembly by fission chamber has been conducted with D-T neutron generated at the center. For comparison with the experimental data, as in hybrid concept study, radiation transport tool MCNP5 [6] and nuclear data library ENDF/BV.0 [7] were adopted in this work to be checked.

2 Experiment and procedure

A neutronics experiment of fission rate distribution in an assembly of depleted uranium and polyethylene has been conducted with D-T neutron at the center. Fig. 1 shows the arrangement between the experimental assembly and D-T neutron source. The experimental assembly consisted of two layers of polyethylene and three layers of depleted uranium. The dimensions of the assembly (outer radius/inner radius) is 30 cm(DU)25.4 cm(PE)23.34 cm(DU)19.4 cm(PE)18.1 cm(DU)13.1 cm. The spherical shells of polyethylene were newly constructed while the depleted uranium shells were constructed in the large depleted uranium experiments [8]. There are six channels in the assembly, one of which is for the detector and one for the drift tube while the others were just filled with cylindrical depleted uranium blocks

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and depleted uranium sleeves. The uranium sleeves were also used in the channel for the detector to keep the blocks of uranium and polyethylene from dropping down and to install the detector at a specific position. The material in the sleeves is kept strictly the same as the material in the shells by combining various blocks with different thickness of 0.2 cm to 5.0 cm except for the detector.

D-T neutrons were generated by bombardment of 250 keV and 100 μA deuteron onto a tritiated titanium target (90 GBq). The intensity was about 1.5×10^{10} neutrons/s. The fluence of D-T neutrons was monitored by counting the associated 3.5 MeV alpha particles with a silicon surface barrier detector at the angle of 178.2° to the incident deuteron beam and the experimental error of the fluence of D-T neutrons was about 2.5% [9]. Depleted uranium and enriched uranium Plate-Micro-Fission-Chambers (PMFC) were used to measure the fission rate along the channel as a function of distance from the core of the assembly to the detector position. The atomic densities of fissile material in the PMFCs have been accurately measured from the alpha particle emitted from the fissile material before the PMFCs were fabricated. Table 1 shows the characterizations of PMFCs, including atomic densities, dimensions, active area and so on. The sources of the uncertainties in measured results were summarized in Table 2.

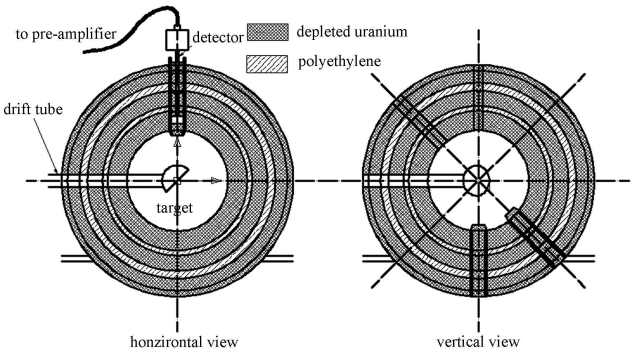

Fig. 1. The schematic view of spherical assembly of depleted uranium and polyethylene.

Table 1. The characterization of the PMFCs.

PMFCs	nuclide	number/atoms	uncertainty (%)	detector size/cm	active area/cm^2	efficiency (%)
DU	^{238}U	3.1×10^{18}	1.3	ϕ 3×2	4.52	95.64
	^{235}U	1.3×10^{16}				
EU	^{238}U	2.1×10^{17}	2.4	ϕ 3×2	4.52	96.63
	^{235}U	2.3×10^{18}				

Table 2. The sources of the uncertainties in fission rate measurement.

sources	%
absolute neutron yield	2.5
statistics of alpha counts	0.5
solid angle viewed by the detector	1.7
correction for neutron anisotropy	1.2
fission chamber measurement	1.64(DU)/2.6(EU)
atomic number density	1.3(DU)/2.4(EU)
counting statistics	1.0
total uncertainty	2.99(DU)/3.61(EU)

Measured fission rates have been estimated by the following Eq. (1)

$$f(r)=\frac{N_{\rm f}}{AN\eta}, \tag{1}$$

where $f(r)$ is the fission rate of depleted uranium (DU) or enriched uranium (EU) at a specific position. r is the radius and $N_{\rm f}$, A, N, η are the sum of yields obtained by PMFCs in which the fission reactions occurred with ^{238}U, ^{235}U, ^{234}U and so on, the atom number of all uranium isotopes in the PMFCs, the total amount of neutrons emitted from D-T reaction and the efficiency of fission fragments shaping a pulse in PMFCs, respectively. In addition, $N_{\rm f}$ has been corrected by dead time.

For the ^{234}U contribution as small as 1.0% in DU PFMC and EU PFMC, there is just ^{238}U and ^{235}U which have to be considered for contribution to the measured fission yields in Eq. (1). So in a specific DU or EU PFMC, the fission yields $N_{\rm fd}$ or $N_{\rm fe}$ are formed mainly by that of ^{238}U $N_{\rm f8d}$ or $N_{\rm f8e}$ and that of ^{235}U $N_{\rm f5d}$ or $N_{\rm f5e}$. Then, based on the Eq. (1), the fission yields $N_{\rm fd}$ or $N_{\rm fe}$ can be given as the following equations.

$$\begin{aligned} N_{\rm fd}&=N_{\rm f8d}+N_{\rm f5d}\\ &=f_8A_{\rm 8d}N\eta+f_{\rm 5d}A_{\rm 5d}N\eta\\ &=(f_8A_{\rm 8d}+f_5A_{\rm 5d})N\eta\\ &=(f_8A_{\rm d}K_{\rm 8d}+f_5A_{\rm d}K_{\rm 5d})N\eta\\ &=(f_8K_{\rm 8d}+f_5K_{\rm 5d})A_{\rm d}N\eta, \end{aligned} \tag{2}$$

$$\begin{aligned} N_{\rm fe}&=N_{\rm f8e}+N_{\rm f5e}\\ &=f_8A_{\rm 8e}N\eta+f_{\rm 5e}A_{\rm 5e}N\eta\\ &=(f_8A_{\rm 8e}+f_5A_{\rm 5e})N\eta\\ &=(f_8A_{\rm e}K_{\rm 8e}+f_5A_{\rm e}K_{\rm 5e})N\eta\\ &=(f_8K_{\rm 8e}+f_5K_{\rm 5e})A_{\rm e}N\eta, \end{aligned} \tag{3}$$

here, f_8 is the fission rate of ^{238}U fission reaction, f_5 is the sum of fission rate of ^{235}U fission reaction, $f_{\rm d}$ is the fission rate of depleted uranium, $f_{\rm e}$ is the fission rate of

enriched uranium, K_{8d} and K_{5d} are the atom percentages of ^{238}U and ^{235}U in depleted uranium while K_{8e} and K_{5e} are the atom percentages of ^{238}U and ^{235}U in enriched uranium. A_d is the number of fission nuclides in DU chamber while A_e is that in EU chamber. In this work, K_{8d} and K_{5d} were 0.9958 and 0.0042 while K_{8e} and K_{5e} were 0.0863 and 0.9137.

According to Eqs. (1, 2, 3), the relations of the fission rate of DU or EU and the fission rate of ^{238}U and ^{235}U in DU or EU can be presented by the following equations.

$$f_d = \frac{N_{fd}}{A_d N\eta} = f_8 \times K_{8d} + f_5 \times K_{5d}. \quad (4)$$

$$f_e = \frac{N_{fe}}{A_e N\eta} = f_8 \times K_{8e} + f_5 \times K_{5e}. \quad (5)$$

As we can see, there are two equations with two unknown parameters. Based on fission rate of DU f_d and EU f_e, the fission rate of ^{238}U and ^{235}U can be obtained; the method was named equation method.

All the fission rate results are normalized to one atom, one source neutron. Then, based on the fission rate of ^{238}U and ^{235}U, the fission counts of ^{238}U and ^{235}U in a specific uranium layer of the assembly can be obtained by integration method. The following formula shows the integration equation to get the fission count of ^{238}U,

$$P_{f8} = \left(2\pi\rho \times \frac{K_{8d}}{238}\right) \times \iint r^2 f_8(r) \sin\theta \mathrm{d}\theta \mathrm{d}r, \quad (6)$$

$$P_{f8} = \left(4\pi\rho \times \frac{K_{8d}}{238}\right) \times \iint r^2 f_8(r) \mathrm{d}r, \quad (7)$$

where ρ is the density of the depleted uranium assemblies; r is the distance of the measuring position to the core, ranging from 0 to R, R is the outer radius of the assemblies; θ is the angle of the measuring position to incident D^+ beam and K_{8d} has the same meaning in Eq. (2); the fission counts of ^{235}U can be given in a similar equation. According to the previous experimental analysis, the fission rate along the channel which is perpendicular to the incident D^+ beam is almost the average of all fission rates which are at the asymmetric angle to the experimental channel. So the Eq. (6) was simplified to be Eq. (7). Then with ORIGIN 6.0 code and trapezoidal area method, the fission count was obtained. The uncertainty of the fission count is about 3.2% for ^{238}U and 4.2% for ^{235}U. The contribution to the quoted uncertainty coming from the fission rate of ^{238}U (3.0%) and ^{235}U (4.0%), the distance of the measuring position to the core (1.0%).

3 Monte Carlo analysis

Analyses of the fission rate experiment were carried out with the Monte Carlo code MCNP 5 [6] and attached continuous energy cross section ENDF/BV.0 [7]. The experimental configuration was modeled in detail, including target chamber, drift tube, void in sleeves, the sheet for detector and so on. The surface crossing estimator was used. Neutron histories were accumulated to obtain a good statistical accuracy, less than 2.0%. In the calculation, an isotropic distribution of source neutrons was assumed. By filling all the channels with uranium blocks in the calculation model and with a point neutron source with 14.1 MeV placed in the center, fission counts in individual uranium shells were obtained, and the influence of materials in detectors was also analyzed.

4 Results and discussion

4.1 Fission rate distribution

Figure 2 shows experimental and calculated results of depleted uranium and enriched uranium fission rate as a function of distance from the core to measuring position. In the figures, DU means the material in the shell is depleted uranium and PE means the material in the shell is polyethylene. The experimental error band was smaller than the size of the dots showing the experimental results, and the calculation ones also. The depleted uranium fission rate in which the ^{238}U contribution is as high as 90% decreases with the distance from the core increasing, and the enhancement of fission rate around the polyethylene region was also indicated from the two platforms on the curve. As for enriched uranium fission

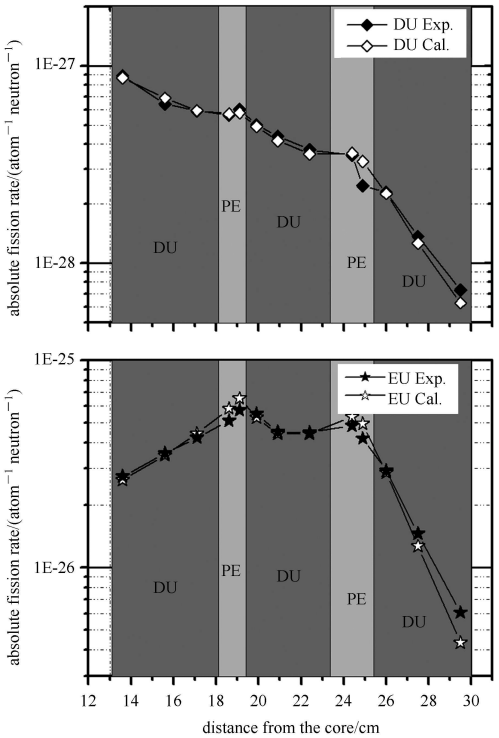

Fig. 2. The measured and calculated fission rate of DU and EU.

rate in which the ^{235}U contribution is the main content that seems very sensitive to the distance and also to the polyethylene, this is caused by the change of slow neutron strength in the assembly which is sensitive to the average free path and the elastic scattering of carbon and hydrogen.

The tendency of fission rate distribution as a function of the distance from the core is well repeated by the calculation, and the calculated results to experimental ones (C/Es) are limited to the range of ±1.1 while the results in polyethylene and at the positions near the outer surface are beyond this range (see Fig. 3). The main reason for the overestimation around the polyethylene region is considered to be the contribution of neutron resonance while underestimation at the further point on the assembly was due to the contribution of room returned neutrons.

4.2 Fission counts

The ideal spherical shell model was used for calculation of ^{238}U or ^{235}U fission counts in each uranium shell, in which the detector channel was filled with uranium blocks while the others parts are the same as the model in calculation of ^{238}U or ^{235}U fission rate distribution. In this paper, the fission counts were calculated by using two methods: a) calculation of the U fission rate distribution at the same positions as the ones in experiment by using surface cross estimator and then the fission count was obtained with the integration method as in experiment; b) directly by using the three individual uranium shells as volume estimator.

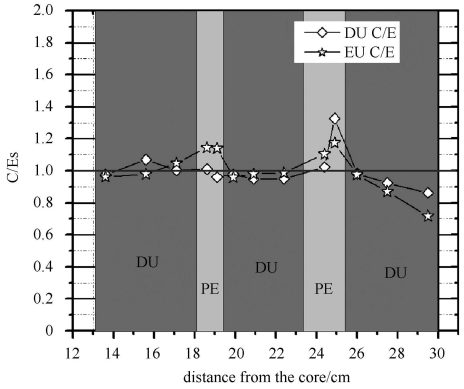

Fig. 3. C/Es of the fission rate of DU and EU in the assembly.

Figure 4 shows the comparison of the calculated results of ^{238}U or ^{235}U fission rate from method a) and that deduced from the experimental DU and EU fission rate distribution by Eqs. (4, 5). The calculation of ^{238}U fission rate agrees well with the experiment in the uranium shell while a large difference exists in the ^{235}U results in the uranium region. The similar comparison results between calculation and experiment fission count are indicated in Fig. 5, where the calculated fission count results with method a) and method b) and the ones from experimental measurement are presented. The discrepancy between the calculated fission count results from method a) and method b) was within 10% while a large discrepancy exists in the comparison of calculated results and experimental results. The difference between calculation model and experimental setup may be the main cause of the discrepancy between calculation and experimental results, especially as the fission chamber was not considered in the calculation. This caused the neutron spectrum to change from experiment to calculation especially in the PE region. For ^{238}U with threshold of about 1.0 MeV, the results obtained by fission chamber were almost the same as the ones with point estimator in the ideal spherical shell. On the other hand, for ^{235}U which is sensitive to slow neutrons, the large discrepancy between the experimental results and the calculation ones is indicated around the polyethylene region.

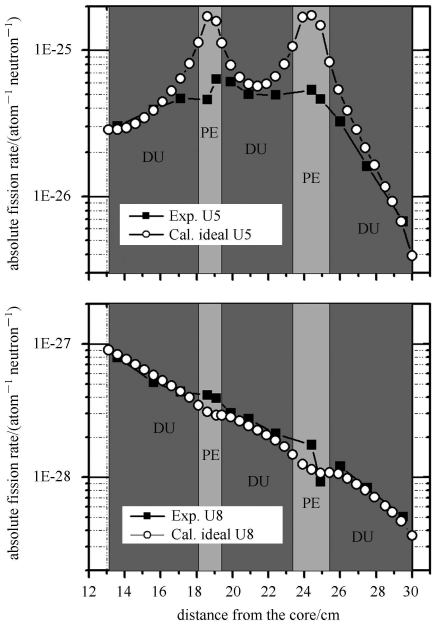

Fig. 4. The comparison of the calculated results of ^{238}U and ^{235}U fission rate with method a) and the experimental ones.

Table 3. The fission rate of ^{238}U and ^{235}U in two detector condition (atom^{-1}neutron^{-1}).

distance from the core/cm	f_5 in EU PMFC	f_5 in DU PMFC	f_8 in EU PMFC	f_8 in DU PMFC
13.6	2.80×10^{-26}	2.85×10^{-26}	7.65×10^{-28}	7.53×10^{-28}
18.6	6.42×10^{-26}	6.30×10^{-26}	2.92×10^{-28}	2.95×10^{-28}
24.9	5.35×10^{-26}	5.20×10^{-26}	1.04×10^{-28}	1.03×10^{-28}
13.6	4.67×10^{-27}	4.78×10^{-27}	4.20×10^{-29}	4.25×10^{-29}

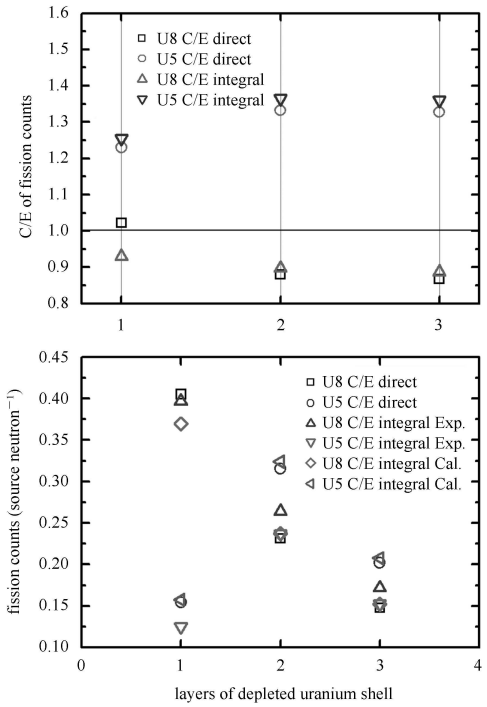

Fig. 5. The fission counts of ^{238}U and ^{235}U in an individual uranium shell.

4.3 The influence of materials in detectors

In the equation solution to get the ^{238}U fission rate from the depleted uranium and the ^{235}U fission rate from the enriched uranium results, it is assumed that the neutron field and the reaction probability of ^{238}U in DU and in EU is the same or almost the same. For getting the ^{235}U fission rate, the same condition was assumed. Because of the large cross sections of ^{235}U and slow neutrons, the neutron energy spectra at the same position may be different between depleted uranium condition and enriched uranium condition. To evaluate the influence of material in detectors on the fission rate results, especially when the enriched uranium detector was used to get the fission rate of ^{235}U, two different models were established, in which a depleted uranium fission chamber and enriched uranium fission chamber were included at the measuring position respectively. The fission rate of ^{238}U, ^{235}U were calculated in these two conditions, and the results were compared in Table 3. No significant difference about 3% was found.

5 Conclusion

The fission rate measurements have been conducted in the spherical assembly of depleted uranium and polyethylene. The experiment was introduced in detail and the results are presented with the calculated ones. The following facts are found: 1) The calculation repeated well the distribution of DU and EU fission rate in the experimental assembly. Except the fission rates at the two positions near the outer surface of the assembly and the results at the positions in the polyethylene area, all the calculations of fission rate for DU and EU agreed with the experimental ones within 10%. The main reason for the overestimation around the polyethylene region is considered to be the contribution of neutron resonance while underestimation at the further point on the assembly was due to contribution of room returned neutrons. 2) Because of the difference between calculation model and experimental setup, the discrepancy of fission counts for ^{238}U with threshold of about 1.0 MeV is about 10% while the overestimation of fission counts for ^{235}U which is sensitive to slow neutrons in the individual uranium shell is up to 35% around the polyethylene region. On the other hand, the difference between the calculated results of ^{238}U and ^{235}U fission counts from method a) and method b) was within 10%. To get more accurate experimental results of fission counts in each shell, more measuring points are needed, especially around the boundary of uranium and polyethylene. 3) More experiments are needed for the fission blanket concept design. The MCNP5 code and ENDF/B-V.0 data can be used in the neutronics design for the hybrid concept with neutron energy above 1.0 MeV. More validations would be considered to check the calculations.

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