The Fission and Capture Reaction for Quinta Experiment of E&T RAW Collaboration (December, 2012)

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(for collaboration "Energy & Transmutation of Radioactive Wastes")

Introduction

Feasibility of application of natural/depleted uranium or thorium without the use of U-235, as well as utilization of spent of fuel elements of atomic power plants is demonstrated based on analysis of results of known experiments, numerical and theoretical works [1-6]. The feasibility of application and economically competitive of depleted fuel depends on number of fission and number of capture reaction.

The main aim of this work is investigation of the number of U-238(n,g) reaction over number of U-238(n,f) reaction relation based on the experimental data and calculation results. The experimental data of U-238(n,f) and U-238(n,g) are from experiment made in JINR, December 2012.

The investigations described in this paper were performed within the framework of the scientific programme called "Investigations of physical aspect of electronuclear energy generation and atomic reactors radioactive waste transmutation using high energy beams of nuclotron JINR (Dubna)"-project "Energy & Transmutation of Radioactive Wastes"[E+T RAW].

The scientific description of the project, experimental methodology used for neutron and proton field properties investigation (activation and Solid State Nuclear Track (SSNT), He-3 detectors etc.), can be found in publications of the E+T RAW collaboration [1-6].

1. The Geometry and base parameter of experiment

The calculation model of 'Kwinta' assembly is based on the work [1]. The simulation of experiments are made by MCNPX 2.7 code[7] based on Monte Carlo method.

In Tab.1 are presented base average parameter of deuteron beam for three value of beam energy it is 2, 4 and 8 GeV.

Table 1: The total beam intensity

Beam Energy [Gev]	Total beam Intensity SSNT method (Belarus)	Total beam Intensity Activation method (Belarus)	Total beam Intensity Activation method (Ukraine)
2	$(3.0\pm0.3)10^{13}$	3.2013	3.0E13
4	$(3.1\pm0.3)10^{13}$	2.9313	2.7E13
8	$(8.6\pm0.9)10^{12}$	0.9613	0.89E13

The differences between SSNT and activation method are low are in the range of experimental errors.

The most difference between beam intensity is for beam energy 4 GeV. All results for 4GeV was normalized using beam intensity equals 3.1e13.

In the case of beam energy equals 2 MeV all results are normalized by beam intensity equal to 3.0e13.

In the case 8 GeV the measures results of Activation method was normalized by Ukraine total beam intensity 0.89e13 and the results of SSNT method was normalized by Belarus total beam intensity 8.6e12 from Tab.1.

Table 2: Beam profile. Detector place – measurent plate 0cm

Beam Energy [GeV]	Coordinates of beam center		FWHM, cm	
	Xc	Yc	$FWHM_X$	$FWHM_Y$
2	1.5±0.2	0.1±0.1	2.0±0.1	1.7±0.2
4	1.8±0.1	-0.3±0.1	1.5±0.2	1.1±0.1
8	0.9±0.1	0.1±0.1	1.0±0.1	1.3±0.1

2. The calculation method

To calculate number of U-238(n,f) or U-238(n,g) reaction Nthe following definition was used

$$N = \int_0^\infty \phi(E)\sigma(E)\rho dE \quad , \qquad (1)$$

where:

ϕ(E) density of neutron flux calculated by MCNPX 2.7 based on ENDF (Evaluated Nuclear Data File)

σ(E) –cross section from JENDL (Japanese Evaluated Nuclear Data Library)

$\rho-$density of U-238

To calculate neutron flux ϕ(E) we used geometry and materials presented in work[1] and deuteron beam parameters from table1.1.

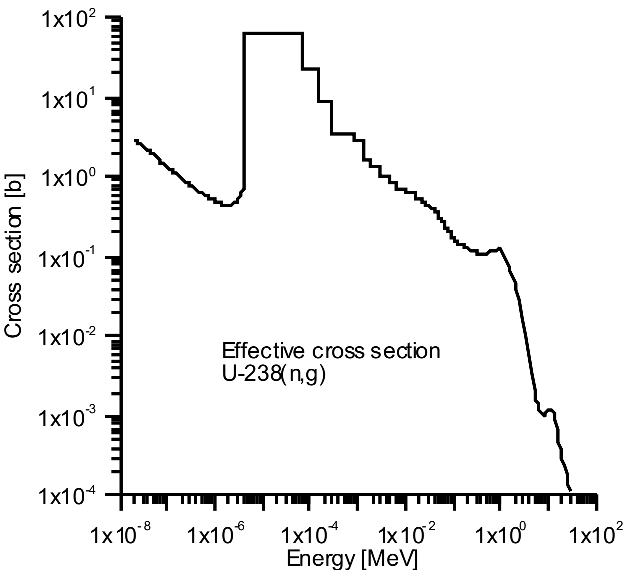

Fig. 2.1. The U-238(n,g) cross section from TENDL and JENDL Data Library. In the range (4.4e-6, 2.0e-2) MeV is effective average cross section.

In this calculation method was used :

- Beam profile parameters from Tab.2

-MCNPX 2.7 code [7]

a. ENDF – Evaluated Nuclear Data File[8]

b. Bertini[9,10]+ FLUKA[11]+Abla model[7]

- Formula (1) and the U-238(n,f) cross section from

JENDL[13] Data Library was used for fission of U-238 .

- The U-238(n,γ) cross section from TENDL[12] and JENDL[13] Data Library (see fig.2.1) was used for calculation of Pu-239 production. Additionally in the range (4.4e-6, 2.0e-2) MeV the cross section is effective cross section (see fig.2.1). The effective cross section we can calculate using the Evaluated Data Cross Section from ENDF[8], JENDL[13] Data Library. Additionally we calculate this cross sections using Neutron Resonance Parameter from [14] and Breite -Wigner formula. The differences of calculations are less than 5%.

3. Experimental and calculation results

The main aim of this section is presentation of distribution of ratio of SI$_{c/f}$=N(n,g)/N(n,f), where N(n,g) and N(n,f) means number of U-238(n,g) capture reaction and number of U-238(n,f) fission reaction. This ratio is named spectral index (see [1]). Please note, that spectral index does not depend on the normalization by total beam intensity. The experimental results from activation detectors and calculation results are presented in this section (figs.3.1-3.6). The experimental errors show on the fig.3.1 are based only on the uncertainty of total beam intensity (see Tab.1).

The spectral index SI$_{c/f}$is increasing function of distance R from assembly axis. The SI$_{c/f}$ function is equal to 1 for R equal about 5cm (fig.3.1). It means that in the volume region R<5cm dominate the U-238(n,f) reaction (fig.3.1). Whereas, for R>5cm dominate the capture reaction. In other words, the fission reaction dominate only around the axis of assembly it means in the volume region of deuteron beam.

It is clear because in this volume region are created high energy neutrons from spallation reaction which induce fission reactions.

The SI$_{c/f}$(z) is relatively complicated function of distance along the beam axis (fig,3,3-3,5). This function has two local extremum. The local minimal value are at z=26cm and local maximum are at z=52cm. This two extremum have both experimental data and calculation results. The SI$_{c/f}$(z) function achieve maximal value for z=0. It means that in this region dominate capture reactions. This is understandably because spallation high energy neutrons moves mainly along the beam direction [15]. The relative differences between experimental and calculation spectral indexes SI$_{c/f}$one can presented using the ratio of r$_{e/c}$=exp.SI$_{c/f}$/calc.SI$_{c/f}$=Exp.N(n,g)/N(n,f)/Calc. N(n,g)/N(n,f) (fig.3.6). If the ratio r$_{e/c}$ is greater than 1.0 means that:

- low energy fraction of neutron flux of experiment is greater than calculation or
- high energy fraction of neutron flux of experiment is less than calculation.

The function r$_{e/c}$(z) depends on beam energy and has relatively complicated form (fig.3.6). The greatest average values (about 1.27) achieve for beam energy 8GeV and lowest average values achieve for beam energy 2GeV(about 0.9).

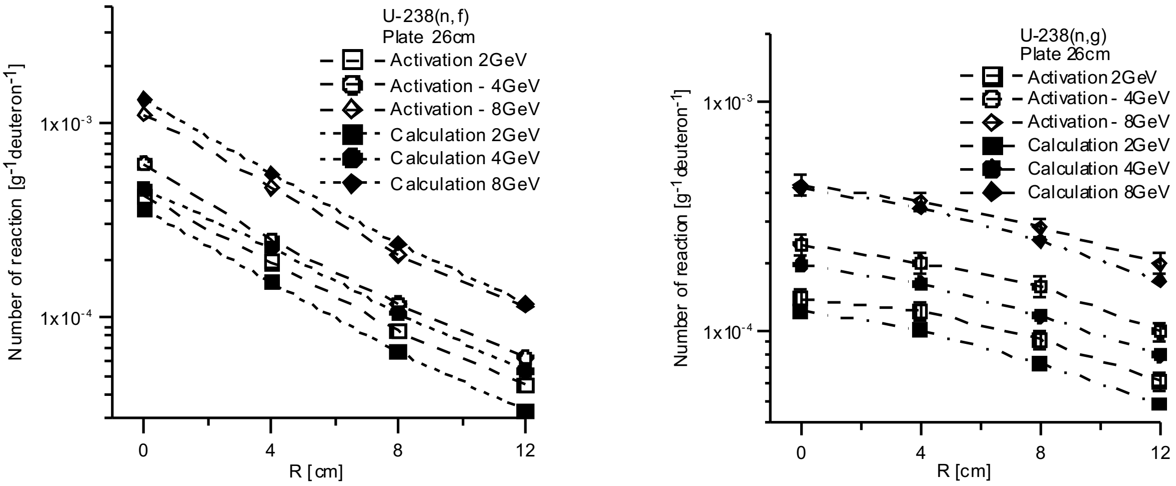

Fig.3.1 The radial distribution of number of reaction U-238(n,f) (left) and U-238(n,g) (right) reaction for measuring plate at 26cm. The relative experimental error is equal to10%.

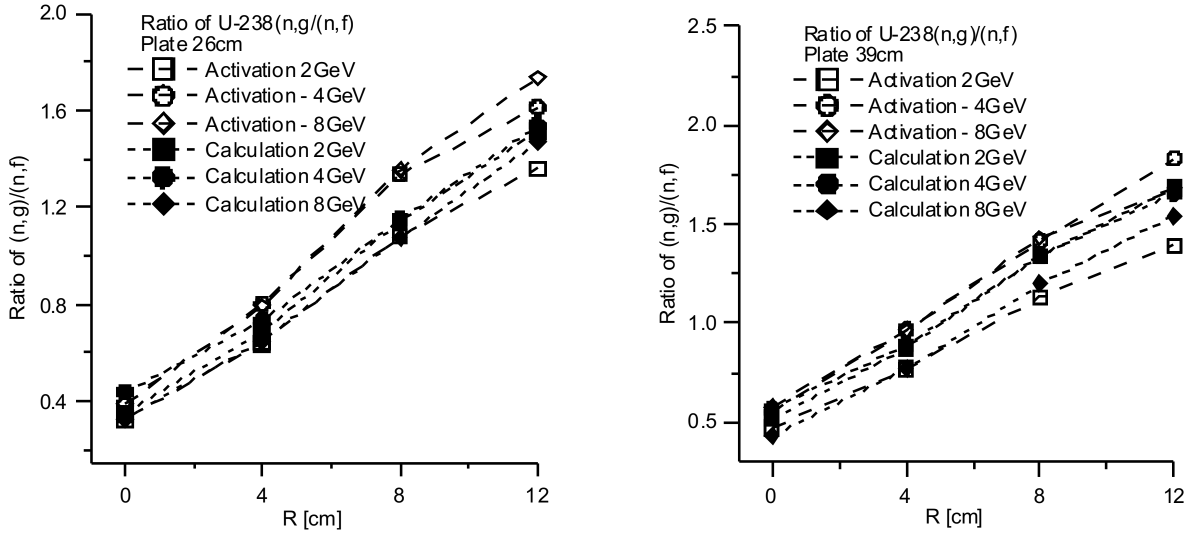

Fig.3.2 The spectral index $SI_{c/f}$ as a function of distance R from assembly axis. Measurement plane is placed at Z=26 (left) and 39cm (right).

The ratio $r_{e/c}$ for detectors at z=13, 26, 39, 52cm is equal to 0.91, 1.11 and 1.27 for beam energy 2, 4, 8 GeV respectively (fig.3.6). Most values of $r_{e/c}$(z) function is in the range 1.0±0.2.

Conclusion and remarks

The average differences of number of fission and capture reaction is about 20% (fig.3.1.). The ratio of spectral idexes $r_{e/c}$(z) gives a similar results (fig.3.6). Please note, that $r_{e/c}$(z) function does not depend on the normalization by total beam intensity (analogically as spectral index $SI_{c/f}$) . We can say that average relative differences of experimental data and calculations are equal to about 0.2. Most value of $r_{e/c}$(z) function is greater than 1(fig.3.6). It can means that low energy fraction of neutron flux of experiment is greater than calculation.

The reasons this situation can be in that simulation model not take into account all moderators in assembly for instance: pieces of paper and scotch tape around detectors. As we know this materials are a little and have very low mass but they are very good moderators and is placed directly on the detectors. The simulations this materials in calculation model increase the number of capture reactions.

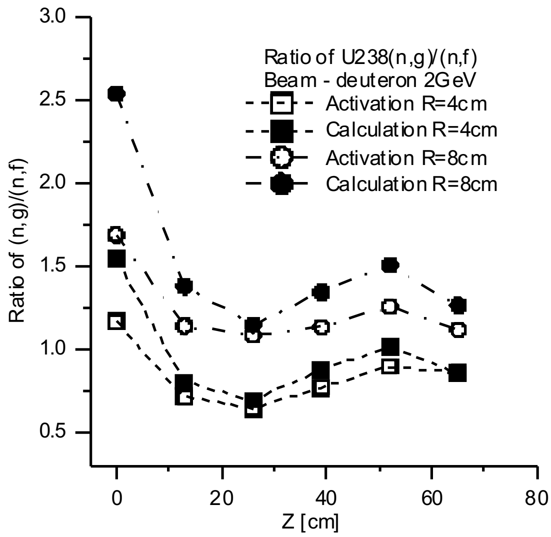

Fig.3.3 The spectral index $\mathrm{Si}_{c/f}$ as a function of distance along the assembly axis. Beam energy 2GeV.

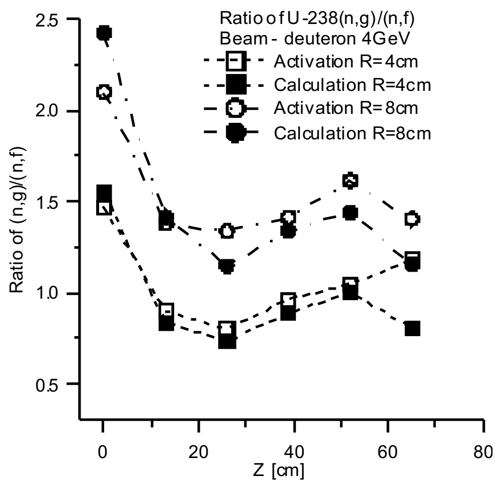

Fig.3.4. The same as in fig.3.3 but for beam energy 4GeV.

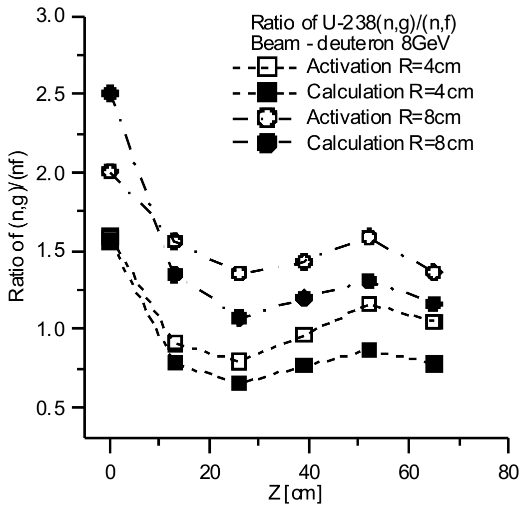

Fig.3.5. The same as in fig.3.2 but for beam energy 8GeV.

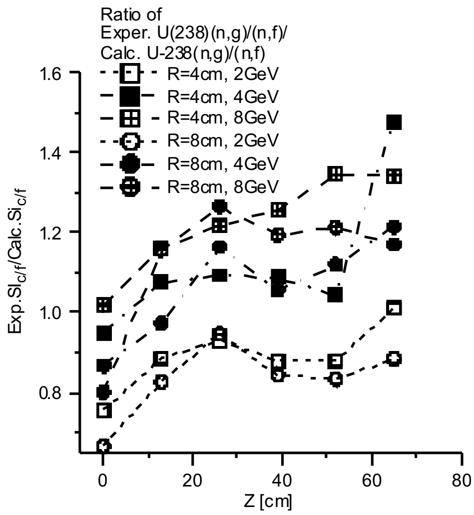

Fig.3.6. The ratio of spectral indexes as a function of distance along the assembly axis.

For instance, we made simulation of influencing the paper at thickness 0.015cm around detector at thickness 0.15cm. The mass ratio of paper to uranium detector was equal to about 1%. This simulation showed increasing the number of capture reaction and spectral index $\mathrm{SI}_{c/f}$ about 4% and simultaneously no changing the number of fission reaction. Strictly speaking the value of relative changing of fission was less than relative statistical calculation error=0.002. Including all moderators into calculation model around detectors we can improve agreement of calculation and experimental results.

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