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Benchmark experiment on molybdenum with graphite by using DT neutrons at JAEA/FNS

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HIGHLIGHTS

- A new benchmark experiment on molybdenum was conducted with DT neutron at JAEA/FNS.
- Dosimetry reaction and fission rates were measured in the molybdenum assembly.
- Calculated results with MCNP5 code were compared with the measured ones.
- A problem on the capture cross section data of molybdenum was pointed out.

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ABSTRACT

In our previous benchmark experiment on Mo at JAEA/FNS, we found problems of the (n,2n) and (n,γ) reaction cross sections of Mo in JENDL-4.0 above a few hundred eV. We perform a new benchmark experiment on Mo with a Mo assembly covered with graphite and Li_2O blocks in order to validate the nuclear data of Mo in lower energy region than in the previous experiment. Several dosimetry reaction and fission rates are measured and compared with calculated ones with the MCNP5-1.40 code and the recent nuclear data libraries, ENDF/B-VII.1, JEFF-3.2, and JENDL-4.0. It is suggested that the (n,γ) reaction cross section of ^{95}Mo should be larger in the tail region below the large resonance of 45 eV in these nuclear data libraries.

1. Introduction

We are planning and developing a new fusion neutron source in Japan. Type 316L stainless steel (SS316L) will be used as a material of a beam duct and so on in the facility. SS316L contains a few percent of molybdenum (Mo). Previously, we performed a benchmark experiment on Mo in order to validate the nuclear data of Mo at the Fusion Neutronics Source (FNS) facility of Japan Atomic Energy Agency (JAEA) and found problems on the (n,2n) and (n,γ) reaction cross section data of Mo in JENDL-4.0 [1]. Then the nuclear data of Mo only above a few hundred eV were investigated, because there were few neutrons of lower energy in the Mo assembly. In this study, we perform a new benchmark experiment with the Mo assembly covered with graphite and lithium oxide (Li_2O) blocks in order to validate the nuclear data of Mo in lower energy region.

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2. Experiment

Fig. 1 shows an experimental configuration. A rectangular Mo assembly, the size of which is 253 mm × 253 mm × 354 mm, is covered overall with 152 mm thick reactor grade graphite blocks [2]. Furthermore, the assembly is covered with 51, 101 and 101 mm thick Li_2O blocks [3] around the front, side and rear surfaces, respectively. The graphite blocks produce lower energy neutrons and the Li_2O ones eliminate room-returned background neutrons at measurement points inside the assembly. The experimental results of the graphite and Li_2O benchmark experiments are in very good agreement with calculated ones as reported in refs. [2,3]. Therefore, it is considered that nuclear data of carbon, lithium and oxygen are good. Various foils are placed along the central axis in the assembly in order to measure dosimetry reaction rates; Nb and Al (10 mmϕ × 1 mm), In (10 mm × 10 mm × 1 mm), Au (10 mm × 10 mm × 0.0025 mm), W (10 mm × 10 mm × 0.025 mm), and Mo (10 mm × 10 mm × 0.9 mm). The assembly is placed at a distance of 103 mm from a DT neutron source. After 6-h irradiation in which the total yield is 2.81×10^{15} neutrons, γ-rays from

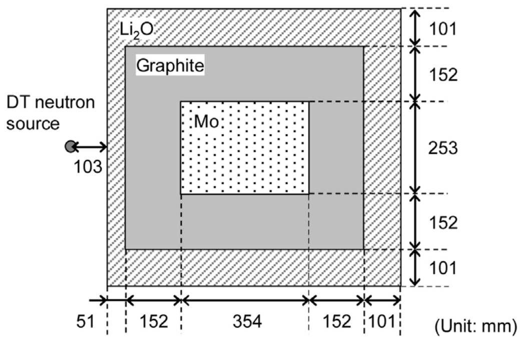

Fig. 1. Experimental configuration.

the foils are measured with high-purity germanium (HPGe) detectors and the dosimetry reaction rates of the ^{93}Nb(n,2n)^{92m}Nb, ^{27}Al(n,α)^{24}Na, ^{115}In(n,n′)^{115m}In, ^{197}Au(n,γ)^{198}Au, ^{186}W(n,γ)^{187}W, and the sum of the ^{98}Mo(n,γ)^{99}Mo and ^{100}Mo(n,2n)^{99}Mo reactions are derived by the foil activation method. Fission rates of ^{235}U and ^{238}U are also measured with micro fission chambers (MFCs) inserted from the rear through a hole of 21 mm in diameter by replacing the ordinary blocks with ones with the hole. The specification of the MFCs is described in ref. [2]. The experimental error consists of the neutron yield error (3%), the efficiency error of the HPGe detectors (2%), the self-absorption correction error in γ-ray measurement (1%), the effective number error of ^{235}U and ^{238}U in the MFCs (4%), and the statistical error of the measured count (mostly around 1%, 6.6% at most).

3. Analysis

The experimental configuration including the activation foils is modeled precisely and the reaction and fission rates are calculated by using the track-length estimator of cell flux (F4 tally) in the Monte Carlo transport code MCNP5-1.40 [4] with the recent nuclear data libraries of ENDF/B-VII.1 [5], JENDL-4.0 [6] (FENDL-3.1b [7]), and JEFF-3.2 [8]. JENDL dosimetry file 99 (JENDL/D-99) [9] is used as a response function in the calculation of the reaction and fission rates except for the ^{98}Mo(n,γ)^{99}Mo and ^{100}Mo(n,2n)^{99}Mo reactions. For these two reactions, reaction rates are calculated with the cross section data in the same general purpose file as that used in the neutron transport calculation, because these are not included in JENDL/D-99. The calculated reaction and fission rates are compared with the experimental ones. Statistical errors in the MCNP calculation are around 1%.

4. Results and discussion

Fig. 2 shows ratios of the calculated values to the experimental ones (C/Es) for the reaction rates of the ^{93}Nb(n,2n)^{92m}Nb, ^{27}Al(n,α)^{24}Na, ^{115}In(n,n′)^{115m}In, ^{197}Au(n,γ)^{198}Au, ^{186}W(n,γ)^{187}W, and the sum of the ^{98}Mo(n,γ)^{99}Mo and ^{100}Mo(n,2n)^{99}Mo reactions and the fission rates of ^{235}U and ^{238}U. The depth position at the front boundary surface between graphite and Mo is taken as 0 cm in Fig. 2. The C/Es for the reaction rates of the ^{93}Nb(n,2n)^{92m}Nb, ^{27}Al(n,α)^{24}Na, ^{115}In(n,n′)^{115m}In, and the sum of the ^{98}Mo(n,γ)^{99}Mo and ^{100}Mo(n,2n)^{99}Mo reactions and the fission rate of ^{238}U show a similar tendency to those in the previous experiment [1]. However, the C/Es for the reaction rates of the ^{197}Au(n,γ)^{198}Au and ^{186}W(n,γ)^{187}W reactions and the fission rate of ^{235}U show different tendencies. Comparisons between the C/Es for the reaction rates of the ^{93}Nb(n,2n)^{92m}Nb, ^{197}Au(n,γ)^{198}Au, and ^{186}W(n,γ)^{187}W reactions in the present and previous experiments are shown in Fig. 3, as an example. The error bars include both calculation and experimental errors in Fig. 3. Fig. 4 shows the neutron spectrum and the reaction rate spectra of the ^{197}Au(n,γ)^{198}Au and ^{186}W(n,γ)^{187}W reactions at the depth of 20.7 cm calculated with JENDL-4.0 by using

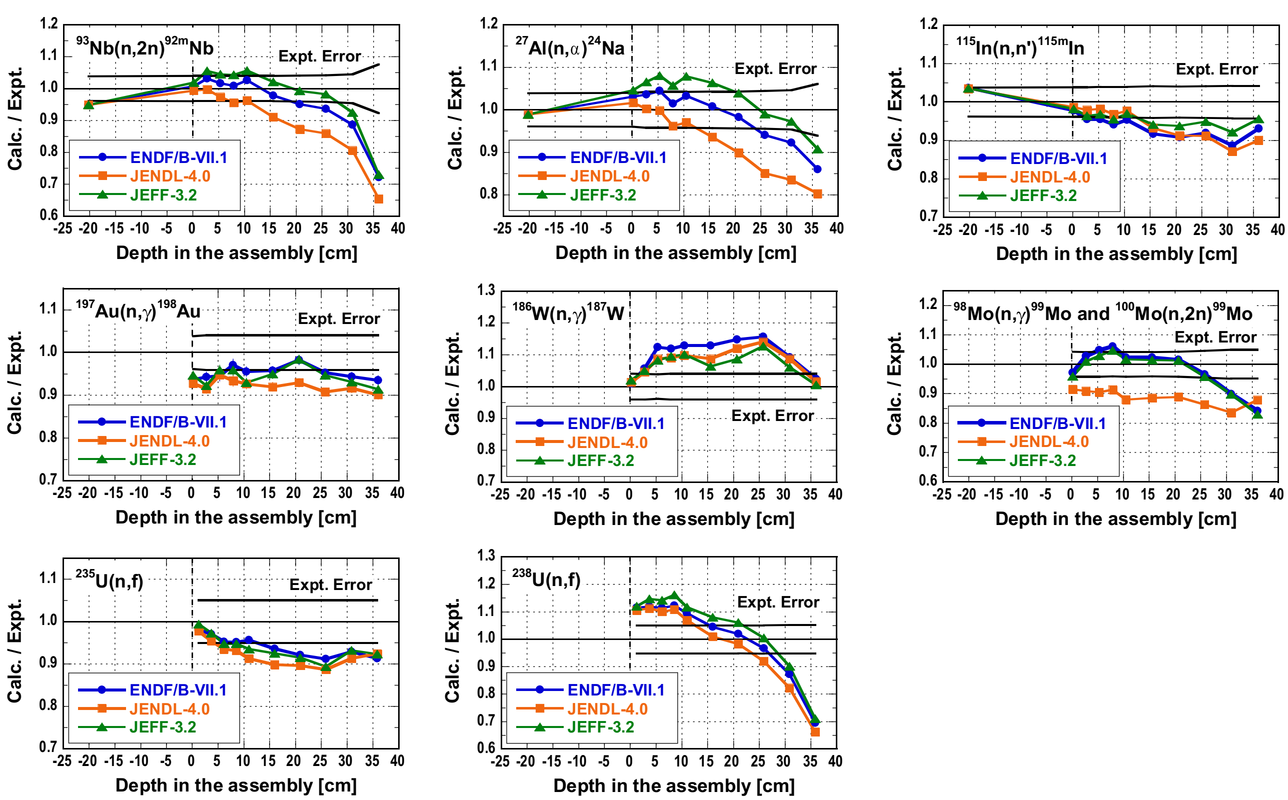

Fig. 2. C/Es for reaction and fission rates.