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Calculations of the thermal and fast neutron fluxes in the Syrian miniature neutron source reactor using the MCNP-4C code

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ABSTRACT

The MCNP-4C code, based on the probabilistic approach, was used to model the 3D configuration of the core of the Syrian miniature neutron source reactor (MNSR). The continuous energy neutron cross sections from the ENDF/B-VI library were used to calculate the thermal and fast neutron fluxes in the inner and outer irradiation sites of MNSR. The thermal fluxes in the MNSR inner irradiation sites were also measured experimentally by the multiple foil activation method (^{197}Au (n, γ) ^{198}Au and ^{59}Co (n, γ) ^{60}Co). The foils were irradiated simultaneously in each of the five MNSR inner irradiation sites to measure the thermal neutron flux and the epithermal index in each site. The calculated and measured results agree well.

1. Introduction

MNSR (miniature neutron source reactor) is a tank-in-pool type research reactor (CIAE, 1993). It is one of the low-power research reactors, which use highly enriched uranium as fuel, light water as moderator, and beryllium as reflector. Heat generated in the core is removed by the natural convection process. The annulus beryllium reflector surrounds the reactor core. The reactor also has a bottom beryllium plate and a set of top beryllium shims. MNSR has 10 irradiation sites: five inner sites are distributed uniformly inside the annulus beryllium reflector, while five outer sites surround the annulus beryllium externally. The inner irradiation sites #1, #2, #3 and #4 in the reactor are connected to a rabbit system Type B, and the inner irradiation site #5 is connected to a rabbit system Type A. The reactor nominal power is 30 kW.

The Monte Carlo technique implemented in the MCNP-4C code can efficiently model fairly complicated geometries, such as the Syrian MNSR core (Briesmeister, 1997). Operators and scientists who work in the neutron activation analysis laboratory and in the reactor physics group need a complete map of the thermal and fast neutron fluxes in the most important positions in the MNSR reactor core. These positions are the inner and the outer irradiation sites because they are the only places in the reactor core used for irradiation of the unknown samples and production of medium- and short-lived isotopes. To create this map, the MNSR reactor has been modeled in three dimensions with the MCNP-4C code.

In a reactor, certain stable nuclides undergo neutron-induced transmutations. The activity of the radioactive products can be measured with an appropriate counting system, and the neutron flux can then be determined. The energy of neutrons in a reactor ranges widely from the fission energy down to the thermal region. Responses of many detectors are well suited for measuring the neutron fluxes in certain parts of this energy range.

This paper also reports Westcott (thermal and epithermal) neutron fluxes and the epithermal indexes for all the inner irradiation sites of the MNSR reactor. The values were obtained by the multiple foil activation method using the reactions ^{197}Au (n, γ) ^{198}Au and ^{59}Co (n, γ) ^{60}Co.

The fast neutron flux in the MNSR inner irradiation site #1 had been measured before with ^{238}U foils covered with cadmium filters (Khattab, 2007).

2. Methodology

2.1. The MCNP-4C model of the Syrian MNSR reactor

The Monte Carlo simulation of the Syrian MNSR reactor was carried out with the MCNP-4C code and continuous energy cross sections from the ENDF/B-VI library. This 3D model comprises all the reactor components, namely: 347 fuel rods, three dummy rods, four tie rods, control rod with its guide tube, reactor vessel, top and bottom aluminum alloy grids, annular beryllium reflector, bottom beryllium reflector, five inner and five outer irradiation

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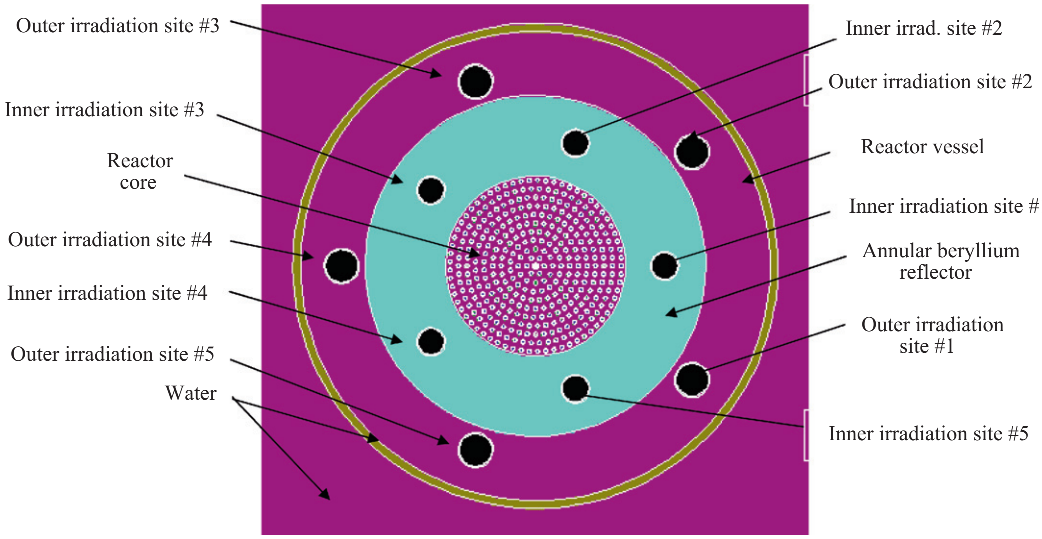

Fig. 1. Horizontal cross section of the reactor used in the model.

sites, and the reactor pool (Fig. 1). All the fuel elements, beryllium reflector and the irradiation sites are represented as cylinders of appropriate materials and dimensions positioned in their exact locations. The calculation used the full continuous energy cross sections for 20 °C available from the MCNP-4C library. The MCNP-4C input file consisted of 210 cycles, including 10 inactive cycles, with 5000 histories per cycle. We used energy ranges 0–0.625 eV for thermal neutrons and 0.5–10 MeV for fast neutrons. The diameters of the inner and outer irradiation sites were 3.5 and 4.2 cm, respectively. The centers of the inner and outer irradiation sites were at 16.5 and 24.8 cm from the reactor center, respectively. The depths of both the inner and the outer irradiation sites were 18 cm. The top 3 cm of each site contained only air necessary for inserting and ejecting the irradiated capsules. Therefore, the values of the thermal and fast neutron fluxes in the inner irradiation sites were extracted from the output file at a depth of 15 cm from the top of the reactor core and at a radius of 16.5 cm from the core center. The corresponding values for the outer irradiation sites were 15 and 24.8 cm, respectively.

2.2. Determination of the Westcott conventionality thermal neutron flux using multiple foil method

According to Westcott convention (IAEA, 1970), the effective cross section σ in a thermal reactor can be written as

$$\sigma = \sigma_0(g + rs), \tag{1}$$

where σ_0 is a 2200 m s^{-1} cross section, g is the Westcott tabulated factor, and r is the epithermal index. The value s is defined as

$$s = \frac{1}{\sigma_0}\sqrt{\frac{4}{\pi}\frac{T}{T_0}}I_0', \tag{2}$$

where T_0 is the room temperature, T is the temperature in the inner irradiation site (45 °C) and $I_0' = I_0 - 0.45\sigma_0$ (I_0 is the resonance integral). The temperature-dependent parameters g and s characterize the deviations of the cross sections from the $1/v$ value (v is the neutron speed) in the thermal and the epithermal regions, respectively.

According to IAEA (1970), the reaction rate can be written as

$$R = nv_0\sigma_0\left(gG_{th} + r\sqrt{\frac{T}{T_0}}s_0G_{epi}\right), \tag{3}$$

where n is the neutron density for both the thermal and the epithermal neutrons and v_0 is 2200 m s^{-1}. Quantities G_{th} and G_{epi} are self-shielding corrections for the thermal and epithermal neutrons, which can be calculated using the following equations:

$$G_{th} = \frac{1 - 2E_3(x)}{2x}, \tag{4}$$

$$G_{epi} = \frac{1}{\sqrt{1 + 2\mu_{a0}\delta}}. \tag{5}$$

In Eq. (4), $x = \Sigma_a t$, where Σ_a is the macroscopic absorption cross section for 2200 m s^{-1} and t is the detector thickness. In Eq. (5), μ_{a0} is the mass absorption coefficient at the peak of the resonance, and δ is the surface mass loading.

The parameter s_0 can be calculated from equation

$$s_0 = \frac{1}{\sigma_0}\sqrt{\frac{4}{\pi}}I_0'.$$

The total number of counts over a period of time t_c can be expressed as

$$C = \frac{\varepsilon\gamma R}{\lambda}(1 - e^{-\lambda t_i})e^{-\lambda t_d}(1 - e^{-\lambda t_c}), \tag{6}$$

where ε is the detector efficiency, γ is the branching ratio, t_i is the irradiation time, t_d is the cooling time, and t_c is the counting time (IAEA, 1970).

With the total number of counts measured, the activation rate R can be calculated using Eq. (6). Substituting the activation rate in Eq. (3) results in one equation with two unknowns: one, the thermal and the epithermal neutron fluxes combined (nv_0) and, the other, the epithermal index (r). Therefore, measurements of another foil activity are necessary to produce the lacking second equation. This is known as the two-foil induced activation method.

Activation rates of two foils irradiated simultaneously can be written, using Eq. (3), as

$$R_1 = nv_0\sigma_{01}\left(g_1G_{th1} + r\sqrt{\frac{T}{T_0}}s_{01}G_{epi1}\right) \quad \text{and} \tag{7}$$

$$R_2 = nv_0\sigma_{02}\left(g_2G_{th2} + r\sqrt{\frac{T}{T_0}}s_{02}G_{epi2}\right). \tag{8}$$