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"Journal of Economics and Social Sciences"

Hybrid Simulation of a Radioisotope Neutron Source based on a Mixture of Stable Crystals of Am and Be Oxides

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Abstract

In this paper, a procedure was proposed for calculating the neutron yield and the spectrum of the Am-Be source for different granular size of the source mixture. With the combined use of the calculation codes SOURCES-4C and MCNP5, computational models of the source differing in the structure of the mixture and the energy spectrum were presented. By calculating of neutron transport, an analysis of the appearance and loss of neutrons in the sources and in the steel shell of the source was carried out. As a result of the work, it became clear how the grain size of the source mixture affects the neutron yield.

Keywords: Spectra, AmBe-source, neutron current, Monte Carlo, modelling;

1. Introduction

For isotope synthesis, process of irradiating a target with a flux of fast or thermal neutrons, which can be obtained using special neutron sources, is often used. Radioisotope neutron sources are widely used in various fields of activity and are manufactured by mixing an emitter element (Pu, Am, Ra) and a target element (Be, B). Currently, AmBe sources are most often used, which, due to the long half-life of ^{241}Am (432.8 years), have a stable neutron release. These sources are compact and do not require bulky gamma radiation shielding. The energy spectra of typical (α, n) - sources, as a rule, can vary depending on the source design, composition and presence of impurities [1, 2]. In this regard, determination of the spectrum of neutron sources in a wide energy range is not an easy task. Its solution can be obtained using the Monte Carlo method, known as the statistical test method.

2. Materials and Methods

Modeling of (α, n) - sources was carried out using the calculation codes SOURCES-4C and MCNP5. The SOURCES-4C code allows the calculation of the neutron release from the (α, n) - reaction in simplified planar geometry and in heavy metallic materials. Since neutrons generated in (α, n) - reactions have high energies (E_{avg} = 4.2 MeV, E_{max} = 12 MeV), there is a possibility of (n, 2n) - reaction occurring on beryllium nuclei, which can affect the neutron release. The contribution of this neutron reaction can be taken into account when simulating particle transfer process with the MCNP5 calculation code.

Due to the lack of information on the (α, n) - reaction in MCNP5, the results of calculating the neutron release and energy spectrum in SOURCES-4C are used as input data for modeling the transfer of secondary particles (neutrons) in MCNP5.

To accommodate the neutron AmBe source, an Amersham X-14 [1] capsule was used, which linear dimensions correspond to 60x30 mm. According to [1], the source contains 23 g of beryllium and 1.85 g of americium dioxide. The capsule casing is made of stainless steel. The spectra for a homogeneous (0 μm) and heterogeneous finely dispersed mixture of AmO_2 and Be, were selected as the initial spectra [2,4], which were calculated using the SOURCES-4C code [6].

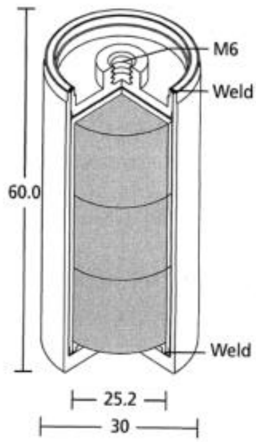

Fig. 1. X14 capsule [2]

3. Results and Discussion

At the next stage, an isotropic source consisting of a homogenized mixture of americium dioxide and beryllium with a density of 0.987 g/cm^3 was specified in MCNP5 [3]. The neutron spectrum of the source was divided into 52 energy groups. The initial power of the source is $1.5 \cdot 10^7$ n/s.

The calculated spectrum resulted from neutron transfer is close to the ISO spectrum recommended for such studies [5]. The observed increase in the neutron current in low-energy region is associated with moderation and diffusion.

As a result of the particle transfer, neutron power increased by 6.48%, of which 97.68% - as a result of the (n, 2n) reaction; 2.32% - result of prompt fission. The energy loss for moderation was 0.18 MeV per source particle.

A similar calculation of the neutron release from the source surface was carried out for a mixture with a grain size of 8 μm, which is typical to a condition after pressing. For this purpose, source spectrum in the calculation model was changed to the corresponding one. The initial power of the source is $7.37 \cdot 10^6$ n/s. The calculated spectrum resulted from neutron transfer is shown in Figure 2. In this option, neutron power of the source increased by 5.09%, of which 97.57% - as a result of the reaction (n, 2n); 2.43% - result of prompt fission. The energy loss for moderation was 0.19 MeV per one particle of the source.

The contribution to the neutron current from the ends for both models was 17.79%.

To assess the effect of the steel wall on neutron transfer, an additional calculation of the neutron current through the inner wall of the casing was performed. An increase in the neutron flux is observed in the range 1.4-1.82 MeV as a result of the neutron moderation with higher energy. The total loss of particles as a result of neutron capture in steel was 2.79% of the total neutron current for a homogeneous source and 2.63% for a source with a grain size of 8 μm.

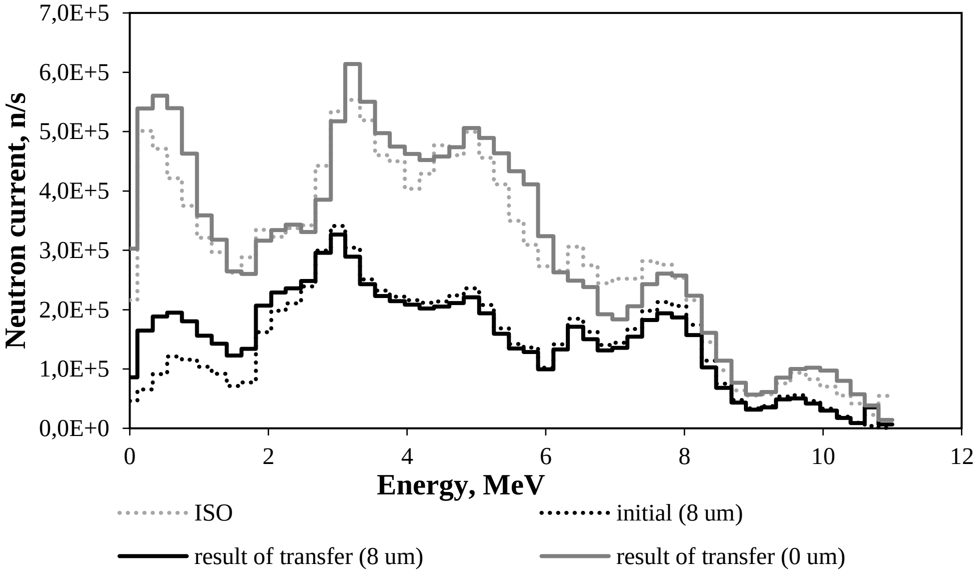

Fig. 2. Comparison of the recommended and calculated (initial) neutron spectra with the transfer results

As it can be seen in Figure 2, with an increase in the AmO_2 grain size, the neutron spectrum less than 4 MeV shifts towards lower energies, which indicates an increase in (α, n) -reactions by $^{17,18}O$ and a shortage of alpha particles to Be nuclei.

4. Conclusion

An analysis of the calculated data shows that the energy spectrum of a source consisting of a dispersed mixture, the grain size of which is 8 μm, has an integral neutron release similar to the homogeneous version of the source, but differs when passing from relative units to absolute ones. This difference can be explained by the fact that with an increase in the grain diameter of the mixture, the effect of self-absorption of alpha-particles in the interior of the grain appears due to their short path length. These calculations show that quality of pressing and mixing, i.e. the mixture structure should be taken into account when calculating, since it affects the result.

The obtained result shows that for sources with the largest diameter of the mixture, it is possible to use the lighter protection of the capsule from neutrons during its transportation and long-term storage. This will have an economic effect when a large number of capsule sources are used together.

For possibility of increase the neutron yield, the stable crystals of intermetallic compounds of $AmBe_{13}$ can be used.

References

1. Amersham/Searle, 1976. Neutron sources 241Am-Be and 252Cf. Tech. Bull. 76-77.
2. Bedenko, S.V., Vlaskin, G.N., Ghal-Eh, N., et al. (2020). Nedis-Serpent simulation of a neutron source assembly with complex internal heterogeneous structure. *Applied Radiation and Isotopes*. 160. [109066]. https://doi.org/10.1016/j.apradiso.2020.109066
3. Briesmeister, J.F. (2000). MCNP – A General Monte Carlo N-Particle Transport Code. LANL Report LA-13709-M. Los Alamos.
4. Ghal-Eh N., Rahmani, F., Bedenko, S.V. (2019). Conceptual design for a new heterogeneous ^{241}Am-9Be neutron source assembly using SOURCES4C-MCNPX hybrid simulations. *Applied Radiation and Isotopes*. Vol. 153. [Available at https://doi.org/10.1016/j.apradiso.2019.108811] [Accessed on 12.11.2020]
5. International Organization for Standardization Reference Neutron Radiations-Part 1, Characteristics and Methods of Production (2001) ISO 8529–1 (Geneva: ISO)].
6. Wilson, W.B., Perry, R.T., Charlton, W.S. (2009). Parish Sources: a code for calculating (alpha,n), spontaneous fission, and delayed neutron sources and spectra Prog. *Nucl. Energy*, 51 (4–5). pp. 608-613 [Available at https://doi.org/10.1016/j.pnucene.2008.11.007] [Accessed on 12.11.2020]