ISOTOPIC COMPOSITION AND RADIATION CHARACTERISTICS OF THE 4TH GENERATION REACTOR SPENT FUEL

Shamanin I. V., Bedenko S. V., Knyshev V. V., Ukrainets O. A., Zor'kin A. I.

National Research Tomsk Polytechnic University, Tomsk, Russia

E-mail: bedenko@tpu.ru

Neutron transfer computational methods are used in reactor physics, in process control of nuclear fuel manufacturing and reprocessing, in determination of spent fuel burn-up, in biological protecting design as well as in perfection of nuclear materials handling procedures and regulations. The computational method which is based on the draw of neutron interaction probability is called Monte-Carlo method [1].

There are a lot of specialized software applications and computational codes implementing Monte Carlo method. The following packaged programs: MMKKENO and MMKC (State Scientific Center of the Russian Federation, Institute for Physics and Power Engineering), MCU (National Research Center «Kurchatov Institute», Russia), MCNP (USA), KENO-3D (USA), MONK (Great Britain) and other, are internationally-recognized. These programs allow getting high precision results due to creation of three-dimensional geometry, simulating the object and the process of real neutron transfer in the matter.

High accuracy of the verified computational codes based on Monte Carlo method is achieved by consumption of huge computing resources and is determined mainly by the fault in interaction cross section of neutrons with the matter nuclei. Algorisms based on group and subgroup approaches used when solving Boltzmann equation (neutron-transport equation) [2], do not require big computational resources. Although, the accuracy of the desired functional ($\Phi(\mathbf{r},E,t)$, $\Sigma(\mathbf{r},E)$, k_{eff}) in such solutions is lower, multigroup diffusion approximation is considered suitable and is currently used in Russian and foreign programs. For example, there is a multigroup module in MCNP and many other programs. Using additional computational procedures such approaches significantly simplify the principle simulation stage, which consists in transport equation solution. In some cases they provide an accurate and reliable result [3]. In the present paper computation of neutron generation intensity and neutron spectrum distribution is an additional procedure.

It should be noted that in the solution of the transport equation (irrespective of the solution method) in subcritical systems neutron spectral distribution has the determining value. When the number of calculated iteration increases, such approach makes it possible to improve the neutron-physical functions.

Neutrons in spent nuclear fuel are created due to spontaneous and induced fission and as a result of (α,n)-reactions on light elements nuclei. In case of the traditional UO_2 (α,n)- reaction proceeds on oxygen. For burnup fuel of the researched reactor there are some other nuclei besides oxygen on which such reaction is possible, namely: C –carbon, PyC – pyrolytic carbon, $TiSiC_2$ – titanium silicon carbide, SiC_2 – silicon carbide.

The normalized neutron spectrum on carbon is essentially higher, than on oxygen [4,5]. Consequently, for creating an efficient instrument used for developing procedures and regulations of spent fuel handling (α,n)-reaction should be treated separately as an independent task.

Computation of intensity and neutron spectrum of spontaneous and induced fission is not a difficult task. But calculation of (α,n)-neutron spectrum is a more complicated task which is solved using such specialized programs as NEDIS-2.0, SOURCE-4C and other.

At present the developers of these programs pay great attention to accuracy of the neutron yield computation. However, existing high precision experimental data show significant variation (10 % and even more) with calculated estimates carried out using NEDIS and SOURCE.

In the present paper spectral neutron distribution was obtained using approximation of an extensive list of experimental (Experimental Nuclear Reaction Data − EXFOR) and computation data (Evaluated Nuclear Structure Data File – ENSDF).

REFERENCES

1. Bell D, Glesston S M 1974 Teorija jadernyh reaktorov [Theory of nuclear reactors]. Atomizdat, 1974. (in Russian).

2. Abagyan L P, Bazazyanets N O, Nikolaev M ., Tsibulya A. M. 1981 Gruppovye konstanty dlja rascheta jadernyh reaktorovMoscow [Group constants for the calculation of nuclear reactors]. *Energoatomizdat* (in Russian).

3. Plevaka M N, Bedenko S V, Gubaidulin I M, Knyshev V V 2015 Neutron-physical studies of dry storage systems of promising fuel compositions. *Bulletin of the Lebedev Physics Institute* **42**, 240−243.

4. Heaton R, Lee H, Skensved P, Robertson B C 1989 Neutron production from thick-target (α, n) reactions. *Nucl. Instr. Meth. Phys. Res.* A **276**, 529–538.

5. Murata T, Shibata K, et al 2002 Japanese Evaluated Nuclear Data Library Version 3 Revision-3: JENDL-3.3. *J. Nucl. Sci. Technol.* **39**, 76–79.