MULTIGROUP METHOD FOR CALCULATING THE SPECTRUM OF THE NEUTRON FLUX DENSITY OF RBMK-1000

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The article describes a method for calculating the spectrum of the multigroup neutron flux. The paper shows the basic rules and principles of multigroup calculation, presented the basic formulas. Just article shows the calculated neutron spectrum RBMK-1000 using multigroup method. The calculation is performed within the job of teaching and research students TPU.

Multigroup diffusion equations system for critical nuclear reactor (stationary problem) has the following form [1, 2]:

$$D^{(i)}\Delta\Phi^{(i)} - \Sigma_a^{(i)}\Phi^{(i)} - \sum_{k=i+1}^{26}\Sigma_R^{i\to k}\Phi^{(i)} + \sum_{k=1}^{i-1}\Sigma_R^{k\to i}\Phi^{(k)} + \varepsilon^{(i)}\sum_{k=1}^{26}\nu_f^{(k)}\Sigma_f^{(k)}\Phi^{(k)} = 0 \qquad (1)$$

where i - number of the group for which the equation is written; k - group number; $D^{(i)}$ - neutron diffusion coefficient of i- th group; $\Phi^{(i)}, \Phi^{(k)}$ - neutron flux density in the respective groups; $\Sigma_a^{(i)}$ - macroscopic neutron absorption cross section of i-th group; $\Sigma_R^{i\to k}, \Sigma_R^{k\to i}$ - macroscopic neutron cross section transition of the i-th in the lying below k-th (from lying above k-th in the considered i-th) group, respectively ; $\varepsilon^{(i)}$ - the probability for fission neutron to have a direct access to the i-th group; $\nu_f^{(k)}$ - the average number of neutrons per fission; $\Sigma_f^{(k)}$ - macroscopic fission cross section for neutrons of k-th group.

According to the equation of the critical reactor in the diffusion-age approximation, the first term in the equation (1), which describes the neutron leakage from the core, will be determined by the relation:

$$D^{(i)}\Delta\Phi^{(i)} = -D^{(i)}B^2\Phi^{(i)} \qquad (2)$$

where B^2 - geometrical parameter.

To organize the iterative process when solving the system of equations in the problem a system of equations for determining the neutron flux density of the following form should be set up:

$$\Phi_j^{(i)} = f(\Phi_{j-1}^{(1)}, \Phi_{j-1}^{(2)}, ..., \Phi_{j-1}^{(k)}, ..., \Phi_{j-1}^{(26)}),\ k \neq i$$

where j - number of iterations, starting with the first.

To this end, a system of multigroup diffusion equations using the relation (2) should be brought to the form:

$$-D^{(i)}B^2\Phi^{(i)} - \Sigma_a^{(i)}\Phi^{(i)} - \sum_{k=i+1}^{26}\Sigma_R^{i\to k}\Phi^{(i)} + \sum_{k=1}^{i-1}\Sigma_R^{k\to i}\Phi^{(k)} + \varepsilon^{(i)}\sum_{\substack{k=1\\k\neq i}}^{26}v_f^{(k)}\Sigma_f^{(k)}\Phi^{(k)} + \varepsilon^{(i)}v_f^{(i)}\Sigma_f^{(i)}\Phi^{(i)} = 0 \quad (3)$$

Expressing the flux density in the i-th group from (3) we obtain:

$$\Phi_j^{(i)} = \frac{\varepsilon^{(i)}\sum\limits_{\substack{k=1\\k\neq i}}^{26}v_f^{(k)}\Sigma_f^{(k)}\Phi_{j-1}^{(k)} + \sum\limits_{k=1}^{i-1}\Sigma_R^{k\to i}\Phi_j^{(k)}}{D^{(i)}B^2 + \Sigma_a^{(i)} + \sum\limits_{k=i+1}^{26}\Sigma_R^{i\to k} - \varepsilon^{(i)}v_f^{(i)}\Sigma_f^{(i)}} \quad (4)$$

The system of equations is transformed to:

$$\begin{cases}
\Phi_j^{(1)} = \dfrac{\varepsilon^{(1)}\sum\limits_{\substack{k=1\\k\neq i}}^{26}v_f^{(k)}\Sigma_f^{(k)}\Phi_{j-1}^{(k)}}{D^{(1)}B^2 + \Sigma_a^{(1)} + \sum\limits_{k=2}^{26}\Sigma_R^{1\to k} - \varepsilon^{(1)}v_f^{(1)}\Sigma_f^{(1)}}; \\
\Phi_j^{(2)} = \dfrac{\varepsilon^{(2)}\sum\limits_{\substack{k=1\\k\neq i}}^{26}v_f^{(k)}\Sigma_f^{(k)}\Phi_{j-1}^{(k)} + \Sigma_R^{1\to 2}\Phi_j^{(1)}}{D^{(2)}B^2 + \Sigma_a^{(2)} + \sum\limits_{k=3}^{26}\Sigma_R^{2\to k} - \varepsilon^{(2)}v_f^{(2)}\Sigma_f^{(2)}}; \\
\Phi_j^{(3)} = \dfrac{\varepsilon^{(3)}\sum\limits_{\substack{k=1\\k\neq i}}^{26}v_f^{(k)}\Sigma_f^{(k)}\Phi_{j-1}^{(k)} + \Sigma_R^{1\to 3}\Phi_j^{(1)} + \Sigma_R^{2\to 3}\Phi_j^{(2)}}{D^{(3)}B^2 + \Sigma_a^{(3)} + \sum\limits_{k=4}^{26}\Sigma_R^{3\to k} - \varepsilon^{(3)}v_f^{(3)}\Sigma_f^{(3)}}; \\
\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots\cdots \\
\Phi_j^{(25)} = \dfrac{\sum\limits_{k=1}^{24}\Sigma_R^{k\to 25}\Phi_j^{(k)}}{D^{(25)}B^2 + \Sigma_a^{(25)} + \Sigma_R^{25\to 26}}; \\
\Phi_j^{(26)} = \dfrac{\sum\limits_{k=1}^{25}\Sigma_R^{k\to 26}\Phi_j^{(k)}}{D^{(26)}B^2 + \Sigma_a^{(26)}}.
\end{cases}$$

In the present system of equations all options except the flux densities at the previous iteration $\Phi_{j-1}^{(k)}$ are known and, consequently, $\sum\limits_{\substack{k=1\\k\neq i}}^{26}v_f^{(k)}\Sigma_f^{(k)}\Phi_{j-1}^{(k)}$. This amount determines the number of neutrons produced in the second generation during nuclear fission by all neutrons of the first-generation, except neutrons of the i-th group.

To start the iterative process at zero iteration neutron flux density in the i-th group was determined from the relation (1) with (2) using the following relationship:

$$\Phi_0^{(i)} = \frac{\varepsilon^{(i)} \sum_{k=1}^{26} v_f^{(k)} \Sigma_f^{(k)} \Phi^{(k)} + \sum_{k=1}^{i-1} \Sigma_R^{k \to i} \Phi_0^{(k)}}{D^{(i)} B^2 + \Sigma_a^{(i)} + \sum_{k=i+1}^{26} \Sigma_R^{i \to k}}$$

where the number of neutrons produced in the second generation during nuclear fission by all neutrons of the first generation, was set equally to unity ($\sum_{k=1}^{26} v_f^{(k)} \Sigma_f^{(k)} \Phi^{(k)} = 1$).

Having determined the spectrum of the neutron flux at zero iteration, an iterative process using (4) was organized. The results of the 26-group calculation of the neutron spectrum in the RBMK-1000 reactor are shown in Figure 1.

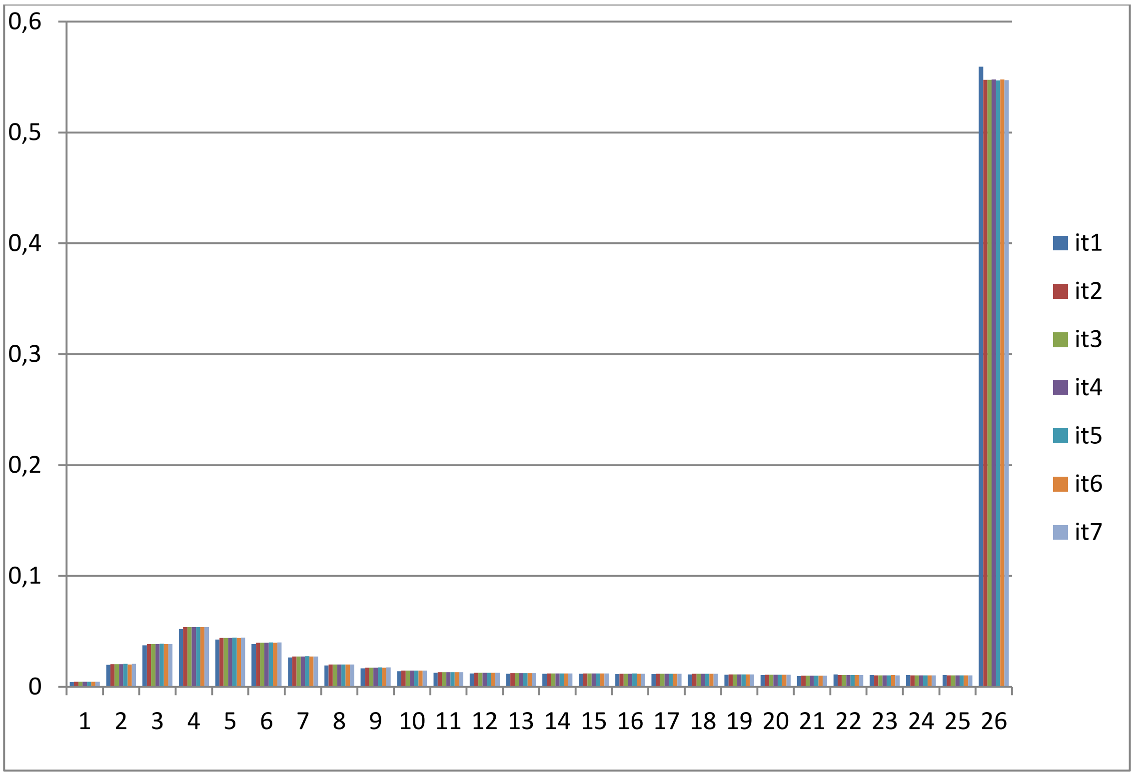

Figure 1. Range of 26 neutron groups in 7 iterations

REFERENCES

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