A METHOD FOR CALCULATING THE NEUTRON FLUX IN A SHUT-DOWN REACTOR WITH A PHOTONEUTRON SOURCE

L. V. Konstantinov and B. I. Kochetov

Translated from Atomnaya Energiya, Vol. 14, No. 4, pp. 402-404, April, 1963
Original article submitted June 9, 1962

Repeated starts of nuclear reactors involve difficulties in the measurement of the neutron flux in the subcritical state, caused by the influence of γ-radiation of fission fragments on the instruments recording the neutrons. In repeated starts artificial neutron sources are therefore used to increase the neutron flux to a value needed for reliable operation of the apparatus.

Po-α-Be-sources should not be used because the preparation of these sources with an intensity of $10^9-10^{10}$ neutrons/sec is complex and expensive. These sources with an intensity of $10^7-10^8$ neutrons/sec are only used for the initial physical starts of reactors.

Another type is the photoneutron source in which beryllium or heavy water are used. The γ-quanta sources can be radioactive isotopes Na$^{24}$, Sb$^{124}$, Mn$^{56}$, etc. The decay of these sources after the reactor has been shut down will be compensated by their activation by thermal neutrons during further operation of the reactor. A fault of photoneutron sources is the fact that they introduce an additional negative reactivity into the reactor; this can be reduced by the (γ, n) reaction for beryllium or heavy water due to γ-quanta of fission fragments.

A method is given below for the calculation of a photoneutron source, the γ-radiator of which is provided by fission fragments formed during operation of the reactor. Photoneutron sources using radioactive isotopes are calculated from analogous formulas but with an allowance for the relative arrangement of the beryllium and source of γ-quanta. Furthermore, corrections must be made for the additional yield of neutrons due to the γ-radiation of the fission fragments.

The essential amount of beryllium is calculated for a point source of neutrons placed in the active zone of a cylindrical reactor in which the effect of the reflector is allowed for.

According to [1], the neutron flux in the active zone of a subcritical reactor in the presence of a neutron source will be represented in the form of an expansion in eigenfunctions of the reactor using the notations used in [1]:

$$
\Phi_n (r; z) = \sum_{m, n} \frac{A_{mn} J_0 (\alpha_m r) \cos \frac{n \pi z}{H^*}}{\Sigma_0 (1-K_{mn} \alpha_0 \xi_m) (1-B_{mn} L^2)} 
$$

(1)

where $H^* = H_0 + \delta$ is the extrapolated height of the active zone ($\delta$ is the effective addition of the reflector); $\alpha_m = \xi_m / R^*$ (here the $\xi_m$ are the roots of the zero order Bessel function; $R^* = R_0 + \delta$ is the extrapolated radius of the active zone).

The coefficients $A_{mn}$ are determined in the expansion of the function of a source placed at the point with coordinates $r_1, z_1$ in the active zone, from the eigenfunctions of the cylindrical reactor:

$$
S_n \delta (r_1; z_1) = \sum_{m=1}^{\infty} \sum_{n=1}^{\infty} A_{mn} J_0 (\alpha_m r_1) \cos \frac{n \pi z_1}{H^*} 
$$

(2)

where $S_n$ is the intensity of a point source of neutrons (neutrons/sec). Later we will only consider the first approximation; according to the evaluation conducted for the reactor of the First Atomic Power Station for $\Delta k \leq 2\%$, this approximation gives an error in the determination of the coefficient $A_{mn}$ of not more than 2\%.
To determine the power of the subcritical reactor the flux is integrated over the whole volume of the active zone:

$$P = P_0 \eta = \int_{\text{a.z.}} \Phi(r, z) \sum_a \frac{K_n}{\nu_{\gamma,\gamma}} \, dr,$$

(3)

where \(P\) is the power of the subcritical reactor; \(P_0\) is the nominal power of the reactor; \(w = 3.1 \times 10^{-29}\) fissions/\(\text{W-sec}\).

Substituting the value \(\Phi(r_1, z_1)\) in formula (1) and integrating, we obtain

$$S_n = \frac{\frac{\pi \nu_\gamma \xi_1 (1 - K_{\text{eff}})}{K_{\text{eff}}}}{8K_{\text{eff}} \int \frac{R_0 J_1}{\sin \frac{\pi H_{\gamma}}{2H_{\gamma}}} J_1 \left( \frac{\xi_1 r_1}{H_{\gamma}} \right) \cos^2 \frac{\pi z_1}{H_{\gamma}}},$$

(4)

This expression determines the intensity of the point neutron source placed in the active zone needed to increase the power of the reactor in the subcritical state to the required quantity \(P\), depending on the subcriticality of the reactor.

By similar discussions, we can obtain formulas for the intensity of a linear source of neutrons, the volume source of neutrons distributed according to the eigenfunction of a cylindrical reactor, and for other cases.

The formula for the calculation of intensity of a volume source distributed according to the eigenfunction of the reactor (which corresponds to uniform arrangement of beryllium over the active zone) has the simplest form

$$S_n = \frac{\nu_\gamma (1 - K_{\text{eff}})}{K_{\text{eff}}} \eta P_0.$$

(5)

Having found the required intensity of the neutron source we determine the required amount of beryllium. In the calculation we should allow for the decay in time of the flux of \(\gamma\)-quanta with energy above the threshold of the \((\gamma, n)\)-reaction for beryllium (\(E_\gamma > 1.67\) MeV). According to [2, 3] the flux of \(\gamma\)-quanta with energy greater than \(E_\gamma\) is split up into three energy groups: 1.8-2.2 MeV, 2.2-2.6 MeV, and 2.8 MeV. In the center of the active zone of the reactor at time \(t\) after its shutdown, according to [23], the \(\gamma\)-quanta flux

$$\Phi_{\gamma_k}(t) = \frac{P_{\text{eff}} J_{\gamma_k} (t) \gamma_k}{4\pi \nu_\gamma \xi_1 \int \frac{R_0 J_1}{\sin \frac{\pi H_{\gamma}}{2H_{\gamma}}} E_{\gamma} \int \frac{r_1}{H_{\gamma}}},$$

(6)

where \(E_k\) is the energy of the \(k\)-th group; \(\gamma_k\) is the self-absorption coefficient of the \(\gamma\)-quanta of the \(k\)-th group in the fuel. (The other notations are given in [2].) The neutron intensity due to beryllium being carried into the center of the active zone of the reactor is determined by the formula

$$S_{n_k} (r_1 z_1) = \Phi_{\gamma_k}(\nu_\gamma) \sigma_{\gamma_k} (\gamma, n) N_{\text{Be}} J_{\gamma_k} < \left( \frac{\xi_1 r_1}{H_{\gamma}} \right) \cos^2 \frac{\pi z_1}{H_{\gamma}}.$$

(7)

where \(N_{\text{Be}}\) is the total number of beryllium nuclei and \(\sigma_{\gamma_k}(\gamma, n)\) is the effective cross section of the \((\gamma, n)\)-reaction by beryllium. Substituting the value of the \(\gamma\)-quanta flux in formula (7), from formula (6) we obtain

$$S_n = \frac{P_{\text{eff}} N_{\text{Be}} \nu_\gamma \xi_1}{4\pi \nu_\gamma \xi_1 \int \frac{R_0 J_1}{\sin \frac{\pi H_{\gamma}}{2H_{\gamma}}}} \sum_{k=1}^{3} \frac{\sigma_{\gamma_k} (\gamma_n) J_{\gamma_k} (t) \gamma_k}{E_{\gamma_k} \int \frac{r_1}{H_{\gamma}} \int \frac{z_1}{H_{\gamma}}},$$

(8)

where \(k\) is the number of groups of \(\gamma\)-quanta able to cause a photoneutron reaction.

Substituting the values of \(S_n\) in expressions (4) and (5), we obtain formulas for the calculation of the reactor power with an artificial neutron source and given subcriticality in its active zone:

1) for a point source

$$\eta = \frac{P}{P_0} = \frac{2K_{\text{eff}} N_{\text{Be}} \nu_\gamma \xi_1}{\pi \nu_\gamma \xi_1 \int \left( 1 - K_{\text{eff}} \right) \int \frac{r_1}{H_{\gamma}} \int \frac{z_1}{H_{\gamma}}} \sum_{k=1}^{3} \frac{\sigma_{\gamma_k} (\gamma_n) J_{\gamma_k} (t) \gamma_k}{E_{\gamma_k} \int \frac{r_1}{H_{\gamma}} \int \frac{z_1}{H_{\gamma}}}.$$