TECHNIQUE FOR FISSILE MATERIALS DETECTION USING ELECTRON LINAC

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The possibilities of technique for fissile materials detection using the pulse γ-quantum fluxes generated by electron linac are studied. The technique is based on detection of neutrons escaping from fissile materials after gamma irradiation (technique of delayed neutrons). The diffusion approach is developed for description of space-time evolution of neutron fluxes inside the prototype system, which is irradiated by external γ-source. The simulation of electromagnetic interaction with matter is performed using the program package GEANT. The feasibility of this technique is proved for the case of plane one-dimensional model for a three-zone homogeneous subcritical assembly consisting of ^{235}U and ^{56}Fe.

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1. INTRODUCTION

The problem of detecting fissile materials (FM) is recently question of the hour in the context of the risk of nuclear proliferation and the danger of executing acts of terrorism. The perspective methods of FM detection are the so-called active ones. Neutrons often use in these methods, since they have a capacity for traversing the materials without an essential attenuation of the initial flux (see, e.g., [1]). The active techniques of FM diagnostics can be based on detection of neutrons escaping from FM that undergo fisssion, which can be initiated by neutron bombardment or gamma irradiation (technique of delayed neutrons). So, the beam of gammas is offered to use in [2] for detection of transuranium scraps.

The neutron or gamma fluxes of required intensity can be produced using linear accelerators. It should be stated that the neutrons, which are produced by the corresponding (e,n) converter, have an angular distribution close to spherically symmetric. Therefore, the considerable part of neutron flux will be spent ineffectively, and it is necessary to increase an electron current for producing a required neutron flux. It will gain in an irradiative loading examined sample. In contrast to neutrons a γ-quantum flux produced by the (e,γ) converter (at energy of an initial electron beam more than 20 MeV) is mainly propagated in the direction of initial electron beam with a small angular divergence. It essentially facilitates a problem of localizing a γ-quantum flux on an examined region.

FM, that are prepared for illegal transport, can be disposed in special assemblies. These assemblies can have multi-layer structure, in which the FM layers alternate with the layers from other materials. It is reasonable to assume also, that the assemblies should be subcritical ones, in order to eliminate a possibility of initiating an uncontrolled chain nuclear reaction.

The simplest model of subcritical assembly is the one-dimensional model of plane infinite layers. This model corresponds to the situation when neutron leakage from assembly is absent in the transverse direction. We shall also assume, that a material composition of assembly does not include the hydric materials. It allows considering only fast neutron-induced nuclear processes, which occur in the assembly.

The purpose of the present work is to investigate a feasibility of detecting FM with the help of the technique of delayed neutrons, which is based on using the bremsstrahlung flux produced by electron linac.

The diffusion approach, which has been developed in [3], is used for describing the space-time evolution of neutron fluxes inside the subcritical assembly irradiated by external γ- source. The simulation of electromagnetic interaction with the corresponding materials is performed using the programme package GEANT. The calculations are carried out for the case of a three-zone homogeneous subcritical assembly, in which two ^{56}Fe layers surround ^{235}U layer.

2. THE CALCULATION FORMALISM

In the one-group approximation the non-stationary one-dimensional diffusion equation for the scalar neutron flux Φ can be written in the form

$$\frac{1}{v}\frac{\partial \Phi}{\partial t}-\frac{\partial}{\partial x}(D\frac{\partial \Phi}{\partial x})+\Sigma_a\Phi-(1-\overline{\beta})(\nu_f\Sigma_f)\Phi$$

$$=\sum_l\sum_i\lambda_l^i C_l^i+Q\,, \qquad (1)$$

$$\overline{\beta}=\sum_l\beta_l(\nu_f\Sigma_f)_l/\nu_f\Sigma_f\,, \qquad (2)$$

where $\Phi(x,t)$ is the scalar neutron flux, $\Sigma_\alpha(x)=\Sigma_j\sigma_\alpha^j N_j(x)$ is the macroscopic cross section of the neutron reaction of the α-type, (the index α corresponds to the reactions of neutron absorption (a) and fission (f)), $N_j(x)$ is the concentration of j'th nuclide at the point x; σ_α^j is the corresponding effective one-group microscopic cross section of the j'th nuclide; $\nu_f\Sigma_f=\Sigma_j\nu_f^j\sigma_f^j N_j(x)$, ν_f^j is the mean number of neutrons produced at the single nuclear fission event for the j'th fissile nuclide; $\overline{\beta}$ is the effective fraction of delayed neutrons, $\beta_j=\Sigma_i\beta_j^i$, here β_j^i, C_j^i and λ_j^i are the portion of delayed neutrons, the concentration and decay constant of the precursor nuclei in the i'th group of the j'th fissile nuclide, correspondingly; $D(x)=1/(3\Sigma_{tr}(x))$ is the diffusion coefficient, $\Sigma_{tr}(x))$ is the macroscopic transport cross-section, v is the one-group neutron velocity.

To create a neutron flux in the system under consideration we assume that the left boundary of the system is subjected to an external photon flux Φ_γ coming from a γ-source. The corresponding rate of neutron generation in each space point of the assembly due to the (γ,n) and (γ,f) reactions with nuclei involved in the assembly composition is defined by the relation

$$Q(x)=\int_{E^{th}_{(\gamma,n)}}^{E_\gamma^{\max}}\Sigma_{(\gamma,n)}\Phi_\gamma dE_\gamma+\int_{E^{th}_{(\gamma,f)}}^{E_\gamma^{\max}}\nu_{(\gamma,f)}\Sigma_{(\gamma,f)}\Phi_\gamma dE_\gamma\,, \qquad (3)$$

$$\Sigma_{(\gamma,n)}(x,E_\gamma)=\sum_j\sigma^j_{(\gamma,n)}(E_\gamma)N^j(x)$$

$$\nu_{(\gamma,f)}(E_\gamma)\Sigma_{(\gamma,f)}(x,E_\gamma)$$

$$=\sum_j\nu^j_{(\gamma,f)}(E_\gamma)\sigma^j_{(\gamma,f)}(E_\gamma)N^j(x),$$

where $\Sigma_{(\gamma,n)}(x,E_\gamma)$ ($\Sigma_{(\gamma,f)}(x,E_\gamma)$) is the macroscopic cross section for the (γ,n) (photo fission (γ,f)) reaction, $\sigma^j_{(\gamma,n)}(E_\gamma)$ ($\sigma^j_{(\gamma,f)}(E_\gamma)$) is the total microscopic cross section for the (γ,n) ((γ,f)) reaction with the j'th nuclide, $\nu^j_{(\gamma,f)}(E_\gamma)$ is the mean number of neutrons which are produced at the single photo fission event for the j'th fissile nuclide; $\Phi_\gamma(E_\gamma)$ is the photon flux with energy E_γ, $E^{th}_{(\gamma,n)}$ ($E^{th}_{(\gamma,f)}$) is threshold energy for the corresponding (γ,n) ((γ,f)) reaction, E_γ^{max} is the upper limit of the gamma-radiation energy.

We consider a finite one-dimensional space region $0\le x\le L$ with a certain distribution of FM and other materials, which simulates the subcritical assembly. The boundary conditions of the third kind for the flux $\Phi(x,t)$, which correspond to the free assembly boundaries, are used

$$\Phi(0,t)-2D(0,t)\left.\frac{\partial\Phi(x,t)}{\partial x}\right|_{x=0}=0\,, \qquad (4)$$

$$\Phi(L,t)+2D(L,t)\left.\frac{\partial\Phi(x,t)}{\partial x}\right|_{x=L}=0\,. \qquad (5)$$

Since the assembly under consideration is a multi-layer one the continuity conditions for the neutron scalar flux and neutron current must be satisfied at the boundary of media with the different physical properties:

$$\Phi^{/}(x,t)=\Phi^{//}(x,t)\,, \qquad (6)$$

$$D^{/}(x,t)\frac{\partial\Phi^{/}(x,t)}{\partial x}=D^{//}(x,t)\frac{\partial\Phi^{//}(x,t)}{\partial x}\,, \qquad (7)$$

where the primes mark the quantities belonging to different media.

These conditions are valid for any moment of time within the time interval $0\le t\le T$ considered. The initial condition for the neutron flux $\Phi(x,t)$ at the moment $t=0$ for all values x from the space interval $0\le x\le L$ is chosen as

$$\Phi(x,t=0)=0\,. \qquad (8)$$

The burn-up of FM will be neglected, since we confine ourselves to consideration of the assembly operation during small time

The equations of nuclear kinetics for 6 groups of the precursor nuclei of delayed neutrons take the form

$$\frac{\partial C_l^i}{\partial t}=-\lambda_l^i C_l^i+\beta_l^i(\nu_f\Sigma_f)_l\Phi \qquad (9)$$

with the initial conditions

$$C_l^i(x,t=0)=C_l^{i0}(x)\,. \qquad (10)$$

In the case under consideration the flux Φ weakly varies during the characteristic decay time of the precursor nuclei that emit delayed neutrons. Therefore, number of the kinetic equations (9) can be reduced, using the approach of one equivalent group of the precursor nuclei

$$\frac{\partial C_l}{\partial t}=-\lambda_l C_l+\beta_l(\nu_f\Sigma_f)_l\Phi\,, \qquad (11)$$

$$C_l(x,t=0)=C_l^0(x)\,, \qquad (12)$$

where $\lambda_l=\beta_l/\sum_i\beta_l^i/\lambda_l^i$.

The complete statement of the problem considered includes the set of partial differential equations (1), (11) and corresponding initial and boundary conditions to them as well. For solving this nonstationary problem we have used the finite-difference method. To apply the finite-difference technique a rectangular mesh with steps h and τ (uniform for x and variable for t) in the range of variables x and t is introduced. We shall find the solutions of the set of the algebraic equations obtained from Eq. (1) in this way using the implicit Crank-Nickolson difference scheme [4] (for details, see [3]). The solutions of Eq. (11) can be simplified by assuming that the neutron flux Φ is constant during the time intervals τ. This assumption can easily be satisfied by choosing sufficiently small time intervals τ, on which the flux value should be taken as $\Phi=(\Phi_n+\Phi_{n+1})/2$ (where Φ_n is the neutron flux value for the n-th time layer).

Then the expressions for the concentrations of precursor nuclei for the new (n+1)-th layer at every node of the space mesh can be obtained using the analytical approach described in Ref. [3]

$$C_l^{n+1}=e^{-\lambda_l\tau}\left[C_l^n+\frac{\beta_l}{\lambda_l}(\nu_f\Sigma_f)_l\Phi\left(e^{\lambda_l\tau}-1\right)\right]. \qquad (13)$$

The initial condition is chosen as $C_l^0=0$.

Thus, the set of partial differential equations (1) and (11) is reduced using the Crank-Nickolson difference scheme to the set of algebraic equations, in which dependence of the concentrations of precursor nuclei on the sought-for neutron flux Φ is defined by Eqs. (13). The numerical solutions of this set of equations have been calculated using the method like in [3].

For calculations of the effective one-group microscopic cross sections we used the group neutron fluxes Φ^g (g is the number of neutron energy group) for the initial assembly calculated from solving the stationary multigroup problem. Calculations were performed in the 26-group approximation using the library of group neutron constants from Ref. [5]. The procedure for calculating the one-group effective cross sections is defined by the relations (see [3])

$$\sigma_\alpha^l = \sum_{g=1}^{26} \frac{\sigma_\alpha^{gl} \Phi^g(K)}{\Phi_S(K)}, \qquad (14)$$

$$\Phi_S(K) = \sum_{g=1}^{26} \Phi^g(K),$$

where $\Phi_S(K)$ is the neutron flux summed over 26 groups, the index α corresponds to the reactions of neutron capture, fission and scattering, the index K numerates the node of the space calculation mesh.

The one-group neutron velocity is given by

$$\frac{1}{v} = \frac{1}{\Phi_S(K)} \sum_{g=1}^{26} \frac{\Phi^g(K)}{v^g}, \qquad (15)$$

where v^g is the neutron velocity for the group g.

The transport cross section σ_{tr} is averaged according to the following expression

$$\sigma_{tr}^l = \sum_{g=1}^{26} \sigma_{tr}^{gl} D^g \Phi^g(K) / \sum_{g=1}^{26} D^g \Phi^g(K), \qquad (16)$$

where D^g is the diffusion coefficient for the group g.

3. RESULTS OF CALCULATIONS

To solve the main problem of the present work it is necessary to simulate intense neutron field inside the subcritical assembly by the external γ-source. The description of corresponding evolutionary problem is based on the diffusion approach described above.

We consider a three-zone (layer) homogeneous subcritical assembly that consists of the high-enriched (100%) metal ^{235}U fuel of porosity $p = 0.8$ and the constructional material ^{56}Fe (see Fig. 1).

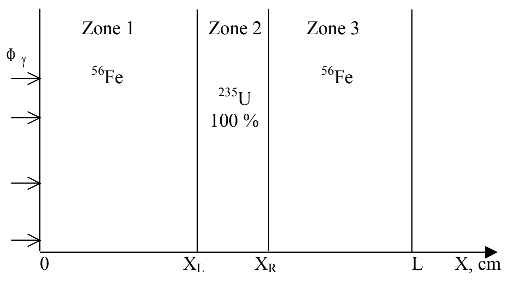

***Fig. 1.** The subcritical assembly*

The calculations start from distributing ^{235}U and ^{56}Fe nuclei on the corresponding zones. The first (third) zone (near the left (right) edge of the assembly) with the width 25 cm of every one is filled only with ^{56}Fe. The second zone represents the thin layer of ^{235}U. The subcritical assembly width, $0 \le x \le L$, is divided into $M = 200$ intervals of the spatial calculation mesh. We impose the boundary conditions (4)-(7) on the scalar neutron flux $\Phi(x,t)$. To create the intense neutron flux in the system we assume that the left boundary of the assembly is subjected to an external photon flux $\Phi_\gamma(E_\gamma)$ coming from a γ-source. The photon flux simulates inside the assembly the neutron flux that is defined by the density of neutron generation rate $Q(x)$ (3).

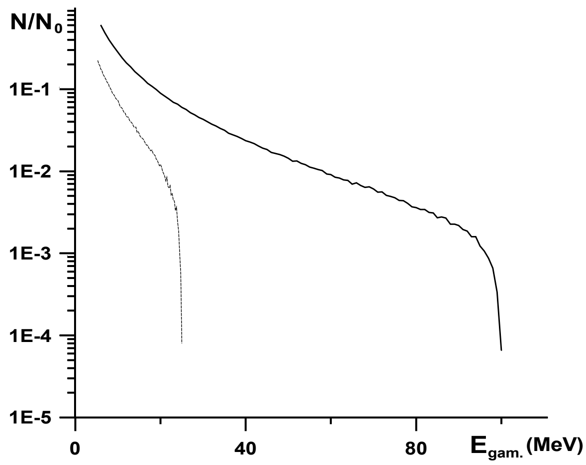

***Fig. 2.** Relative dependence of γ - quantum in energy E_γ calculated for the electron energies 25 MeV (dashed curve) and 100 MeV (solid curve)*

The parameters of the subcritical assembly under consideration were determined by the numerical solution of the multigroup criticality problem. So, the value of effective multiplication factor of neutrons in this system $k_{eff} = 0.93$ when the width of ^{235}U layer was chosen to be equal to 13 mm.

The simulation of electromagnetic interaction with the corresponding materials was carried out for an electron linac with the following parameters: the average current 1 ma, the maximal energy of electron beam 100 MeV and the frequency 300 Hz. The tantalum target was used as the (e,γ)-converter. The thickness of the target is 6 mm.

Fig. 2 presents the energy distributions of γ - quanta, which are calculated for two values of the electron energy using the software package GEANT. These spectra define the number of γ - quanta N of certain energy that escape from the converter in a case, when the initial electron beam contains $N_0 = 10^6$ particles.

Fig. 3 shows the spatial distribution of the integral γ - quantum flux, which is defined by the following relation

$$N / N_0 = \frac{1}{N_0} \int_{5MeV}^{E_\gamma^{max}} N(E_\gamma) dE_\gamma, \qquad (17)$$

where $N(E_\gamma)$ is the number of γ - quanta of energy E_γ.

As can be seen from Fig. 3, the γ - quantum flux appreciably decreases depending on the distance of penetrating into the assembly materials. This dependence has the exponential character. The jump that the curve undergoes in the second zone is explained by more strong flux attenuation in ^{235}U layer in contrast with ^{56}Fe layers. This is stipulated by value of the mass attenuation factor for ^{235}U, which is greater than that for ^{56}Fe in the γ - quantum energy region of interest.

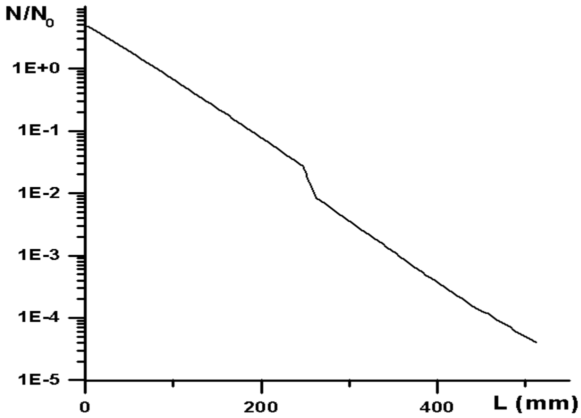

Fig. 3. *Relative space distribution of the integral γ - quantum flux inside the subcritical assembly for the electron energy 100 MeV*

Fig. 4 presents the spatial distribution of density of neutron generation rate $Q(x)$ (3) inside the subcritical assembly that is calculated for the electron beam energy 100 MeV.

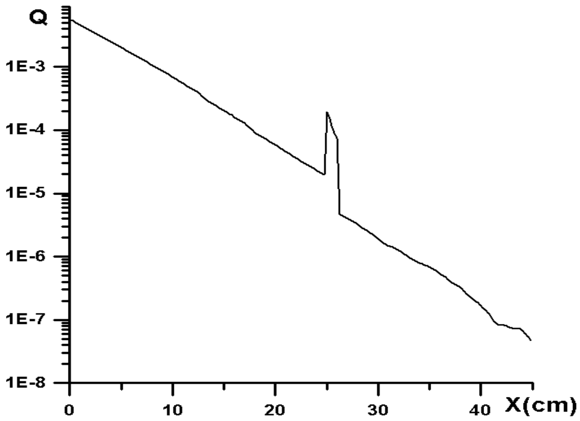

Fig. 4. *Spatial distribution of $Q(x)$ [$\times 10^{16}$ $cm^{-3}s^{-1}$] inside the subcritical assembly*

In the first and third zones the curve of $Q(x)$ exponential decreases that corresponds to the photon flux damping when the γ-radiation traverses the iron. In the second zone the noticeable enhancement of $Q(x)$ value is observed. This burst of the neutron generation rate is mainly conditional on the contribution of the (γ,f) reaction on ^{235}U to the formation of $Q(x)$. The photofission cross section takes on values that are greater as compared with the (γ,n) cross sections for ^{235}U and ^{56}Fe. It should be also noted that mean number of neutrons produced at the single fission event of ^{235}U is greater than two and has the tendency to increase with increasing photon energy. Besides the (γ,n) cross section for ^{235}U is greater than that for ^{56}Fe in the giant dipole resonance region.

Results of solving the nonstationary problem to define the neutron flux inside the subcritical assembly are presented at different time moments of turning the external photon flux off t_{off} in the succeeding figures.

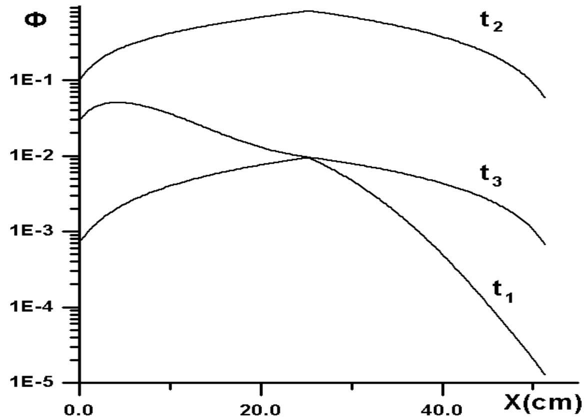

Fig. 5. *Spatial distribution of the neutron flux $\Phi(x)$ [$\times 10^{16}$ $cm^{-2}s^{-1}$] at time moments $t_1 = 3.544 \cdot 10^{-7}$, $t_2 = 1$ and $t_3 = 2$ seconds. The photon flux is turned off at $t_{off} = 1$ second*

Fig. 5 shows the space distribution of the neutron flux for different moments of time. At the initial stage for very small irradiation time (see the time moment t_1) the distribution shape to a considerable extent differs from the distribution shape of the neutron flux for greater intervals of irradiation time. The neutron flux attains its maximum in the second layer (see Fig. 5) at the time moment t_2 when the photon flux is turned off. The calculations show that the maximum value of the neutron flux Φ_{max} at this time moment constitutes only about 80 % of Φ_{max} value calculated for the corresponding stationary problem. After turning the photon flux off the space distribution of neutron flux at the time moment t_3 has the same shape as for the time moment t_2. However at the time moment t_3 the Φ_{max} value becomes by about two orders of magnitude smaller than that at the time moment t_2.

Time dependence of Φ_{max} for two values of $t_{off} = 1$ and $t_{off} = 0.01$ second is presented in Fig. 6. In both cases at the time moment t_{off}, when the photon flux is turned off, Φ_{max} reaches practically the same magnitude. However during time after turning the photon flux off the magnitudes of Φ_{max} differ to a considerable degree in going from one case to another. The distinction in the corresponding magnitudes of Φ_{max} is about two orders.

Note that the precursor nuclei of delayed neutrons plays role of the neutron source inside the subcritical assembly after turning the external photon flux off (see Eq.(1)). The concentration C_U of precursor nuclei, that are product of fission of ^{235}U, is proportional to the so-called neutron fluence $F = \Phi\, t$. Since the flunce in the former case is greater than that in the latter one approximately by a factor of 10^{2}, the same ratio is observed between the C_U values for these two time moments t_{off}. Apparently the distinction in the corresponding magnitudes of Φ_{max} mentioned above is associated with this ratio between the C_U values for the time moments following after $t_{off} = 1$ and $t_{off} = 0.01$ second, correspondingly. One can see that the ratio between the corresponding magnitudes of Φ_{max} conserves during one second after turning the photon flux off in both cases. As a matter of fact this ratio takes place a longer time period.

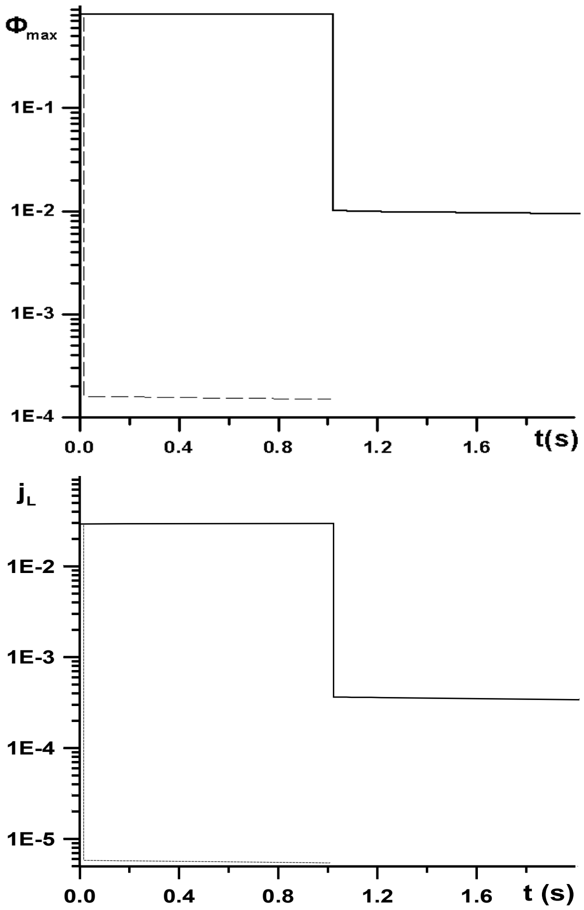

Fig. 6. *Dependences of* Φ_{max} *[×10 ^16^ cm^-2^s^-1^] (top) and* j_L *[×10^16^ cm^-2^s^-1^] (bottom) versus time. Curves are calculated for the photon flux turned off at* $t_{off} = 1$ *second (solid) and* $t_{off} = 0.01$ *second (dashed)*

The similar picture is observed in the time dependences of the neutron leakage current $j_L = D\partial\Phi/\partial x$ from the right boundary of the assembly. The maximum value, which j_L reaches at the time moment of turning the photon flux off $t_{off} = 1$ second, is $3{\cdot}10^{14}$ $cm^{-2}s^{-1}$ (see Fig. 6 (bottom)). After turning the external γ - source off at $t_{off} = 1$ second the j_L value rapidly falls off at first. Then j_L changes very slowly during one second and takes on the magnitude of $3.4{\cdot}10^{12}$ $cm^{-2}s^{-1}$ at the time moment $t = 2$ seconds. Under the condition $t_{off} = 0.01$ second j_L takes on the value of $5.4{\cdot}10^{10}$ $cm^{-2}s^{-1}$ at the time moment $t = 1.01$ second.

Note that the energy production density reaches the maximum value of about 17 kW t cm^{-3} in ^{235}U layer at both time moments of turning the external photon flux off $t_{off} = 1$ and $t_{off} = 0.01$ second. After turning the photon flux off the energy production density decreases rapidly and takes on the values of 0.2 kW t cm^{-3} and $0.32{\cdot}10^{-2}$ kW t cm^{-3} at the time moments $t = 2$ and $t = 1.01$ seconds, correspondingly.

4. CONCLUSIONS

The feasibility of FM detection by the technique of delayed neutrons has been proved in the special case of the three-zone homogeneous subcritical assembly consisting of ^{235}U and ^{56}Fe. The main features of the technique have been studied for the case of the electron linac with beam energy 100 MeV and the tantalum (e,γ)-converter.

The neutron leakage current, which characterizes the neutron flux emerging from the right boundary of the assembly, is initiated and driven by the γ - quantum flux, which generates neutrons with the rate *Q(x)* in each space point of the assembly. The maximum value of this neutron flux is equal to $3.5{\cdot}10^{14}$ $cm^{-2}s^{-1}$ for the stationary problem that has been calculated with the time-constant external photon flux. After turning the external γ-source off the neutron flux is completely maintained inside the assembly by precursor nuclei of delayed neutrons. These nuclei, which are concentrated in the second zone after fission of ^{235}U, play the role of inner source that generates neutrons with the rate $\lambda_U C_U$. Thus, the neutron flux substantially changes after turning the external γ-source off. The flux decreases at first very rapidly by two orders of magnitude. After that the neutron flux changes very slowly during rather long time. The j_L value takes on $3.4{\cdot}10^{12}$ $cm^{-2}s^{-1}$ at the time moment $t = 2$ second for time of turning the external γ-source off $t_{off} = 1$ second. For the case $t_{off} = 0.01$ second the neutron flux takes on the value of $5.4{\cdot}10^{10}$ $cm^{-2}s^{-1}$ at the time moment $t = 1.01$ second.

Hence, one can change the neutron flux emerging from the assembly by choosing the gamma irradiation time t_{off} or the corresponding value of neutron fluence. In the present analysis it is shown that these neutron fluxes can take on the values, which are available for detection by the existing neutron detectors. The corresponding measurements can be carried out during rather long period (several seconds) since the neutron flux does not appreciably change over this period.

To embody the technique of delayed neutrons under consideration the corresponding technology for producing photon fluxes of high intensity is developed using 100 MeV variable linac.

It should be noted, that the results obtained in the present work, have a somewhat qualitative character, since the plane one-dimensional model of the subcritical assembly is not entirely corresponding to real cases. For this reason, the results are better of theoretical interest and serve to ascertain the main features of initiation and evolution neutron flux that is generated inside the subcritical assembly by the external γ-source. However, all the above-made approximations allowed us to describe the observed qualitative picture of the processes well enough. It should also be stated that results obtained with taking into account the neutron leakage from the assembly in the transverse direction, that is an attribute of the real assembly model, can somewhat quantitatively alter the results presented above. Of course, results obtained in the framework of approach developed substantially depend on the material composition of the subcritical assembly. The special consideration is necessary in the case when the hydric materials are in the composition of the assembly. The main reason is associated with peculiarity of the neutron interaction with the hydrogen nuclei, which is characterized by strong anisotropy of scattering in laboratory coordinate system and heavy loss of energy in single collision as well. Thus, the use of diffusion approximation in the case leads to appreciable distortion of the neutron penetrability.

A further work is necessary to study the technique under consideration using more complete mathematical models.

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МЕТОД ОБНАРУЖЕНИЯ ДЕЛЯЩИХСЯ МАТЕРИАЛОВ С ИСПОЛЬЗОВАНИЕМ ЛИНЕЙНОГО УСКОРИТЕЛЯ ЭЛЕКТРОНОВ

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Исследуются возможности метода обнаружения делящихся материалов с использованием импульсных потоков γ-квантов, генерируемых линейным ускорителем электронов. Метод основан на регистрации нейтронов, испускаемых делящимся материалом после γ-облучения (метод запаздывающих нейтронов). Развит диффузионный подход для описания пространственно-временной эволюции потока нейтронов в исследуемом объекте, который облучается внешним источником γ-квантов. Моделирование процессов электромагнитного взаимодействия с веществом проводится с помощью пакета программ GEANT. Осуществимость этого метода подтверждена результатами расчетов, проведенных в случае плоской одномерной модели для гомогенной трехзонной подкритической сборки, в которой слой ^{235}U окружен двумя слоями ^{56}Fe.

МЕТОД ВИЯВЛЕННЯ МАТЕРІАЛІВ, ЩО ПОДІЛЯЮТЬСЯ, З ВИКОРИСТАННЯМ ЛІНІЙНОГО ПРИСКОРЮВАЧА ЕЛЕКТРОНІВ

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Досліджуються можливості методу виявлення матеріалів, які поділяються, з використанням імпульсних потоків γ-квантів, що генерує лінійний прискорювач електронів. Метод засновано на реєстрації нейтронів, випромінюваних матеріалом, що поділяється, після γ-опромінення (метод запізнілих нейтронів). Розвинуто дифузійний підхід для опису просторово-часової еволюції потоку нейтронів у досліджуваному об'єкті, що опромінюється зовнішнім джерелом γ-квантів. Моделювання процесів електромагнітної взаємодії з речовиною проводиться за допомогою пакету програм GEANT. Можливість запровадження цього методу підтверджено результатами розрахунків, проведених у випадку плоскої одновимірної моделі для гомогенної трьохзонної підкритичної збірки, у якій шар з ^{235}U оточений двома шарами з ^{56}Fe.