THE PULSED SUBCRITICAL AMPLIFIER OF NEUTRON FLUX DRIVEN BY HIGH-INTENSITY NEUTRON GENERATOR

by

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The subcritical reactor driven by external neutron source could apply as useful instrument for modern nuclear energy applications requiring high-level irradiation of different materials by the high-energy and high-intense neutron flux (*e. g.*, nuclear waste transmutation, radiopharmaceutical production, *etc*.). The propagation of neutron pulses through the subcritical nuclear system was considered in the present paper. Simple homogeneous subcritical systems and a model of two-zone subcritical reactor were computationally investigated using Monte Carlo MCNP4c transport code. The propagation of one initial neutron pulse and series of one hundred neutron pulses through the presented subcritical nuclear models were simulated. In this study, the neutron multiplication factor, the neutron flux, the energy amplification factor, the total energy of neutrons in initial pulse, *etc*. were obtained and analyzed. The presented calculations have shown that the considered pulse subcritical systems can be successfully used as effective amplifiers of neutron flux from the initial source. The modeling results indicate that there is an achievement of a stable, high level of neutron flux caused by the accumulation of delayed neutrons from previous pulses in series of one hundred pulses for both homogeneous and heterogeneous systems.

Key words: accelerator-driven system, Monte Carlo calculation, pulsed research reactor, D-T neutron generator

INTRODUCTION – SOME GENERAL CONCEPTS OF USES OF THE SUBCRITICAL MULTIPLYING SYSTEMS

A great number of modern applications require high-level irradiation of different materials by a high-energy and high-intense neutron flux (for example, problem of nuclear waste transmutation, production of medical radionuclides for nuclear medicine, *etc*.) [1, 2]. The usual way to solve this problem lies in irradiation of these materials by neutron flux inside a research nuclear reactor or inside an industrial power reactor. The second way is used by providing the possibility of the access to the nuclear reactor core during operation of the reactor, which is highly problematic. So, the further alternative possibilities to solve this problem are highly welcomed, especially in the view of problems with contemporary nuclear reactors. In general, most of the world's neutron sources were built decades ago, and although the uses and demand for neutrons have greatly increased throughout the years, just a few new neutron sources have been built.

One of the alternative possibilities to obtain high neutron flux lies in the use of neutron generators [3]. Neutron generators are neutron source devices which contain linear accelerators that produce neutrons by fusing isotopes of hydrogen together. The D-T nuclear reaction is used in the majority of such devices, where fusion of a deuteron ^{2}H and a triton ^{3}H nuclei (D + T) results in the formation of alpha particle ^{4}He and a neutron with kinetic energy of approximately 14.1 MeV. But, the yield of neutrons in the D-T reaction is limited by the current strength of the deuteron beam, inside the accelerator tube of neutron generator. Thus, average neutron yield in the majority of neutron generators amounts to 10^8-10^9 neutron/s, which is absolutely not enough for application purposes. But in addition, there is some information about neutron generators that produce average neutron yield ~10^{13}-10^{14} neutron/s [4, 5].

Another alternative possibility to obtain high neutron flux, lies in the use of nuclear spallation process. In nuclear physics, spallation is the process in which a heavy nucleus emits a large number of nucleons, particularly neutrons, as a result of being hit by a

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high-energy particle, thus greatly reducing its atomic weight. For example, protons or deuterons being accelerated up to ultra-high energies ~1 GeV may produce a large number of neutrons by the spallation process on heavy metal nuclei [2, 6] (energy of ~25-40 MeV is required to produce one neutron). In this connection, Spallation Neutron Source (SNS) at US Oak Ridge National Laboratory, can be mentioned as an example [7].

In general, a lot of different projects of subcritical systems driven by proton accelerators, exist in the world. A subcritical reactor is a nuclear fission reactor concept that produces fission without achieving criticality. Instead of a sustaining chain reaction, a subcritical reactor uses additional neutrons from an outside source. The neutron source can be a nuclear fusion machine or a neutron source, producing neutrons through spallation of heavy nuclei by charged particles such as protons accelerated by a particle accelerator. Such a device with a reactor coupled to an accelerator is called an accelerator driven system (ADS). Thus, an ADS is, in general, a neutron source created by coupling a proton accelerator, a spallation source and a subcritical core. For example, European project FREYA (Fast Reactor Experiments for hybrid Applications) involves development and creation of ADS MYRRHA [8].

Let us also note that experimental low-power subcritical assembly "YALINA-BOOSTER", driven by D-T neutron generator, is already built in Joint Institute for Energy and Nuclear Research (Sosny, Minsk, Belarus) [9]. The yield of neutrons in both aforementioned cases is essentially limited by the current strength of the particle beam inside the accelerator, while increase of the current strength up to a value needed for applications is a rather complicated problem [10]. Thus, subcritical source-injected multiplying systems are today proposed as a viable and acceptable nuclear means for producing energy and transmuting actinide and fission products produced by conventional reactors [11]. A large effort is devoted worldwide to assessment of ADS, where source neutrons are produced by spallation reactions caused by high energy protons impinging on a heavy-nuclide target.

Despite the fact that many research ADS projects consider an external neutron source, based on spallation reaction, authors of the present paper think that DT neutron generators with high intensity (10^{14}-10^{15} neutrons/s) deserve alternative consideration instead of high-energy proton accelerators, first of all from the viewpoint of cost parameter of neutron source.

Another important ADS aspect is the discontinuity of the charged particle beam in most accelerators, which is a consequence of the pulsating nature of the neutron source in ADS. In this regard, questions arise about the formation of the neutron pulse for the source (in the target) and of the passage of the neutron pulse in the subcritical reactor – establishing the average neutron flux, relaxation times, the influence of delayed neutrons, *etc*.

Therefore, within analysis of propagation of neutron pulses through the subcritical systems in the present paper, we will consider monoenergetic 14.1 MeV neutrons from D-T reaction as external particles for subcritical reactor.

SOME ASPECTS OF PROPAGATION OF NEUTRON PULSE THROUGH THE SUBCRITICAL NUCLEAR SYSTEM

We can also use the amplifying property of subcritical multiplying systems in order to enhance neutron flux of the external source. The quantity of neutrons n in the steady-state subcritical multiplying system, within the framework of the point model of nuclear reactor, can be determined by the expression

$$\frac{n}{\tau} \quad \frac{q_0}{l \quad k} \quad Mq_0, \quad M \quad 1 \quad k \quad k^2 \ \dots \qquad (1)$$

This means that in a time τ number of neutrons q^0 emitted by the external source increases k times. Here τ is the neutron generation time, which is connected to the lifetime of the neutrons in the system τ_0 by the relation $\tau = \tau_0/k$, while k is, as usually, the effective neutron multiplication factor. Once again, let us note that eq. (1) is valid only for the steady-state subcritical systems in the one-velocity point approximation. Let us also note that subcritical systems require an external source to maintain a steady-state, and the resulting neutron distribution is dominated by source injections, rather than by the neutrons emitted through the inner multiplication process [11].

Generally speaking, the property of the subcritical system to amplify neutron flux of the external source is used in most cases in the steady-state subcritical systems [12, 13]. Therefore, it would be of great interest to investigate and analyze the problem of amplifying properties of the subcritical system in the case of pulsed neutron sources. The efficiency of the subcritical system to amplify neutron flux of the pulsed neutron source, until to now is in question. Let us note that pulsed neutron sources (spallation sources, neutron generators or reactors) do not emit a continuous neutron beam, but pulses. For spallation sources this is an intrinsic feature, whereas reactors can be forced to produce pulses – in order to achieve a higher neutron flux density. Thus, in order to increase the maximum neutron flux, special pulsed reactors were developed. Spallation sources and neutron generators, in general, are pulsed because they run in connection to a pulsed particle accelerator.

Propagation of neutron pulses through the subcritical system was widely investigated earlier, in connection with the problem of criticality measurements in subcritical systems (see for ex. [14]). The results of those investigations show very clearly that neutron pulse is damped as it propagates through the subcritical system. Damping of pulse wave, obviously, is determined by the properties of multiplying medium, whereas the damping time period is specifically determined by both prompt fission neutrons and delayed neutrons. In other words, neutron pulse is quickly damped during the initial time interval according to the law of exponential decrease, where the decrease coefficient α is given by

$$\alpha = \frac{\beta - \rho}{\tau} \tag{2}$$

Here β is a portion of delayed neutrons, ρ is nuclear reactivity of the system. After the fast decrease caused by prompt neutrons, the long overextended *tail* of the neutron pulse is formed due to delayed neutrons.

If the frequency R of propagation of neutron pulses satisfies the relation $\alpha << R << \lambda$, where λ is the characteristic decay coefficient of the emitter of the delayed neutrons, with the maximum lifetime, then the system comes to a steady-state (see, *e.g.*, [14]). Let us remind that a system in a steady-state has numerous properties that are unchanging in time. In this specific case of the steady-state process, the number of the delayed neutrons n_d, within the interval between pulses, comes to a flat plateau that is unchanging in time. This number of the delayed neutrons n_d is determined by

$$R\int_0^{1/R} n_\mathrm{d}\,\mathrm{d}t = \frac{\beta q_0 R}{\rho\alpha} \tag{3}$$

Thus, in the case of a pulsed neutron source, we should observe amplification of neutron flux, both during the time period of prompt neutron pulse and during the time period of the delayed neutrons flat plateau. The number of the prompt neutrons n_p, during the prompt neutron pulse, is determined by

$$\int n_\mathrm{p}\,\mathrm{d}t = \frac{q_0\tau}{\beta - \rho} \tag{4}$$

Therefore, amplification of neutron flux is higher during the time period of the delayed neutrons flat plateau than during the time period of prompt neutron pulse. The latter statement is a consequence of the fact that number of the delayed neutrons during the time period of the flat plateau is, among other factors, determined by contribution of the delayed neutron *tails* arising from the previous neutron pulses.

Generally speaking, the previous analyzed picture of propagation of neutron pulses through the subcritical system lacks many important details. Firstly, as a sequence of the use of one-velocity approximation, we did not take into account neutron multiplication due to fast neutrons. This would probably make a substantial difference in the case of neutron source devices which use the D-T nuclear reaction producing neutrons with a kinetic energy of approximately 14.1 MeV. Another example of the significance of taking into account the fast-neutron multiplication effect, comes from the use of spallation neutron sources, which can produce neutrons with the energy up to 300 MeV, depending on the energy of protons or deuterons being accelerated up to high energies, as mentioned above. Fast multiplication factor for the spallation neutrons may considerably exceed the value 1.2, which is typical for fast neutron reactors. Secondly, the previous analysis does not take into account the neutron pulse propagation in space, whereas knowledge of the spatial distribution of the neutron flux in the reactor core is needed to choose the best and optimal location of targets and detectors. Thirdly, the previous discussion does not make clear the picture of saturation attainment, which is observed in the process of propagation of pulse group through the subcritical system. Fourthly, the papers [15-17] demonstrated how the amplification factor is affected by the presence of two different material zones in the system. Moreover, this influence is mainly determined by the multiplication factor of each zone. However, according to our calculations, the amplification factor is also substantially affected by the material composition of each zone. The latter fact is explained by neutron multiplication due to fast neutrons, as previously mentioned.

MODELS FOR CALCULATION AND METHODS

In this study we investigate some general physical properties and characteristics of the pulsed subcritical amplifier of neutron flux, within the framework of different design models of nuclear reactor, starting with some simplified homogeneous models of the subcritical reactor, and finishing with more realistic heterogeneous reactor model, which is close to actual construction of the future subcritical research reactor driven by high-intensity neutron generator. These models have been developed in the framework of the research and development project of the subcritical research nuclear reactor at Institute for Nuclear Research of the National Academy of Sciences of Ukraine and at Institute for Safety Problems of NPP of the National Academy of Sciences of Ukraine.

It is reasonable to investigate and analyze the propagation of neutron pulses through the nuclear subcritical system with the help of computational experiment that completely simulate the system under consideration. Simulation experiment of this kind should involve a choice of simulation method, and also total definition and specification of the prototype system, including nuclide composition of the system, its geometry, characteristics of pulsed neutron source, boundary conditions, and so on.

The following considerations should be taken into account in the course of the choice of the simulated system.

Firstly, the system under consideration must be subcritical enough to meet the requirements of regulatory authorities for the totally safe nuclear reactor systems. It means that the effective multiplication factor of the system should not exceed the value $k_{\text{eff}} = 0.97$. Theoretically, we can also consider the prototype system with a higher value of the quantity k_{eff}, primarily to estimate possible benefits of such increase.

Secondly, the quantity of fissile material (such as ^{235}U) should be as minimal as possible. Consequently, certain steps must be taken to reduce the amount of fissile material in order to make system more cheap and safe. Among these steps we can mention the use of the most effective neutron reflectors, beryllium coming up in the first place due to efficiency of nuclear reaction $(n, 2n)$ inside this material. We should also use the most optimal ratio of the fissile material quantity to the moderator material quantity. As a result, the most effective ratio of the amount of ^{235}U nuclei to the amount of hydrogen nuclei is 1:500, according to reference books (see, *e. g.*, [18]).

Thirdly, it is advantageous to use the property of some heavy nuclei, first of all of ^{238}U, to be fissionable due to absorption of fast neutrons with energy higher than fission threshold. Let us remind that ^{238}U is known to be the most common isotope of uranium found in nature. It is not fissile, but is a fertile material: it can capture a slow neutron and after two beta decays become fissile ^{239}Pu. Though ^{238}U is fissionable with fast neutrons, it cannot support a chain reaction because inelastic scattering reduces neutron energy below the range where fast fission of one or more next-generation nuclei is probable. In summary, the initial slowdown of the neutrons should be performed inside of the heavy material, such as ^{238}U or ^{232}Th. The secondary slowdown of the neutrons should then be performed inside of the good moderator mixed with the fissile material. It is important to note that analytical analysis of such a system, in the framework of the one-velocity approximation, is impossible.

Homogeneous calculation models

We start the consideration with the following simple configurations of homogeneous systems. The point neutron source is placed in the center of a spherical subcritical nuclear assembly and it emits either a single pulse of the monoenergetic neutrons with the energy of 14 MeV and duration of about 10^{-8} s, or a group of such pulses with interpulse period, *i. e.* pulse repetition interval, of about 10^{-4} s. Subcritical nuclear assembly is a homogenized fuel system, and is built up from a number of concentric spherical layers. The inner layer of the assembly is composed of either natural uranium or depleted uranium, which is fissionable by fast neutrons. Let us remind that around 99.3 % of natural uranium is ^{238}U, while depleted uranium has even higher concentration of ^{238}U. The next (external) layer of the assembly is composed of water solution with the dissolved ^{235}U, where the ratio of the amount of ^{235}U nuclei to the amount of hydrogen nuclei is 1:500, or so. The layer thicknesses are chosen on the basic principle that the effective multiplication factor of the system should not exceed unity. On the other hand, the layer thicknesses and, respectively, design of the assembly are chosen on the principle that the amplification factor of neutron flux coming from the source should reach a maximum value.

Heterogeneous calculation model

It is also of great interest to study propagation of neutron pulses through the realistic subcritical nuclear system close to actual nuclear reactor. For this purpose, we shall use the suggested neutronic design model of two-zone research subcritical nuclear reactor, which is developed in the framework of the State Project of the National Academy of Sciences of Ukraine *Study of non-linear and stochastic properties of multiplying nuclear systems* [19-22]. Design model of this reactor is a two-zone subcritical system driven by high-intensity neutron generator (see fig. 1). The internal zone of the proposed reactor has fast neutron spectrum, while the external zone of the reactor has thermal neutron spectrum. Such a design of the reactor core allows to perform different types of research investigations on transmutation of the whole spectrum of radioactive wastes, *i. e.* both on transmutation of minor actinides in the internal zone with fast neutron spectrum [23-25], and on transmutation of long-lived fission products in the external zone with thermal neutron spectrum [26]. A great variety of different experiments and studies could also be available at such a research facility. It is also of great importance that well-optimized two-zone subcritical system makes it possible to scale down a request to the external neutron source and to bring down the cost of the subcritical research reactor, as demonstrated in [27, 28].

Geometrical and material characteristics of the subcritical reactor system are chosen on the basis of

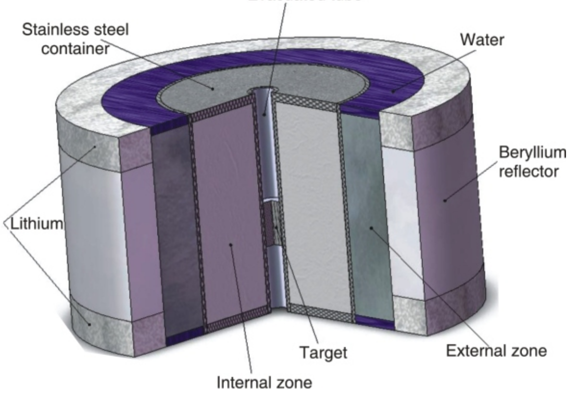

Figure 1. The scheme of the two-zone research subcritical reactor

simulation, investigation and optimization of the system performed in [20-22, 24]. The effective multiplication factor of the system takes the value k_{eff} 0.97. External deuteron beam travels through a vacuum tube in the center of the system. This vacuum tube is made of stainless steel SS304 [29]. Tritium-saturated titanium target is placed in the center of the vacuum tube. The target diameter coincides with the inner diameter of the tube and amounts to 5 cm [30]. The processes occurring in a target under deuteron bombardment, namely the D-T nuclear reaction, produce neutrons with a kinetic energy of approximately 14 MeV. As a result, we get an intense 14 MeV neutron source in the center of the subcritical system. Titanium target is placed on a copper base cooled by water, and in turn, water cooling pipes are placed in the lower part of the central tube, under the target.

The internal zone with fast neutron spectrum, surrounding the central tube, is placed in the tank made of stainless steel SS304. Thickness of the steel wall of the tank amounts to 1 cm. The internal fast zone of the system is composed of the reactor WWER-1000 shortened fuel pins cooled by the helium coolant. The whole internal zone has a diameter of 45.3 cm and height of 50 cm. The external thermal zone of the system, surrounding the fast zone, is also composed of the reactor WWER-1000 shortened fuel pins, but cooled by the water coolant. The whole external zone has a diameter of 73.6 cm and the same height of 50 cm. Cladding of the fuel element of the reactor WWER-1000 has a diameter of 0.91 cm, while fuel pellet has a diameter of 0.786 cm. Chemical composition of the cladding is zirconium-1 % niobium alloy.

Calculation results show that optimal enrichment of uranium in the internal fast zone is 15-20 % in ^{235}U, while optimal enrichment of the uranium in the thermal zone comprises 4 % in ^{235}U. Therefore, fuel of the fuel elements in the fast internal zone is uranium dioxide, which has ^{235}U concentration of 15 %. Fuel of the fuel elements in the external thermal zone is uranium dioxide with enrichment of 4 % in ^{235}U. Fuel pins in both zones of the subcritical reactor are arranged in a square lattice with lattice pitch 1.275 cm. Such type of arrangement of fuel pins into assemblies, based on a square lattice, is typical for thermal nuclear reactors. Efficient beryllium reflector of 10 cm thickness is placed on the outside border of the reactor core. Tritium production, destined for saturation of the neutron generator target, takes place in the lithium units, which are allocated at the top and at the bottom of the beryllium reflector (see fig. 1). Tritium is produced, as usual, in nuclear reaction by neutron activation of ^{6}Li. This is possible with neutrons of any energy, and is an exothermic nuclear reaction yielding tritons according to the scheme [28]

$$^{6}\text{Li} \quad n \quad T \quad ^{4}\text{He} \quad E \qquad (5)$$

CALCULATION METHOD AND MCNP CODE

The computational experiment lies in direct simulation of source neutron transport through analyzed reactor system with the help of Monte Carlo method. This method is implemented in MCNP code, which was developed in the Los Alamos National Laboratory (LANL, N. Mex., USA) [31, 32].

The Monte Carlo method simulates the transport of particles (neutrons, photons, electrons) through media subjected to all possible interactions between particles and nuclei of media. The interactability of a specific type is formed on the base of macroscopic cross-sections. These cross-sections are calculated in accordance with the macroscopic data of a specific task and in accordance with the evaluated microscopic cross-section libraries. With the help of this method we can trace a specific particle (*e. g.* neutron) from its generation to absorption or leakage from the system. After accumulation of the data for a certain number of particle histories, the user can obtain particle distribution in the system and calculate other physical quantities.

The Monte Carlo method prevails over deterministic methods of transport equation solutions because it is well-suited for reactor systems with complex structure and geometry [33]. The accuracy of obtained results for Monte Carlo method depends on accuracy of input parameters and, of course, statistical uncertainty of simulated process. This uncertainty can be reduced by increasing the number of particle histories. As a rule, the accuracy for Monte Carlo calculations can reach 1-3 % in comparison with experimental data.

RESULTS AND DISCUSSION

Results for the homogeneous subcritical systems

The MCNP4c code can apply for two types of calculations: the estimation of neutron multiplication factor and the solving of tasks with different types of neutron sources. There is an option of simulation for time-dependent tasks in the second type of calculations. Initially, we launch series of criticality tasks for obtaining of the neutron multiplication factor for different reactor systems. Thereafter, the simulation of pulse propagation in nuclear reactor and obtaining of the amplification factor of subcritical system is carried out in *the most interesting* subcritical configurations under consideration. Thus, calculations for both the homogeneous and the heterogeneous systems were done with the help of MCNP4c Monte Carlo transport code, which employs the ENDF/B-VI nuclear data library.

The spherical systems for MCNP simulation had the following conditions: the first internal sphere is natural uranium with different radii *R*, and the second external spherical layer is water solution of 300 g ^{235}U with different ratios of concentrations N_U/N_H = 1:500; 1:600; 1:400. These two neutron multiplication zones are enclosed into the outer spherical beryllium reflector with different thicknesses *d*. The calculation results for neutron multiplication factor for the 1st and the 2nd cases are presented in tabs. 1 and 2 correspondingly.

The values of neutron multiplication factor k_{eff} for the subcritical system with the ratio of number of ^{235}U atoms to number of Hydrogen atoms in water solution H/U = 400 and the radius of the internal sphere of natural uranium R = 5 cm, were also calculated. These values comprise k_{eff} = 0.939, 0.968, 0.984 for the beryllium reflector thicknesses d = 20, 30, 40 cm, respectively.

Tables 1 and 2 show that with increasing thickness of the beryllium reflector, the neutron multiplication factor increases, as expected. Moreover, the obtained results show the decrease of the neutron multiplication factor following the increase of the thickness of internal sphere with natural uranium. The presented results also confirmed the well-known conclusion about optimality of ratio N_U/N_H = 1:500, from viewpoint of efficiency of neutron moderation.

Five reactor systems A-E (see tab. 3) were selected for study of the simulation of the neutron pulse propagation through subcritical zone. This particular choice of the systems A-D to be analyzed was made on the basis of tabs. 1 and 2, calculations in a way to meet the requirement of the slight subcriticality of the systems (k_{eff} ~ 0.98-0.99) and possibly to obey the rule of optimal H/U ratio. In addition, the system *E* (only internal zone – water solution) with k_{eff} = 0.999 was brought up into consideration for obtaining more information with regard to the dependence of energy amplification factor *vs*. the increase of neutron multiplication factor k_{eff}. The thicknesses of water solution is chosen on the basic principle that the effective multiplication factor of the system should not exceed unity.

For the calculation of pulse propagation through considered subcritical systems, for both the homogeneous and the heterogeneous systems, the following characteristics were used: the number of neutrons for one pulse $n_{pul} = 10^{14}$ neutrons, the pulse width is equal to 10^{-8} s, and the neutron energy is 14 MeV. The neutron angular distribution of the initial pulse is chosen to be isotropic. Time distribution of the initial pulse is stepwise. The average number of neutron histories, used in simulations, was chosen to be not less than one million, depending on the obtained uncertainties of

Table 1. The value of neutron multiplication factor k_{eff} for subcritical system with ratio of number of ^{235}U atoms to number of hydrogen atoms in water solution H/U = 500. R_1 – the radius of the internal sphere of natural uranium. R_2 – the outer radius of the second external spherical zone. d – the thickness of the beryllium reflector

R_1 [cm]	R_2 [cm]	k_{eff}			
		d = 20 cm	d = 25 cm	d = 30 cm	d = 40 cm
0	11.123	0.989	–	1.015	1.027
1	11.126	0.988	–	1.014	1.024
2	11.145	0.983	–	1.012	1.024
3	11.196	0.973	–	1.003	1.016
4	11.293	0.957	0.980	0.990	1.000
5	11.451	0.942	–	0.974	0.986
6	11.677	0.919	–	0.954	0.964
7	11.979	0.896	–	0.931	0.941

Table 2. The value of neutron multiplication factor k_{eff} for subcritical system with ratio of number of ^{235}U atoms to number of hydrogen atoms in water solution H/U = 600. R_1 – the radius of the internal sphere of natural uranium. R_2 – the outer radius of the second external spherical zone. d – the thickness of the beryllium reflector

R_1 [cm]	R_2 [cm]	k_{eff}		
		d = 20 cm	d = 30 cm	d = 40 cm
0	11.818	0.976	1.0013	1.0119
1	11.821	0.974	1.0012	1.0117
2	11.840	0.971	0.998	1.0084
3	11.882	0.965	0.992	1.003
4	11.969	0.954	0.980	0.989
5	12.109	0.938	0.968	0.979
6	12.313	0.922	0.953	0.963
7	12.586	0.899	0.930	0.942

Table 3. Geometry and material composition of the subcritical systems considered for simulation of the neutron pulse propagation. Water solution is chosen with the optimal ratio $N_U/N_H = 1:500$ from the viewpoint of the efficiency of neutron moderation

System	Internal zone	External zone	R_1 [cm]	R_2 [cm]	d [cm]	k_{eff}
A	Natural uranium	Water solution	4	11.293	25	0.980
B	Natural uranium	Water solution	4	11.293	30	0.990
C	Natural uranium	Water solution	5	11.451	40	0.986
D	Depleted uranium	Water solution	4	11.456	25	0.981
E	Water solution	–	11.123	–	23.1	0.999

Table 4. The parameters of pulse subcritical homogeneous assemblies

Parameters	System A $k_{eff} = 0.980$	System B $k_{eff} = 0.990$	System C $k_{eff} = 0.986$	System D $k_{eff} = 0.981$	System E $k_{eff} = 0.999$
n_{s1}(neutrons)	$4.715 \cdot 10^{15}$	$7.768 \cdot 10^{15}$	$9.506 \cdot 10^{15}$	$4.964 \cdot 10^{15}$	$7.196 \cdot 10^{16}$
φ_{s1}(neutrons per cm^2)	$2.345 \cdot 10^{13}$	$3.863 \cdot 10^{13}$	$3.026 \cdot 10^{13}$	$2.469 \cdot 10^{13}$	$4.628 \cdot 10^{13}$
$\overline{\varphi}_1$ [cm^{-2}]	$2.509 \cdot 10^{13}$	$3.981 \cdot 10^{13}$	$3.088 \cdot 10^{13}$	$2.611 \cdot 10^{13}$	$6.936 \cdot 10^{13}$
n_{s2} (neutrons)	$3.341 \cdot 10^{16}$	$5.783 \cdot 10^{16}$	$5.087 \cdot 10^{16}$	$3.652 \cdot 10^{16}$	–
φ_{s2} [cm^{-2}]	$2.085 \cdot 10^{13}$	$3.608 \cdot 10^{13}$	$3.087 \cdot 10^{13}$	$2.279 \cdot 10^{13}$	–
$\overline{\varphi}_2$ [cm^{-2}]	$2.355 \cdot 10^{13}$	$4.004 \cdot 10^{13}$	$3.309 \cdot 10^{13}$	$2.529 \cdot 10^{13}$	–
N_f (fissions)	$2.822 \cdot 10^{15}$	$4.843 \cdot 10^{15}$	$3.995 \cdot 10^{15}$	$3.069 \cdot 10^{15}$	$4.934 \cdot 10^{15}$
Q_f [MeV]	$5.104 \cdot 10^{17}$	$8.760 \cdot 10^{17}$	$7.227 \cdot 10^{17}$	$5.551 \cdot 10^{17}$	$8.926 \cdot 10^{17}$
G	364.6	625.7	516.2	396.5	637.57

calculations, with larger numbers used for heterogeneous calculations accordingly – at least five million neutron histories for heterogeneous case (see later). It is also important to provide the information on maximal statistical uncertainties of the calculated results. Here, we note that maximal standard deviation for criticality calculations does not exceed the value of 0.0005, whereas maximal standard deviation, for time domain analysis and the external source calculations, does not exceed the value of 0.001.

The following parameters were estimated in the MCNP calculations: the number of neutrons that cross sphere 1 (sphere of natural uranium) – n_{s1}, the number of neutrons for surface unit of sphere 1 – φ_{s1}, the averaged neutron flux in spherical zone 1 – $\overline{\varphi}_1$, the number of neutrons that cross sphere 2 (sphere of water solution of ^{235}U) – n_{s2}, the number of neutrons for surface unit of sphere 2 – φ_{s2}, averaged neutron flux in spherical zone 2 – $\overline{\varphi}_2$, the total number of fissions in sphere with water solution of ^{235}U – N_f, the fission energy, released in sphere with water solution of ^{235}U – Q_f, energy amplification factor $G = Q_f/Q_{pul}$, where Q_{pul} – the total energy of neutrons in initial pulse. Obtained results are presented in tab. 4. Let us remark here that spatial averaging of the quantities given in tab. 4 means volume or surface averaging for the corresponding volume zone, or surface quantities, respectively, whereas time averaging is understood in this case as averaging on the infinite time interval.

The typical time dependences of pulse propagation through subcritical assembly are presented in figs. 2 and 3 for System E with $k_{eff} = 0.999$. The time in figs. 2 and 3 is presented with using of shakes – special unit of time measurement for MCNP code (1 shake = 10^{-8} s). The damping of averaged neutron flux, in the zone of water solution of ^{235}U, is demonstrated in fig. 2. The neutron pulse propagation across the outer spherical surface with water solution of ^{235}U, is demonstrated in fig. 3. Let us also note that the calculated neutron flux for all computer time analysis simulations was normalized, or rather averaged, in such a way that all neutron history simulation results are falling into the same time interval/bin of the width of 1 shake form a single unified averaged result. All the below figures in the present paper are prepared with the help of MCNP plotter [31]. Obviously, the times of damping are different for every considered system and depend on k_{eff}.

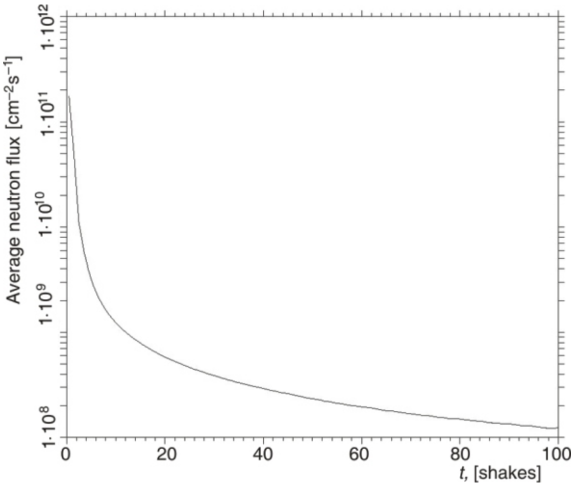

Figure 2. The time dependence of the averaged neutron flux in zone of water solution of ^{235}U for System E with $k_{eff} = 0.999$ in case of one initial pulse

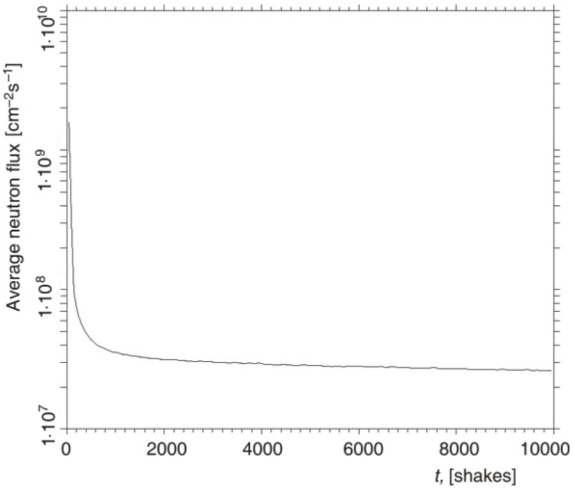

Figure 3. The time dependence of the neutron flux across outer boundary sphere 1 of water solution of ^{235}U for system *E* with $k_{\mathrm{eff}} = 0.999$ in case of one initial pulse

For example, the total time for damping of neutron pulse in System *D* is 0.04 s, and in System *C* is 0.12 s. The obtained results for simulation of System *E* with $k_{\mathrm{eff}} = 0.999$: energy amplification factor $G = 1351$ and total time for damping of neutron pulse is 0.21 s.

Therefore, the obtained results show that considered subcritical systems with sufficiently small amount of fissile material (~300 g ^{235}U) could be used as effective amplifier of source neutron even in the case of one pulse.

It is logical to assume that the energy amplification factor from the series of pulses (providing that the pulse repetition frequency would be sufficiently high and the time between following pulses would be much less than the time for damping of neutron pulse) should greatly exceed the amplification factor for one pulse. Calculation results for the series of pulses, given below, confirm these assumptions. The simulation of propagation of one hundred pulses, with relative pulse duration of 10^{-4} s through the System *B*, is carried out. The averaged neutron fluxes across the spheres 1 and 2, see fig. 4 (the calculated neutron flux normalized on time interval = 100 shakes), and 5 (the calculated neutron flux normalized on time interval = 10000 shakes), the energy amplification factors, were calculated.

Figure 5 shows that the number of neutrons and, therefore, amplification factor increases significantly in the case of one hundred pulses compared with the case of a one pulse. Besides, the saturation was not reached after one hundred pulses. The increase of amplification after series of one hundred pulses is caused by the accumulation of delayed neutrons from previous pulses.

Thus, using of the subcritical systems for amplification of neutron pulses from external neutron source could be applicable, besides, the value of amplification could be enough for practical application (for example, production of radiopharmaceuticals).

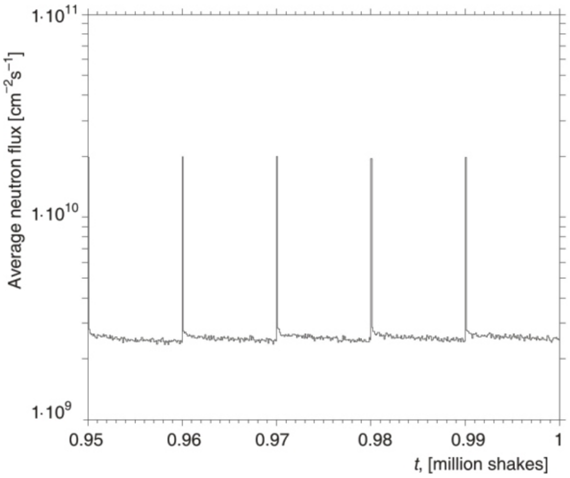

Figure 4. The time dependence of averaged neutron flux across sphere 1 of natural uranium for System *B* in case of one hundred neutron pulses. The figure shows the time interval for the last five pulses of series

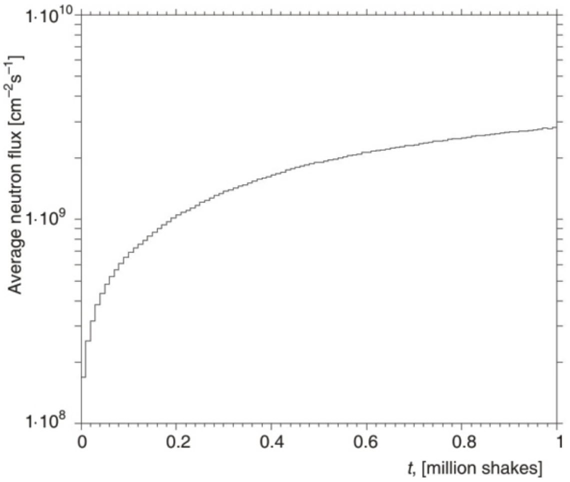

Figure 5. The time dependence of averaged neutron flux in zone of water solution with ^{235}U for System *B* in case of one hundred neutron pulses

Results for the heterogeneous subcritical systems

The calculation of pulse propagation through the considered subcritical heterogeneous systems the following characteristics have been used: the number of neutrons for one pulse $n_{\mathrm{pul}} = 10^{14}$ neutrons, the pulse width equal to 10^{-8} s, and the neutron energy 14 MeV. At least five million neutron histories were modeled for all calculations. At first, the propagation of one neutron pulse through the presented subcritical model (see fig. 1) was done and the averaged neutron fluxes in internal and external zones, due to one pulse for time interval = = 100 shakes, are shown in fig. 6 (the calculated neutron flux normalized on time interval = 1 shake) and fig.

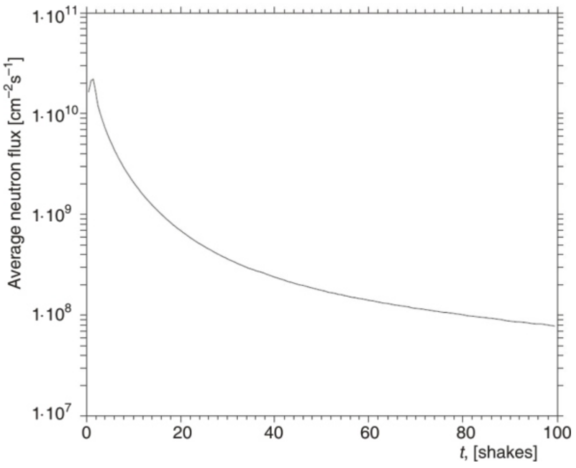

Figure 6. The time dependence of averaged neutron flux in the *internal zone* of subcritical reactor in case of one neutron pulse

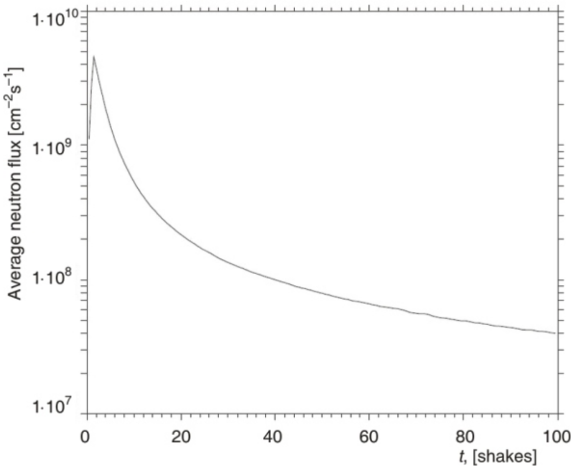

Figure 7. The time dependence of averaged neutron flux in the *external zone* of subcritical reactor in case of one neutron pulse

7 (the calculated neutron flux normalized on time interval = 1 shake).

The comparison of homogeneous subcritical system (see fig. 2) and heterogeneous subcritical system (see fig. 6) demonstrate that there is a decreasing of level of neutron flux for the case of heterogeneous model. The figs. 6 and 7 also show that there is a little peak of averaged neutron flux at first 10 shakes for heterogeneous model. This effect is explained by the time of passage of pulse from the target to the internal and external zones.

Secondly, the propagation of one neutron pulse through the presented subcritical model was done for time interval = 10000 shakes. The averaged neutron fluxes for internal and external zones are shown in fig. 8 (the calculated neutron flux normalized on time interval = 100 shakes) and fig. 9 (the calculated neutron flux normalized on time interval = 100 shakes) respectively.

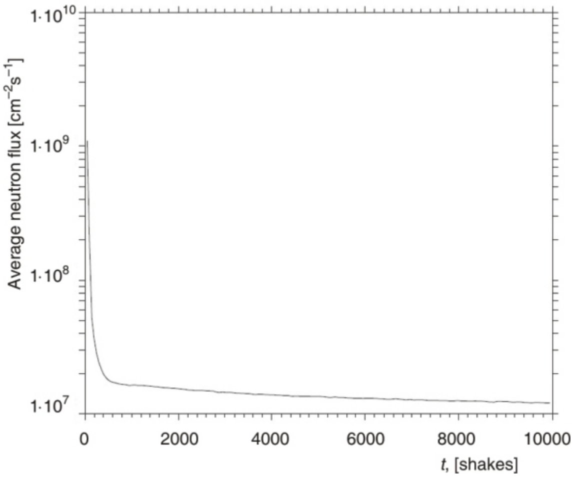

Figure 8. The time dependence of averaged neutron flux in the *internal zone* of subcritical reactor in the case of one neutron pulse

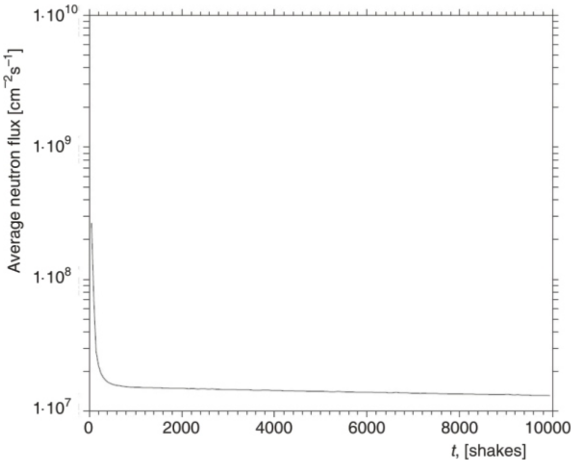

Figure 9. The time dependence of averaged neutron flux in the *external zone* of subcritical reactor in the case of one neutron pulse

The figs. 8 and 9 show that there are differences between the averaged neutron fluxes in the internal and external zones with regard to time and to the level of neutron flux. The averaged neutron flux is higher immediately after the pulse in the internal zone than in the external zone. But the averaged neutron flux is higher at the flat plateau of pulse (after 4000-5000 shakes) in the external zone than in the internal zone. It could be explained by the fact that there is more intense thermal-neutron fission, due to thermal neutron spectrum, in the external zone.

The simulation of propagation of one hundred pulses, with pulses duty cycle 10^{-4} s through the heterogeneous subcritical system, is also carried out within the scope of the presented paper. The averaged neutron fluxes for internal and external zones for time interval = 1000000 shakes are shown in fig. 10 (the calculated neutron flux normalized on time interval = 10000 shakes) and fig. 11 (the calculated neutron flux nor-

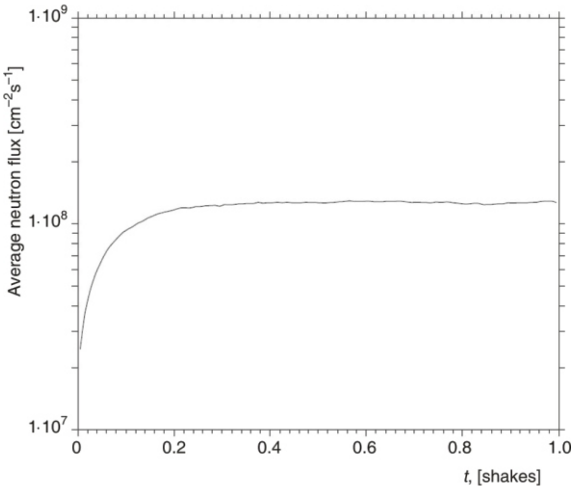

Figure 10. The time dependence of averaged neutron flux in the *internal zone* of subcritical reactor in the case of one hundred neutron pulses

malized on time interval = 10000 shakes), respectively.

The figs. 10 and 11 show that there are not essential distinctions between the averaged neutron fluxes in the internal and external zones with regard to time and to the level of neutron flux. In addition, the comparison of homogeneous subcritical system (see fig. 5) and heterogeneous subcritical system (see figs. 10 and 11) demonstrate that there is a decrease of level of neutron flux for the case of heterogeneous model.

CONCLUSIONS

Therefore, the considered pulse subcritical amplifier of neutron flux could combine, on the one hand, the advantages of subcritical assembly driven by external neutron source with, on the other hand, the advantages of conventional pulse nuclear reactor. Such systems will maintain the following advantages of traditional ADS, that operate in a steady state mode: subcritical (safe) operating mode; the possibility for transmutation of nuclear waste from conventional nuclear reactors, the potential use of different types of nuclear fuel. Also, such systems will maintain the following advantages of pulse reactor: high power and high-intensity neutron flux for short time intervals. The proposed pulse systems can be used for a number of research goals and experiments. It should also be emphasized that construction of a pulsed neutron source driven by high-intensity neutron generator could be cheaper and easier than construction of *conventional* ADS driven by linear proton accelerator in a steady state mode. Also, there is a significant experience for construction and operation of high-intensity neutron generators in Institute for Nuclear Research (Ukraine) [30].

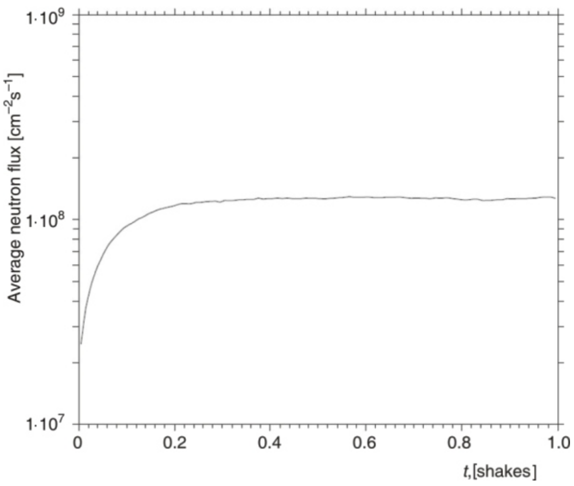

Figure 11. The time dependence of averaged neutron flux in the *external zone* of subcritical reactor in the case of one hundred neutron pulses

Thus, we have considered a subcritical system that can be described as a *booster* in which the power pulse is initiated by the initial neutron pulse, or a series of neutron pulses, from external source, and the neutron multiplication in the core is reducing during the damping fission chain reaction after power cutoff. In this case, the neutron pulse width is longer than the external source width to the order of $\tau/(1-k)$, where τ – lifetime of prompt neutrons, k – effective neutron multiplication factor. The number of neutrons generated in a pulse exceeds the number of neutrons of the source by $1/(1-k)$ times.

The presented calculations have shown that the considered pulse subcritical systems can be successfully used as effective amplifiers of neutron flux from initial source. Thus, the calculated amplification factor, from series of neutron pulses, greatly exceeded the amplification factor from one neutron pulse, which, however, is quite natural. In general, the conducted investigations and the obtained results show that using of MCNP code allows a reliable calculation of realistic subcritical nuclear systems and propagation of the series of neutron pulses from external neutron source through these systems. In other words, we could consider the temporal dynamics of occurring processes in the subcritical system.

The homogeneous and heterogeneous subcritical systems were considered in the present paper. The neutron-physical model of two-zone subcritical reactor with different neutron spectra was developed with the help of MCNP4 Monte Carlo code. The propagation of one neutron pulse and series of one hundred neutron pulses through different subcritical models was calculated. The modeling results indicate that there is an achievement of a stable, high level of neutron flux, caused by the accumulation of delayed neutrons from previous pulses in series of one hundred pulses, for both homogeneous and heterogeneous systems. The presented results indicate that the zone with thermal neutron spectrum has a good potential for amplification of neutron flux due to more intense thermal-neutron fission.

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AUTHORS' CONTRIBUTIONS

V. Babenko – performing of the MCNP calculations and simulations, preparation of the figures, writing and editing of the paper.

V. Pavlovych – initialization of the idea for this research, statement and formulation of the problem, writing and editing of the paper, scientific supervision and coordination.

V. Gulik – performing of the MCNP calculations and simulations, preparation of the figures, writing and editing of the paper.

All authors extensively interacted, exchanging ideas, especially during th preparation of manuscript.

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ИМПУЛСНИ ПОТКРИТИЧНИ ПОЈАЧАВАЧ НЕУТРОНСКОГ ФЛУКСА ПОКРЕТАН ГЕНЕРАТОРОМ НЕУТРОНА ВИСОКОГ ИНТЕНЗИТЕТА

Поткритични реактор покретан екстерним неутронским извором може наћи своју употребу као користан инструмент у модерној примени нукеларне енергије која захтева озрачивање различитих материјала неутронским флуксевима високих енергија и интензитета (на пример, код трансмутације нуклеарног отпада, у производњи радиофармацеутика, итд). У овом раду разматрано је простирање неутронског импулса кроз поткритични нуклеарни систем. Применом Монте Карло MCNP4c транспортног програмског пакета испитан је једноставан модел хомогеног поткритичног система и модел двозонског поткритичног реактора. За ове моделе симулирано је простирање једног иницијалног импулса неутрона и серије од сто импулса. Прикупљени су и анализирани резултати за мултипликациони фактор неутрона, неутронски флукс, фактор појачања енергије и укупну енергију неутрона у иницијалном импулсу. Анализа приказаних прорачуна показује да се разматрани импулсни поткритични системи могу успешно користити као ефективни појачавач и неутронског флукса од иницијалног извора. Резултати моделовања указују на остваривање стабилног неутронског флукса високог нивоа изазваног акумулацијом закаснелих неутрона од претходних импулса из серије од сто узастопних импулса и код хомогених и код хетерогених система.

Кључне речи: систем покретан акцелератором, Монте Карло прорачун, импулсни истраживачки реактор, D-T неутронски генератор