COMPUTATION STUDYING OF THE NEUTRON YIELD FROM THE NEUTRON-PRODUCTION TARGET IRRADIATED WITH ELECTRONS

I.M. Prokhorets, S.I. Prokhorets, Y.V. Rudychev, A.I. Skrypnik, D.V. Fedorchenko, M.A. Khazhmuradov *

National Science Center "Kharkov Institute of Physics and Technology", 61108, Kharkov, Ukraine

It is considered the modeling of neutron yield from the targets with high atomic numbers irradiated with accelerated electrons. Modeling results from the MCNPX and GEANT software are compared with existing experimental results and deterministic calculations.

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1. INTRODUCTION

Problems of increasing of quality, reliability and longevity of technical devices, equipment, components, materials and complex constructions are of special importance in modern conditions. Solving of these problems significantly depends on efficiency of control devices and methods. Methods of non-destructive control are the most interesting and promising ones in the industrial conditions. One of such methods is material and device defectoscopy using gamma- and roentgen-radiation or bremsstrahlung from the electron accelerators. Neutron radiography is one of the non-destructive control methods that are being strongly developed in the highly industrialized countries. There are no neutron-radiography facilities in the Ukraine now, though its development and production will allow keeping up to modern development of science and technics. Practical implementation of the neutron radiography method will give new non-destructive control possibilities in aerospace, fuel and atomic industry. Neutron radiography and neutron tomography will give new instruments for testing of many products with both light and heave elements and their isotopes in composition.

Advantage and benefits of the non-destructive control method with neutron usage come from large cross-sections of the neutron interaction with some chemical elements compared with those one for gamma-quanta (Fig. 1) [1].

Qualitative advantage of neutron beam is shown in possibility of obtaining more contrasting and informative image, comparing with radiogram one, e.g. images of fuel elements in the reactor experimental or working device (Fig. 2).

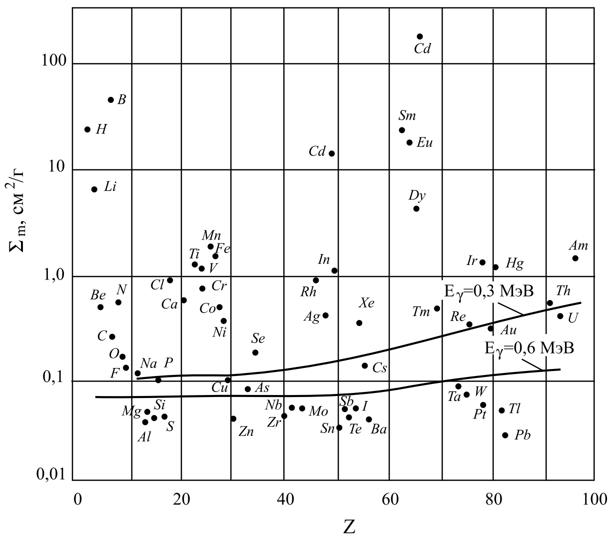

Fig.1. *Mass cross-sections of the thermal neutrons ($E_n = 0,025$ eV) and γ-radiation with different energies versus matter atomic number (for natural isotope composition)*

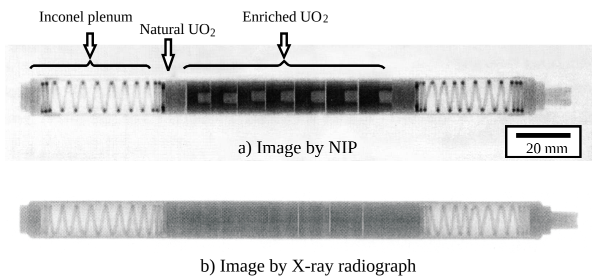

Fig.2. *Neutronogram and radiogram of the experimental device with fuel elements*

Neutron target is of the most important parts of the neutronography facility. Neutron target is device where neutrons are born as the result of radiation

*Corresponding author. E-mail address: khazhm@kipt.kharkov.ua

source interaction with target media. In our case radiation source is electron from the linear accelerator.

In this article results of modeling of neutron yield for wide energy range of electrons irradiating the target are given and our modeling results are compared with analytical ones [2, 3]. This work is necessary because of absence of modern data for modeling and calculation of new physical and technical facilities, where nuclear radiation is used.

2. RESULTS OF THE COMPUTER EXPERIMENT

Basis of the computer experiment is computer with proper characteristics and software, the user familiar with. For our work we used processor AMD Athlon64 1.8 MHz and software MCNPX.

Tasks considered in our work concern radiation passing through the matter and their solving is being studied very intensively for a long time. Originally they were solved using Boltzman transport equation but the largest success was achieved when probabilistic methods were applied. One of such methods is Monte Carlo method, used for solving of different mathematical tasks by sampling the random variables [4]. In such a way it is simulated the whole track of the nuclear particle - from the "birth" place to the "death" (capture, scattering, escaping from the modelling object etc.) Different versions of MCNPX simulate neutron transfer in 3D-geometry using the random variable sampling [5]. The most important in this software is taking into account the continuous energy dependence of the modeled parameters. Such method is very realistic one and in some articles is called "theoretical experiment".

As many other Monte Carlo programs, MCNPX uses the Lehmer method for random variables sampling. Pseodurandom number sequence I_n is generated by $I_{n+1} = mod(MI_n, 2^{48})$, where M - is random number multiplier, and 48-bit integers and 48-bit floating point mantissas are assumed. The default value of M, which can be changed by the user, is $M = 5^{19} = 19073486328125$.

Then pseodurandom number is then $R_n = 2^{-48} I_n$, and starting pseodurandom number for each sampling is $I_{n+S} = mod(M^S I_n, 2^{48})$, where S - pseodurandom number stride. Thus, for MCNPX algorithm period we obtain $P = 2^{46} \approx 7,04 \cdot 10^{13}$, or after some modifications of algorithm we obtain $P = 2^{48} \approx 2,81 \cdot 10^{14}$.

Sufficiently great algorithm period provides for MCNPX stable work with random variables. MCNPX warns about number of histories, where stride S was exceeded, and also warn about algorithm period exceeding. In computer experiment, considered in this article we used electron beam from the linear accelerator. Neutrons were obtained via process of direct electron interaction with target material and as the result of double conversion process electron → bremsstrahlung gamma-quantum → neutrons after $(\gamma, n), (\gamma, 2n), \ldots, (\gamma, xn)$ reactions.

All neutron-producing targets have the similar form with cross-section $4.5 \times 4.5\ in^2$. For easy comparing of the modeling results and experimental data the thickness and the shape of targets were the same. Modeling was done using MCNPX code, based on Monte Carlo method, with taking into account the neutron-producing target real parameters. Modeling results obtained in this article are compared with results from [6], modeling results obtained using GEANT [7, 8] and theoretical investigations [2, 3]. Comparing results is given in table. According to this dependence of neutron yield from tantalum and lead targets on accelerated electron energy is shown at Fig. 3.

Dependence of neutron yield from targets with large atomic numbers on the target thickness is shown at Fig. 4. Results was obtained by MCNPX modeling.

From the Fig. 5 it is obvious that effective energy of the electrons in the neutron-producing target doesn't exceed 100 MeV, so to estimate neutron yield it is sufficient to consider electron energy greater or equal 100 MeV (electron beam energy is 200 MeV at Fig. 4).

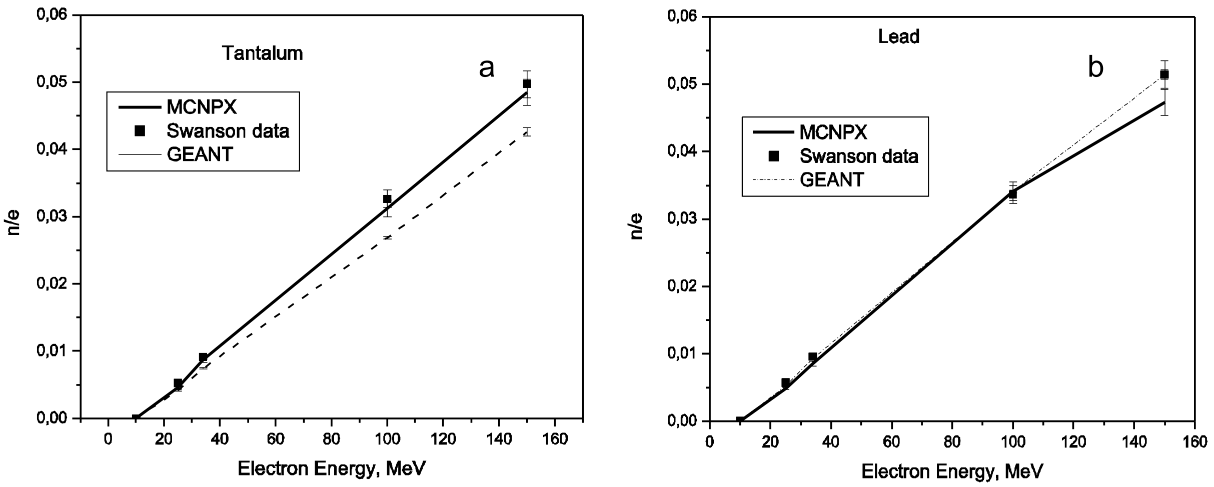

Fig.3. *Neutron yield from the tantalum (a) and lead (b) neutron-producing targets versus electron energy*

Comparison of simulation and experiment

Target Material	Target thickness, cm	Target square shape, in	Beam energy, MeV	Yield, n/e	Yield, n/e MCNPX simulation	Ratio, sim./exp.	Yield, n/e GEANT4 simulation	Ratio, sim./exp
Experiment								
Cu	5.93	4.5×4.5	21	5.90E-04	6.60E-04	0.89	4.89E-04	0.83
Cu	5.93	4.5×4.5	28	2.14E-03	1.85E-03	0.87	1.58E-03	0.74
Cu	5.93	4.5×4.5	34.3	3.34E-03	3.20E-03	0.96	2.58E-03	0.77
Pb	0.519	4.5×4.5	19	7.00E-04	6.90E-04	0.99	8.10E-04	1.16
Pb	0.519	4.5×4.5	28	1.66E-03	1.35E-03	0.82	1.75E-03	1.05
Pb	0.519	4.5×4.5	34.3	2.10E-03	1.53E-03	0.73	2.19E-03	1.04
Pb	3	4.5×4.5	19	2.46E-03	2.20E-03	0.9	2.39E-03	0.97
Pb	3	4.5×4.5	28	6.70E-03	5.41E-03	0.81	6.14E-03	0.92
Ta	0.374	4.5×4.5	19	5.18E-04	6.02E-04	0.86	6.49E-04	1.25
Ta	0.374	4.5×4.5	28	1.38E-03	1.47E-03	0.94	1.48E-03	1.07
Ta	0.374	4.5×4.5	34.3	1.80E-03	1.82E-03	0.99	1.80E-03	1
Swanson								
Ta	8.5	4.5×4.5	10	1.70E-05	1.72E-05	0.99	2.60E-05	1.53
Ta	8.5	4.5×4.5	25	5.29E-03	4.61E-03	0.87	4.19E-03	0.79
Ta	8.5	4.5×4.5	34	9.16E-03	8.68E-03	0.95	7.50E-03	0.82
Ta	8.5	4.5×4.5	100	3.27E-02	3.12E-02	0.95	2.69E-02	0.82
Ta	8.5	4.5×4.5	150	4.97E-02	4.85E-02	0.98	4.26E-02	0.86
Pb	10	4.5×4.5	10	3.22E-05	3.00E-05	0.93	4.10E-05	1.27
Pb	10	4.5×4.5	25	5.73E-03	4.87E-03	0.85	5.25E-03	0.92
Pb	10	4.5×4.5	34	9.65E-03	8.53E-03	0.88	9.29E-03	0.96
Pb	10	4.5×4.5	100	3.36E-02	3.10E-02	0.92	3.41E-02	1.02
Pb	10	4.5×4.5	150	4.36E-02	4.73E-02	0.92	5.14E-02	0.62

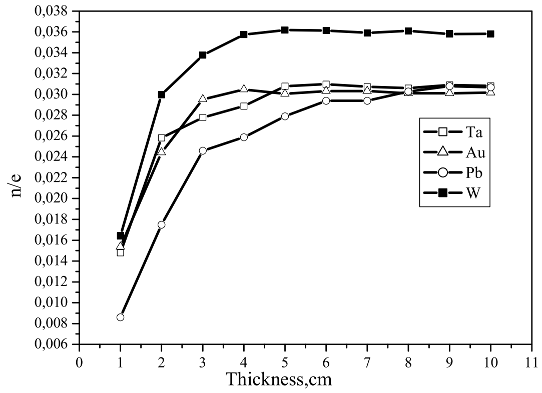

Fig.4. *Neutron yield from different targets versus target thickness (electron beam energy 200 MeV)*

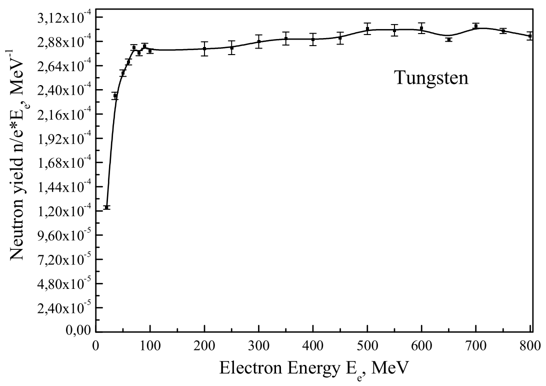

Fig.5. *Neutron yield per energy from tungsten target versus electron energy*

3. CONCLUSIONS

Results of the investigations, carried out in this article, show that Monte Carlo method can be used for modeling of the neutron born process in the irradiated with electrons targets from materials with large atomic numbers, if the database on the electronuclear interactions for these materials exist.

The distinctive feature of our studying is usage of the targets from non-fission materials. Presence of the cross-section databases for these materials allows using MCNP code without additional physical models for simulation processes.

If you have no cross-section databases, you'd used GEANT4 software, where it is possible to change the parameters of the physical models, as you need, and unlike MCNP, GEANT4 is free distributed software.

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РАСЧЕТНЫЕ ИССЛЕДОВАНИЯ ВЫХОДОВ НЕЙТРОНОВ С НЕЙТРОНОПРОИЗВОДЯЩЕЙ МИШЕНИ, ОБЛУЧАЕМОЙ ЭЛЕКТРОНАМИ

И.М. Прохорец, С.И. Прохорец, Е.В. Рудычев, А.И. Скрыпник, Д.В. Федорченко, М.А. Хажмурадов

Рассмотрено применение программных кодов MCNPX и GEANT для расчета выхода нейтронов из различных нейтронопроизводящих мишеней, использующих ускоренные электроны из ускорителя. Показано, что расчеты по методу Монте-Карло, выполненные при помощи программного кода MCNPX, хорошо согласуются с имеющимися данными и данными аналитических расчетов.

РОЗРАХУНКОВІ ДОСЛІДЖЕННЯ ВИХОДІВ НЕЙТРОНІВ З НЕЙТРОНОУТВОРЮЮЧОЇ МІШЕНІ, ЩО ОПРОМІНЮЄТЬСЯ ЕЛЕКТРОНАМИ

І.М. Прохорець, С.І. Прохорець, Є.В. Рудичев, А.І. Скрипник, Д.В. Федорченко, М.А. Хажмурадов

Розглянуто використання програмних кодів MCNPX та GEANT для розрахунку виходу нейтронів з нейтроноутворюючих мішеней, що використовують прискорені електрони з прискорювача. Доведено, що розрахунки з використанням методу Монте-Карло та програмного коду MCNPX добре узгоджуються з експериментом і аналітичними розрахунками.