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ABSTRACT

The paper deals with methodological questions arising in connection with the measurement of neutron leakage from spherical assemblies simulating multiplication zones of controlled thermonuclear reactor blankets. A boron tank, which has a number of advantages over other detectors of total absorption, served as the neutron detector. On the basis of the calculations performed, it was shown that surrounding the assembly under study with a layer of borated water serving as a neutron detector does not affect the processes of 14 MeV neutron multiplication. The boron tank has a constant recording efficiency for neutrons in the 0-6 MeV energy range. A procedure is proposed for measuring the leakage of neutrons from spherical samples with a central 14 MeV source.

1. INTRODUCTION

The need for improved accuracy in calculating the neutron parameters of thermonuclear reactor blankets lends immediate interest to experimental studies on assemblies simulating blanket zones. An important place in these experiments is taken by the study of the multiplication efficiency of 14 MeV neutrons in various materials. The results obtained in these studies afford a means of evaluating the accuracy of the data used in calculations and, accordingly, of assessing the blanket parameters more reliably.

As is well known, in the breeding zone high-energy fusion neutrons generate secondary neutrons by the (n,2n), (n,3n), or (n,f) reactions. The additional neutrons can be used for increasing the rate of accumulation in the blanket of such materials as tritium, uranium-233 and plutonium-239. Among the most efficient breeders are uranium-238, lead, thorium and a number of other elements having large cross-sections for reactions yielding neutrons.

This paper deals with questions of method relating to the measurement of the number of neutrons emitted by a physical assembly

simulating the breeding zone of the blanket of a thermonuclear reactor. Materials with a large atomic number, such as lead, bismuth, uranium and thorium were regarded as breeders. We were interested in the leakage neutron spectrum only from the point of view of choosing a method for measuring the total stream of neutrons.

2. MEASUREMENT OF NEUTRON LEAKAGE

The number of neutrons escaping from the assembly (for the sake of simplicity, in what follows we shall consider assemblies in the shape of spheres), normalized in terms of one 14 MeV neutron from a source, is determined approximately by the expression:

$$M = T + (1 - T)\mu, \qquad (1).$$

where $T = e^{\Sigma_{in}.\ell}$ is the transmission of 14 MeV neutron by an assembly of thickness ℓ; Σ_{in} is the macroscopic cross-section of the inelastic interaction; μ is the neutron yield in a single inelastic interaction event.

In writing expression (1), we made the assumption that:

- elastic scattering does not alter the direction of movement and the energy of the neutrons from the source,
- secondary neutrons from an inelastic interaction do not produce multiplication,
- the neutrons are not absorbed in the assembly.

It is known that 14 MeV neutrons are scattered by heavy nuclei mainly in a forward direction, a fact that justifies the first assumption. Disregard of secondary neutron multiplication is valid for assemblies made of non-fissionable elements (lead, bismuth and a number of others) of any thickness, since the threshold of the (n,2n)-reaction for these materials is higher than the maximum energy of the secondary neutron spectrum of the (n,2n)-reaction (according to data of the ENDL library [1]. In the case of multiplying arrays (uranium, thorium), it is necessary to introduce into expression (1) an additional factor allowing for multiplication; for small thickness, this factor is close to unity.

Essentially, measurement of the value of (M) is similar to measurement of the strength of radionuclide neutron sources and for this reason the same experimental methods can be used here.

As early as the 1950s, when measurement of the multiplication of 14 MeV neutrons in spherical samples was introduced for the purpose of determining the cross-sections of (n,2n)-reactions, use was made of all-wave neutron counters (see, for example, Ref. [2]). The so-called "long counter" of Hanson and McKibben [3], whose efficiency depended only slightly on neutron energy was widely used. In later studies (for example, Ref. [4]) it was shown that in the calibration of sources, account must be taken of the fact that the neutron recording efficiency of the long counter is not constant in the 0.002-1 MeV and 5-14 MeV energy ranges. This, in turn, requires knowledge of the spectrum of emitted neutrons, which inevitably affects the accuracy of neutron yield measurement.

With the development of various methods of neutron spectrometry and, in particular, of the time-of-flight method, the significance of "long counters" diminished considerably and they were displaced almost entirely from precise neutron-physics experiments. Down to the present, however, one of the varieties of all-wave counters - total absorption detectors (manganese bath, boron tank, water tank, etc.) - is used for the calibration of sources. These methods consist essentially in the following: the source under study is placed in an absorbing medium of sufficiently large size, in which thermal neutrons are slowed down and then captured. The strength of the source is determined by integration of the rate of absorption over the volume of the moderator. Since the neutrons are captured in the thermal energy region, knowledge of the neutron spectrum of the source is not required, and integration over a 4π solid angle obviates the need to allow for their angular distribution.

3. THE EFFECT OF AN EXTERNAL MODERATOR ON NEUTRON MULTIPLICATION IN AN ASSEMBLY

An example of the use of the total absorption method in studies on the multiplication of 14 MeV neutrons in the blanket materials of a thermonuclear reactor may be found in the paper of Basu et al. [5],

dealing with the measurement of neutron leakage from beryllium assemblies. A neutron source (E ≃ 14 MeV) was placed at the centre of a rectangular beryllium assembly which, in turn, was positioned at the centre of a large beryllium cube. The moderation of the fast neutrons and the capture of the thermal neutrons took place mainly in the hydrogen nuclei. The thermal neutron flux was measured by boron counters. Along with the main experimental results, the authors report a negligibly small effect of the moderator on the yield of neutrons from the assemblies. However, no quantitative evaluations in support of this fact are presented.

Obviously, the main problems arising in the attempt to use the total absorption method for measuring the multiplication efficiency of neutrons from the assembly is the effect of the external layer of the moderator on the yield of neutrons from the assembly. As a result of scattering in the external moderator, the neutrons escaping from the assembly can return to the assembly and be absorbed in it or give rise to further multiplication. The flow of neutrons into the assembly depends on the geometry of the experiment as well as on the type of moderator and breeder.

A schematic representation of the measurement system is shown in Fig. 1, where the spherical test sample with a central point-source of 14 MeV neutrons is placed inside a moderator of fairly large size. Regulation of the neutron flow for optimization of the experiment consists in varying the dimensions of the inner cavity in the moderator. By increasing its radius we reduce the angle of visibility of the assembly with the internal surface of the moderator and thus reduce the probability of scattered neutrons coming into contact with it. However, the possibilities of such regulation are limited because the large dimensions of the cavity entail a large increase in the volume of the moderator and, as a result, make the experimental arrangement considerably more complicated. In the choice of a moderator, preference may be given to water and some hydrogenous substances, such as paraffin or polyethylene. Water, which has a high moderating capacity, causes a rapid drop in the energy of the neutrons, bringing them below the threshold of the multiplication reactions on the elements entering into the composition of the assembly. The use of an aqueous moderator permits a sufficiently compact installation to be set up - a fact which is of no little importance from the point of view of experimental technique.

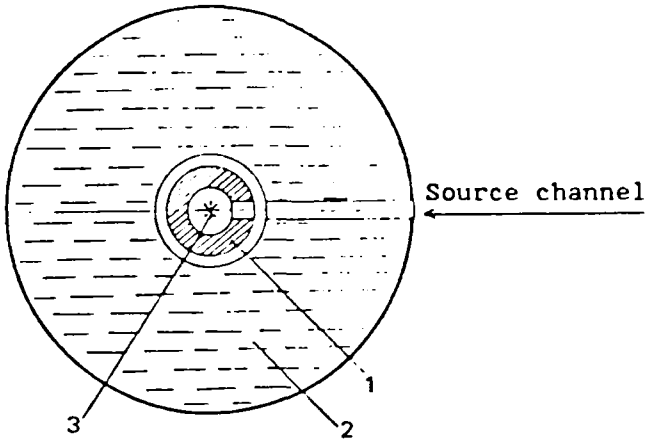

Fig. 1. Schematic representation of the experimental system for measuring the multiplication factor of 14 MeV neutrons: 1. Multiplier; 2. External moderator; 3. Source of 14 MeV neutrons.

The most widespread total absorption methods are those involving the use of the magnesium bath, the boron tank and the water tank. We shall compare these methods in terms of their effect on the assembly under study. Since in all the techniques referred to, the neutrons are moderated as a result of scattering in hydrogen, the spatial distributions of the moderated neutrons are similar to each other, other conditions being equivalent. For this reason the flow of thermal neutrons into the assembly will be determined only by the absorption of the neutrons on the moderator. The maximum macroscopic cross-section of absorption is found in an aqueous solution of boric acid, so that we can fix our choice precisely on this type of moderator.

In going over to a new type of moderator or new test materials, it is necessary to find the basis of a methodology for measurements of this kind. The evaluation of the suitability of a moderator must be considered from various points of view, including that of the effect of an external moderator on the multiplication of neutrons in the assemblies studied.

We consider below the results of calculations performed on the basis of experiments involving the use of a boron tank to measure the neutron yield from assemblies made from beryllium, lead or depleted uranium. In what follows, we shall designate the volume of a boron tank as a total absorption detector (TAD). The characteristic dimensions of the TAD and its material composition are based on a real installation. The arrangement of the TAD is similar to that shown in Fig. 1. The diameter of the internal cavity is assumed to be 40 cm. A cadmium filter

1 mm thick is located at the interface of the internal cavity and the water. The thickness of the moderating layer is 45.0 cm. Th content of the isotope ^{10}B in water, 7.4×10^{19}, corresponds to the maximum solubility of boric acid of natural isotopic composition in water at a temperature of 20°C. The calculations were performed according to the BLANK program [6], in which the neutron transport equation is solved by the Monte Carlo method. Sixty-five group constants were prepared on the basis of evaluated data files of the ENDL library [7] in the 0.1-15 MeV range (the constants for beryllium were taken from the UKNDL library) and 21 group constants [8] for calculations in a P_1 approximation below 0.1 MeV. Tables 1-3 present data on neutron leakage and the main reaction rates in the tested assemblies, surrounded by water and without water.

The error in the calculation of functionals is estimated at 2-5%. The effect of the action of water goes beyond the limits of error only in the case of the number of fission of uranium-235 and the number of absorbed neutrons in uranium assemblies, i.e. the parameters sensitive to thermal neutrons emerging from the moderator. However, this has an insignificant effect on total neutron leakage, since the variation in this value does not exceed 2%. Thus, to within the error limit of the calculation, the influence of water on the multiplication of 14 MeV

Table 1. Rates of main processes in beryllium assemblies.

Parameter	Wall thickness of assembly					
	δ = 1.5 см		δ = 3 см		δ = 8 см	
	Without water	With water	Without water	With water	Without water	With water
1. No. of (n,2n) reactions (normalized in terms of 1 neutron from a source)	0.1073	0.1071	0.2150	0.2149	0.6017	0.6072
2. No. of (n,α) reactions	0.0037	0.0037	0.0675	0.0676	0.0147	0.0148
3. No. of (n,t) reactions	0.0042	0.0043	0.0104	0.0103	0.0434	0.0427
4. Leakage of neutrons from assembly	1.121	1.120	1.223	1.224	1.524	1.553

NB: Radius of internal cavity in beryllium assemblies: 4.5 cm.

Table 2. Rates of main processes in lead assemblies.

Parameter	Wall thickness of assembly			
	δ = 3 см		δ = 6 см	
	Without water	With water	Without water	With water
1. No. of (n,2n) reactions	0,1761	0,1759	0,3271	0,3302
2 Leakage of neutrons from assembly	1,193	1,203	1,347	1,357

NB: The dimensions of the assemblies were as follows:
(1) δ = 3 cm; r_{int} = 9 cm; R_{ext} = 12 cm;
(2) δ = 6 cm; r_{int} = 3 cm; R_{ext} = 12 cm.

Table 3. Rates of main reactions in uranium assemblies (isotopic composition of uranium: ^{238}U - 99.6%; ^{235}U - 0.4%).

Parameter	Wall thickness of assembly					
	δ = 1 см		δ = 4 см		δ = 8 см	
	Without water	With water	Without water	With water	Without water	With water
1. No. of ^{238}U fissions	0,068	0,066	0,2853	0,2892	0,523	0,524
2. No. of ^{235}U fissions	$0{,}715.10^{-3}$	$1{,}106.10^{-3}$	$0{,}492.10^{-2}$	0,000727	0,016	0,020
3. No. of secondary neutrons due to fission	0,283	0,289	1,137	1,157	2,002	2,016
4. No. of (n,2n) reactions (total for ^{238}U and ^{235}U)	0,052	0,055	0,1372	0,1395	0,229	0,226
5. No. of (n,3n) reactions (same as for 4)	0,023	0,023	0,0773	0,0776	0,125	0,123
6. No. of absorbed neutrons	0,072	0,083	0,369	0,414	0,804	0,871
7. No. of neutrons leaking from assembly	1,308	1,325	2,064	2,059	2,681	2,645

NB: The dimensions of the assemblies were as follows:
(1) δ = 1 cm; r_{int} = 10 cm; R_{ext} = 11 cm;
(2) δ = 4 cm; r_{int} = 8 cm; R_{ext} = 12 cm;
(3) δ = 8 cm; r_{int} = 4 cm; R_{ext} = 12 cm.

neutrons in assemblies of beryllium, lead, uranium and a number of other materials of similar properties is negligible.

The quantitative estimates obtained justify the conclusion as to the negligibly slight influence of an external aqueous moderator on the

neutron yield from materials with a high energy threshold for multiplication reactions. Thus, the use of the total absorption method for studying the multiplication of 14 MeV neutrons in assemblies is permissible.

It should be emphasized that there is a difference in principle between the total absorption detector considered here and the detectors used for calibrating radionuclide sources. Ordinarily, the effect is to reduce parasitic absorption of neutrons in the volume of the moderator by reducing the amount of construction material. In our case, the introduction of a strong absorber of thermal neutrons is indispensable for eliminating their flow into the assembly under study, since this flow can lead to further multiplication of neutrons (e.g. in experiments on uranium assemblies containing uranium-235). Calculations of the parameters of a total absorption detector in which there was no cadmium filter, showed that in the uranium spheres, the number of ^{235}U fission increased by a factor of 4 and had an appreciable effect on leakage of neutrons from the aseemblies.

4. THE BORON TANK METHOD

We shall dwell in greater detail on the measurement of neutron leakage by the boron tank method. The neutrons emitted by the assembly are absorbed in an aqueous solution of some boron compound, e.g. boric acid. If N_B, N_H and N_O are the number of boron, hydrogen and oxygen atoms in 1 cm^3, then the number of neutrons with starting energy E emitted by the assembly will be equal to the number absorbed in the tank:

$$Q(E) = \int_0^\infty \int_E N_B \cdot \sigma_B(E') \times \left(1 + \frac{N_H \cdot \sigma_H(E')}{N_B \cdot \sigma_B(E')} + \frac{N_O \cdot \sigma_O(E')}{N_B \cdot \sigma_B(E')}\right) 4\pi r^2 \Phi(r,E,E')\,dr\,dE' . \quad (2)$$

If at point r of the moderator there is a boron counter with a number, N_Z, of ^{10}B nuclei in 1 cm^3, and a volume, V_Z, then, by disregarding variation in flux over the volume of the counter, we can write the following expression for the count rate:

$$Z(r,E) = N_Z \cdot V_Z \int_{E'} \sigma_B(E') \cdot \Phi(r,E,E')\,dE' . \quad (3)$$

It is assumed here that the recording efficiency of the counter is unity. Using the notation for Z(r,E), we represent Q(E) in the form:

$$Q(E) = \frac{4\pi N_B}{N_Z \cdot V_Z} \int_0^{\infty} \int_{E'} \left(1 + \frac{N_H \cdot \sigma_H(E')}{N_B \cdot \sigma_B(E')} + \frac{N_O \sigma_O(E')}{N_B \cdot \sigma_B(E')}\right) r^2 \cdot Z(E'r)dr\, dE' . \quad (4)$$

Varying the value N_{H_2O}/N_B, i.e. changing the concentration of boric acid in the aqueous solution, we can reduce the capture of thermal neutrons by hydrogen (the absorption of fast neutrons by hydrogen is slight). In application to radionuclide sources, expression (4) can be simplified because the absorption by oxygen of neutrons up to 6-7 MeV energies is negligible. For high concentrations of boron, when $N_B\sigma_B \gg N_H\sigma_H$, expression (4) takes the form:

$$Q(E) = \frac{4\pi N_B}{N_Z \cdot V_Z} \int_0^{\infty} r^2 Z(r,E)dr . \quad (5)$$

The total source strength will be:

$$Q_0 = \int_E Q(E)dE . \quad (6)$$

5. NEUTRON RECORDING EFFICIENCY

We shall designate the neutron recording efficiency in the volume of the TAD as the value:

$$\epsilon(E) = \frac{\frac{4\pi N_B}{N_Z \cdot V_Z} \cdot \int_{R_1}^{R_2} r^2 \cdot Z(r,E)dE}{Q(E)} . \quad (7)$$

where R_1 and R_2 are the internal and external radii of the moderator. The efficiency $\epsilon(E)$ is always less than unity, for the following reasons:

(1) Some of the fast neutrons can be captured by oxygen in the $^{16}O(n,p)^{16}N$ and the $^{16}O(n,\alpha)^{13}C$ reactions with thresholds of 10 MeV and 3 MeV, respectively;

(2) With increasing energy, there is an increasing probability of neutron leakage from the volume of a TAD of finite dimensions;

(3) Some of the thermal neutrons are captured by hydrogen and by structural components of the TAD (including the cadmium filter).

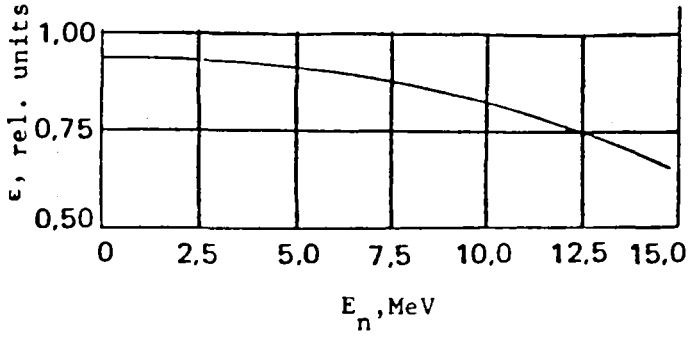

Fig. 2. Energy dependence of neutron recording efficiency of a total absorption detector.

For the arrangement shown in Fig. 1, a calculation was made of the energy dependence of the neutron recording efficiency of the TAD. The dimensions of the total absorption detector are the same as those in the calculation referred to above. The neutron source located at the centre of the TAD is a point-type, isotopic and monoenergetic source. The neutron energy varied in the range of 0.01-14 MeV. The dependence of $\varepsilon(E)$, calculated by the BLANK program, is shown in Fig. 2. The figure shows that the recording efficiency is constant right up to 5-6 MeV energy, amounting to 88% (approximately 6% of the neutrons are captured by hydrogen, 4% by cadmium and the remainder leak out of the TAD). Above 6 MeV, leakage of neutrons from the volume of the TAD and the capture of fast neutrons by the oxygen of the water begin to have a definite effect.

6. CHARACTERISTICS OF NEUTRON LEAKAGE MEASUREMENT BY THE BORON TANK METHOD

Let us consider the energy distribution of the neutrons leaking from assemblies with a central source of 14 MeV neutrons. Figure 3, by way of an example, shows the spectrum of leakage neutrons from a uranium sphere with a wall thickness δ = 8 cm, calculated by the BLANK program. It will be seen that this distribution can be represented in the form of the sum of two components: 14 MeV neutrons from a source and secondary neutrons from an inelastic interaction, including (n',n), (n,2n), (n,3n) and (n,f) neutrons. Because of the insignificant energy loss, the neutrons from elastic scattering can be identified with the neutrons from the source. The relationship between the scattered and unscattered parts of the spectrum will be determined solely by the thickness of the multiplier. In the Appendix, we present the calculated spectra of neutrons leaking from lead and uranium assemblies of various

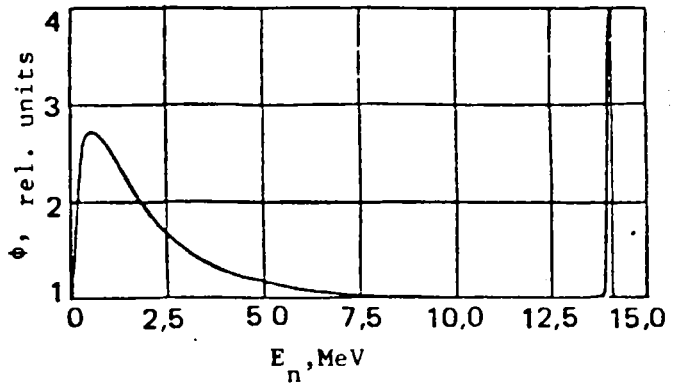

Fig. 3. Spectrum of neutron leaking from a uranium assembly with 8 cm wall thickness.

thicknesses. They show that for all sizes of assemblies made of heavy elements, the representation of the leakage spectrum in the form of the sum of two components is fully justified; the boundary of the spectrum of non-elastically scattered neutrons can be assumed to be 6 MeV.

The leakage spectrum characteristics referred to permits each component to be assigned its own recording efficiency.

Let us write the response of the total absorption detector to the value of M in the following manner:

$$m = \varepsilon_{14} \bullet + \langle\varepsilon\rangle \bullet (1 - T) \bullet \mu \qquad (8)$$

where ε_{14} and $\langle\varepsilon\rangle$ are the recording efficiencies for 14 MeV neutrons and inelastic interaction neutrons. If the multiplication of secondary neutrons in the assembly is not disregarded, the value m can be written thus:

$$m = \varepsilon_{14} \bullet T = + \langle\varepsilon\rangle \bullet N_{sec} \qquad (9)$$

where N_{sec} is the number of secondary neutrons escaping from the assembly, normalized in terms of one 14 MeV neutron from the source.

The efficiency ε_{14} can be determined in a joint measurement of integral $\int_{R_2}^{R_1} 4\pi r^2 Z(r,E)dr$ in the volume of the TAD and the strength of the 14 MeV neutron source by the method of accompanying α-particles. To obtain $\langle\varepsilon\rangle$ it is sufficient to use any radionuclide neutron source of known strength whose spectrum comes within the 0-6 MeV energy range. For instance, as shown by the calculation, the counting efficiency for neutrons from a californium-252 source amounts to approximately 87%,

which coincides with the value of ε in the said range. Photoneutron sources, neutrons from the (d,d) reaction with 2.5 MeV energy and a number of other sources are suitable for purposes of calibration.

By itself, the experimentally measured value of m does not yet afford a means of determining M, the leakage of neutrons from the assembly, since in the general case $\varepsilon_{14} \neq \langle\varepsilon\rangle$. For this reason it is necessary to determine the individual terms entering into the expression for M: transmission, T, and the number of secondary neutrons 1 - T or N_{sec}. The transmission, T, can be found from a supplementary measurement in which the function of neutron detectors can be filled, for example, by threshold detectors. If the value of T is known, we can determine N_{sec} by means of the measured value:

$$N_{sec} = \frac{m - \varepsilon_{14} \cdot T}{\langle\varepsilon\rangle} \tag{10}$$

After this, we find M, which is equal to:

$$M = T + N_{sec} \tag{11}$$

The predicted error in the measurement of M is 5-6% and includes:

- the error in the measurement of ε (according to data in the literature, the measurement error for absolute neutron yield $\simeq$ 3%,
- the error in the measurement of $\langle\varepsilon\rangle$ (the strength of the californium source is known to within 2-3%),
- the error in the measurement of T (in a relative measurement an error of 1-2% can be obtained),
- the error in the measurement of (m) with normalization to one 14 MeV neutron amounts to 3%.

In conclusion, emphasis should be placed on the advantages of measuring M by the total absorption method. First of all, in contrast to the spectrometric methods, M is measured in the entire energy range of the neutrons emitted and the spectrum does not have to be extrapolated below the neutron recording threshold of the spectrometer. Second, owing to the high neutron recording efficiency in the TAD, the measurements can

be performed on low-intensity (10^8-10^9 n/s) sources, there being no special requirements whatever with respect to neutron generators (e.g. there is no need for a pulsed operating regime of a neutron generator with short pulses, which is a prerequisite for time-of-flight spectrometers). Lastly, the measurement technique in the total absorption method is fairly simple and can be carried out easily.

7. CONCLUSIONS

(1) The paper demonstrates the possibility of using the total absorption method to measure the multiplication of 14 MeV neutrons in assemblies simulating the breeding zone of the blanket of a thermonuclear reactor. It is shown that the influence of scattered neutrons in the external moderator on multiplication in the assembly is negligible.

(2) On the basis of a calculation of neutron recording efficiency in a total absorption detector, it is shown that its efficiency is constant in the 0.01-6 MeV energy range.

(3) The proposed measurement procedure which, by means of a total absorption detector, can be used for determining absolute leakage of neutrons from multiplication assemblies with a source of 14 MeV neutrons, is based on breaking down the spectrum of neutron leakage from the assembly into two component, corresponding to the neutrons from the source and the elastically scattered neutrons and to the neutrons from inelastic interactions. The predicted error of these measurements is 5-6%.

APPENDIX

Leakage of neutrons from assemblies (normalized in terms of 1 neutron from a source).

Group No.	Energy range, MeV	Assembly material				
		Lead		Uranium-238		
		δ = 3 см	δ = 6 см	δ = 1 см	δ = 4 см	δ = 8 см
1	$1{,}00.10^{-8} - 6{,}70.10^{-7}$	0	0	0	0	0
2	$6{,}70.10^{-7} - 1{,}01.10^{-6}$	0	0	0	0	0
3	$1{,}01.10^{-6} - 2{,}74.10^{-6}$	0	0	0	0	0
4	$2{,}74.10^{-6} - 7{,}45.10^{-6}$	0	0	0	0	0
5	$7{,}45.10^{-6} - 2{,}03.10^{-5}$	0	0	0	0	0
6	$2{,}03.10^{-5} - 5{,}51.10^{-5}$	0	0	0	0	0
7	$5{,}51.10^{-5} - 1{,}50.10^{-4}$	0	0	0	0	0
8	$1{,}50.10^{-4} - 4{,}07.10^{-4}$	0	0	0	0	0
9	$4{,}07.10^{-4} - 1{,}82.10^{-3}$	$1{,}000.10^{-4}$	0	0	0	0
10	$1{,}82.10^{-3} - 1{,}00.10^{-2}$	$3{,}000.10^{-4}$	$6{,}000.10^{-4}$	0	$1{,}000.10^{-4}$	$2{,}307.10^{-4}$
11	$1{,}00.10^{-2} - 2{,}50.10^{-2}$	$3{,}000.10^{-4}$	$1{,}000.10^{-4}$	$3{,}085.10^{-4}$	$2{,}214.10^{-4}$	$1{,}492.10^{-3}$
12	$2{.}50.10^{-2} - 5{.}00.10^{-2}$	$3{,}100.10^{-4}$	$5{,}000.10^{-4}$	$4{,}000.10^{-4}$	$3{,}564.10^{-3}$	$1{,}015.10^{-2}$
13	$5{,}00.10^{-2} - 1{,}00.10^{-1}$	$1{,}800.10^{-3}$	$2{,}800.10^{-3}$	$5{,}585.10^{-3}$	$2{,}381.10^{-2}$	$5{,}492.10^{-2}$
14	$1{,}00.10^{-1} - 1{,}25.10^{-1}$	$1{,}300.10^{-3}$	$2{,}900.10^{-3}$	$4{,}244.10^{-4}$	$1{,}996.10^{-2}$	$4{,}765.10^{-2}$
15	$1{,}25.10^{-1} - 1{,}50.10^{-1}$	$1{,}700.10^{-3}$	$2{,}000.10^{-3}$	$5{,}361.10^{-3}$	$2{,}405.10^{-2}$	$5{,}399.10^{-2}$
16	$1{,}50.10^{-1} - 1{,}75.10^{-1}$	$1{,}320.10^{-2}$	$2{,}270.10^{-3}$	$5{,}495.10^{-3}$	$2{,}629.10^{-2}$	$5{,}314.10^{-2}$
17	$1{,}75.10^{-1} - 2{,}00.10^{-1}$	$1{,}200.10^{-3}$	$3{,}100.10^{-3}$	$6{,}529.10^{-3}$	$2{,}849.10^{-2}$	$5{,}698.10^{-2}$
18	$2{,}00.10^{-1} - 2{,}26.10^{-1}$	$2\,600.10^{-3}$	$6{,}800.10^{-3}$	$6{,}863.10^{-3}$	$3{,}058.10^{-2}$	$5{,}838.10^{-2}$
19	$2{,}25.10^{-1} - 2{,}50.10^{-1}$	$2{,}300.10^{-3}$	$5{,}500.10^{-3}$	$7{,}187.10^{-3}$	$3{,}283.10^{-2}$	$6{,}646.10^{-2}$
20	$2{,}50.10^{-1} - 2{,}75.10^{-1}$	$3{,}600.10^{-3}$	$4{,}300.10^{-3}$	$5{,}495.10^{-3}$	$2{,}687.10^{-2}$	$5{,}788.10^{-2}$
21	$2{,}75.10^{-1} - 3{,}00.10^{-1}$	$3{,}100.10^{-3}$	$6{,}500.10^{-3}$	$6{,}129.10^{-3}$	$3{,}059.10^{-2}$	$5{,}637.10^{-2}$
22	$3{,}00.10^{-1} - 3{,}50.10^{-1}$	$1{,}320.10^{-2}$	$2{,}53.10^{-3}$	$1{,}665.10^{-2}$	$6{,}504.10^{-2}$	$1{,}113.10^{-1}$
23	$3{,}50.10^{-1} - 4{,}00.10^{-1}$	$5{,}200.10^{-3}$	$9{,}300.10^{-3}$	$1{,}282.10^{-2}$	$5{,}75.10^{-2}$	$1{,}034.10^{-1}$
24	$4{,}00.10^{-1} - 4{,}50.10^{-1}$	$1{,}480.10^{-2}$	$2{,}510.10^{-2}$	$1{,}264.10^{-2}$	$5{,}691.10^{-2}$	$9{,}902.10^{-2}$
25	$4{,}50.10^{-1} - 5{,}00.10^{-1}$	$5{,}300.10^{-2}$	$9{,}500.10^{-3}$	$1{,}137.10^{-2}$	$5{,}384.10^{-2}$	$9{,}369.10^{-2}$
26	$5{,}00.10^{-1} - 5{,}50.10^{-1}$	$1{,}620.10^{-2}$	$2{,}610.10^{-2}$	$1{,}353.10^{-2}$	$5{,}114.10^{-2}$	$9{,}19.10^{-2}$
27	$5{.}50.10^{-1} - 6{,}00.10^{-1}$	$1{,}070.10^{-2}$	$2{,}020.10^{-2}$	$1{,}518.10^{-2}$	$5{,}204.10^{-2}$	$8{,}880.10^{-2}$
28	$6{,}00.10^{-1} - 6{,}50.10^{-1}$	$1{,}140.10^{-2}$	$1{,}640.10^{-2}$	$9{,}516.10^{-3}$	$4{,}771.10^{-2}$	$8{,}055.10^{-2}$
29	$6{,}50.10^{-1} - 7{,}00.10^{-1}$	$1{,}330.10^{-2}$	$2{,}890.10^{-2}$	$1{,}070.10^{-2}$	$4{,}390.10^{-2}$	$7{,}440.10^{-2}$
30	$7{,}00.10^{-1} - 7{,}50.10^{-1}$	$6{,}700.10^{-3}$	$1{,}390.10^{-2}$	$1{,}125.10^{-2}$	$4{,}314.10^{-2}$	$7{,}553.10^{-2}$
31	$7{,}50.10^{-1} - 8{,}00.10^{-1}$	$1{,}310.10^{-2}$	$2{,}230.10^{-2}$	$1{,}287.10^{-2}$	$4{,}521.10^{-2}$	$6{,}448.10^{-2}$
32	$8{,}00.10^{-1} - 9{,}00.10^{-1}$	$1{,}930.10^{-2}$	$3{,}350.10^{-2}$	$1{,}886.10^{-2}$	$6{,}831.10^{-2}$	$1{,}097.10^{-1}$
33	$9{,}00.10^{-1} - 1{,}00$	$1{,}870.10^{-2}$	$3{,}320.10^{-2}$	$1{,}865.10^{-2}$	$6{,}419.10^{-2}$	$9{,}330.10^{-2}$
34	1,00 – 1,20	$3{,}830.10^{-2}$	$6{,}430.10^{-2}$	$3{,}364.10^{-2}$	$1{,}019.10^{-1}$	$1{,}545.10^{-1}$
35	1,20 – 1,40	$2{,}790.10^{-2}$	$5{,}520.10^{-2}$	$2{,}603.10^{-2}$	$8{,}191.10^{-2}$	$1{,}108.10^{-1}$
36	1,40 – 1,60	$3{,}300.10^{-2}$	$5{,}680.10^{-2}$	$2{,}203.10^{-2}$	$5{,}444.10^{-2}$	$7{,}023.10^{-2}$
37	1,60 – 1,80	$1{,}820.10^{-2}$	$3{,}560.10^{-2}$	$1{,}831.10^{-2}$	$4{,}723.10^{-2}$	$5{,}214.10^{-2}$
38	1,80 – 2,00	$1{,}960.10^{-2}$	$3{,}920.10^{-2}$	$1{,}759.10^{-2}$	$4{,}098.10^{-2}$	$4{,}828.10^{-2}$
39	2,00 – 2,25	$2{,}190.10^{-2}$	$3{,}920.10^{-2}$	$1{,}759.10^{-2}$	$4{,}098.10^{-2}$	$4{,}828.10^{-2}$
40	2,25 – 2,50	$1{,}750.10^{-2}$	$3{,}090.10^{-2}$	$1{,}428.10^{-2}$	$3{,}386.10^{-2}$	$4{,}045.10^{-2}$
41	2,50 – 2,75	$1{,}680.10^{-2}$	$2{,}870.10^{-2}$	$1{,}394.10^{-2}$	$2{,}939.10^{-2}$	$3{,}526.10^{-2}$
42	2,75 – 3,00	$1{,}000.10^{-2}$	$1{,}490.10^{-2}$	$1{,}047.10^{-2}$	$2{,}702.10^{-2}$	$2{,}533.10^{-2}$
43	3,00 – 3,25	$1{,}020.10^{-2}$	$1{,}640.10^{-2}$	$1{,}009.10^{-2}$	$2{,}315.10^{-2}$	$2{,}285.10^{-2}$

Group No.	Energy range, MeV	Assembly material				
		Lead		Uranium-238		
		δ = 3 см	δ = 6 см	δ = 1 см	δ = 4 см	δ = 8 см
44	3,25 – 3 50	7,000.10^{-1}	1,620.10^{-2}	7,916.10^{-3}	2,142.10^{-2}	2,413 10^{-2}
45	3,50 – 3,75	2,500.10^{-3}	2,800.10^{-3}	8,148.10^{-3}	1,921.10^{-2}	2,031 10^{-2}
46	3,75 – 4,00	7,400.10^{-3}	9,600.10^{-3}	5,589.10^{-3}	1,434.10^{-2}	1,646.10^{-2}
47	4,00 – 4,50	8,600.10^{-3}	1,300.10^{-2}	1,344.10^{-2}	2,994.10^{-2}	3,168.10^{-2}
48	4,50 - 5,00	1,300.10^{-3}	9,000.10^{-4}	7,582.10^{-3}	1,686.10^{-2}	1,959.10^{-2}
49	5,00 – 5,50	6,500.10^{-3}	1,3401.0^{-2}	5,663.10^{-3}	1.331.10^{-2}	1,874.10^{-2}
50	5,50 – 6,00	1,100.10^{-3}	1,800.10^{-3}	5,195.10^{-3}	1,094.10^{-2}	9,738.10^{-3}
51	6,00 – 6,50	1,100.10^{-3}	1,900.10^{-3}	4,361.10^{-3}	9,464.10^{-3}	9,607.10^{-3}
52	6,50 7,00	2,400.10^{-3}	2,200.10^{-3}	3,068.10^{-3}	5,950.10^{-3}	7,454.10^{-3}
53	7 00 – 7,50	9,000.10^{-4}	1,500.10^{-3}	9,255.10^{-4}	2,586.10^{-3}	3,523.10^{-3}
54	7,50 – 8,00	1,000.10^{-3}	1,400.10^{-3}	1,534.10^{-3}	1,764.10^{-3}	3,192.10^{-3}
55	8,00 8,50	1,000.10^{-3}	1,400.10^{-3}	1,051.10^{-3}	1,221.10^{-3}	2,184.10^{-3}
56	8,50 – 9,00	1,400.10^{-3}	1,700.10^{-3}	5,170.10^{-4}	1,500.10^{-3}	1,792.10^{-3}
57	9,00 – 1,00.10^{1}	2,400.10^{-3}	5,100.10^{-3}	1,208.10^{-3}	2,643.10^{-3}	2,923.10^{-3}
58	1 00.10^{1} – 1,10.10^{1}	3,400.10^{-3}	6,000.10^{-3}	1,625.10^{-3}	1,521.10^{-3}	1,661.10^{-3}
59	1,10.10^{1} – 1,20.10^{1}	2,600.10^{-3}	3,600.10^{-3}	1,208.10^{-3}	6,000.10^{-4}	1,100.10^{-3}
60	1,20.10^{1} – 1,25.10^{1}	1,300.10^{-3}	2,200.10^{-3}	5,000.10^{-4}	4,000.10^{-4}	9,000.10^{-4}
61	1,25.10^{1} – 1,30.10^{1}	2,100.10^{-3}	2,100.10^{-3}	9,000.10^{-4}	1,200.10^{-3}	1,100.10^{-3}
62	1,30.10^{1} – 1,37.10^{1}	3,500.10^{-3}	4,100.10^{-3}	2,400.10^{-3}	1,700.10^{-3}	1,800.10^{-3}
63	1,37.10^{1} – 1,40.10^{1}	1,130.10^{-2}	1,610.10^{-2}	9,200.10^{-3}	1,300.10^{-2}	1,350.10^{-2}
64	1,40.10^{1} – 1,41.10^{1}	1,580.10^{-1}	1,980.10^{-1}	1,169.10^{-1}	1,936.10^{-1}	1,628.10^{-1}
65	1,41.10^{1} – 1,42.10^{1}	5,730.10^{-1}	3,449.10^{-1}	7,102.10^{-1}	3,311.10^{-1}	1,231.10^{-1}
	Total leakage:	1,192	1,347	1,308	2,064	2,681

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