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TRANSLATIONS FROM RUSSIAN PUBLICATIONS OF PROCEEDINGS OF THE INTERNATIONAL CONFERENCE ON NEUTRON PHYSICS, KIEV, 14 - 18 SEPTEMBER 1987

Institute of Physics and Power Engineering
I.V. Kurchatov Institute of Atomic Energy

Translated by the IAEA

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THERMONUCLEAR REACTORS AND NUCLEAR DATA REQUIREMENTS

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A positive plasma energy balance is a prerequisite for a reactor based on any fusion reaction, i.e.

$$W_{TN} - W_{\ell} - W_{BS} - \Phi W_{MB} > 0,$$

where $W_{TN} = f_1 f_2 E \langle\sigma v\rangle$ is the power density (normalized to the square of the plasma ion density) as a result of the deceleration of charged reaction products in the plasma (we shall consider only the two-component reactions of nuclear fusion), f_i is the fraction of the i-th component of the reacting mixture, E is the energy of the charged particles, $\langle\sigma v\rangle$ is the ion energy averaged product (averaged over the plasma ion energy spectrum) of the fusion reaction cross-section σ and the relative velocity of the reacting nuclei v, W_{ℓ} is the specific energy losses of the plasma resulting from the escape of reacting ions from the reactor area (for simplicity, we shall assume below that plasma retention is ideal, i.e. $W_{\ell} = 0$, which gives an optimal evaluation of the plasma energy balance),

$$W_{T\text{И}} = 1.1\cdot 10^{-35}\left(\sum_i f_i Z_i\right)\left(\sum_i f_i Z_i^2\right) T^{1/2} \quad (\text{Вт}\cdot\text{м}^3) -$$

represents the specific losses due to bremsstrahlung, where Z_i is the charge of the i-th plasma ion component and T is the ion temperature, normalized to the electron rest energy (0.511 MeV), and, finally,

$$W_{M.T} = 0.77\cdot 10^{-33}\left(\sum_i f_i Z_i\right)\left(1+\sum_i f_i Z_i\right)\beta^{-1} T^2 \quad (\text{Вт}\cdot\text{м}^3) -$$

represents the specific losses resulting from the magnetic bremsstrahlung (MB) (cyclotron radiation), which occurs in systems with magnetic plasma containment, $\beta = \frac{P}{P_m} = \frac{n(1+\sum f_i Z_i)KT}{H^2/8\pi}$ is the relationship of the plasma pressure to the magnetic pressure, and Φ is the MB attenuation factor resulting from "blocking" of the latter in the plasma volume (we shall assume that $\Phi \equiv 0$ for reactors not employing magnetic containment. Strict calculation of the

Φ factor requires a solution to the magnetic bremsstrahlung channelling equations and is extremely time-consuming, but an approximate theory can be used for evaluations with an accuracy of up to 1.5-2 (depending on the plasma temperature, density and dimensions), as suggested by Trubnikov [1] and perfected by Kogan and Lisitsa [2]. According to this theory,

$\Phi = 0.3\ F(\lambda)/\lambda$,

where $\lambda = 0.3 \frac{\bar{T}^{-5/2} \omega_0^2 a}{(1+\bar{T}) \omega_H c}$, λ is the likelihood parameter,

$f(\lambda)$ is a universal function, determined by numerical integration, a is the characteristic linear dimension of the plasma volume, ω_0 is the plasma frequency, ω_H is the electron Larmor frequency. For the most common values of plasma parameters in the various types of thermonuclear reactor ($n \sim 10^{20}\ m^{-3}$; $T \sim 10$-100 keV; $a \simeq 1$m; $H \sim 5$-10 T) the Φ factor values lie approximately within the limits 0.01 to 0.1.

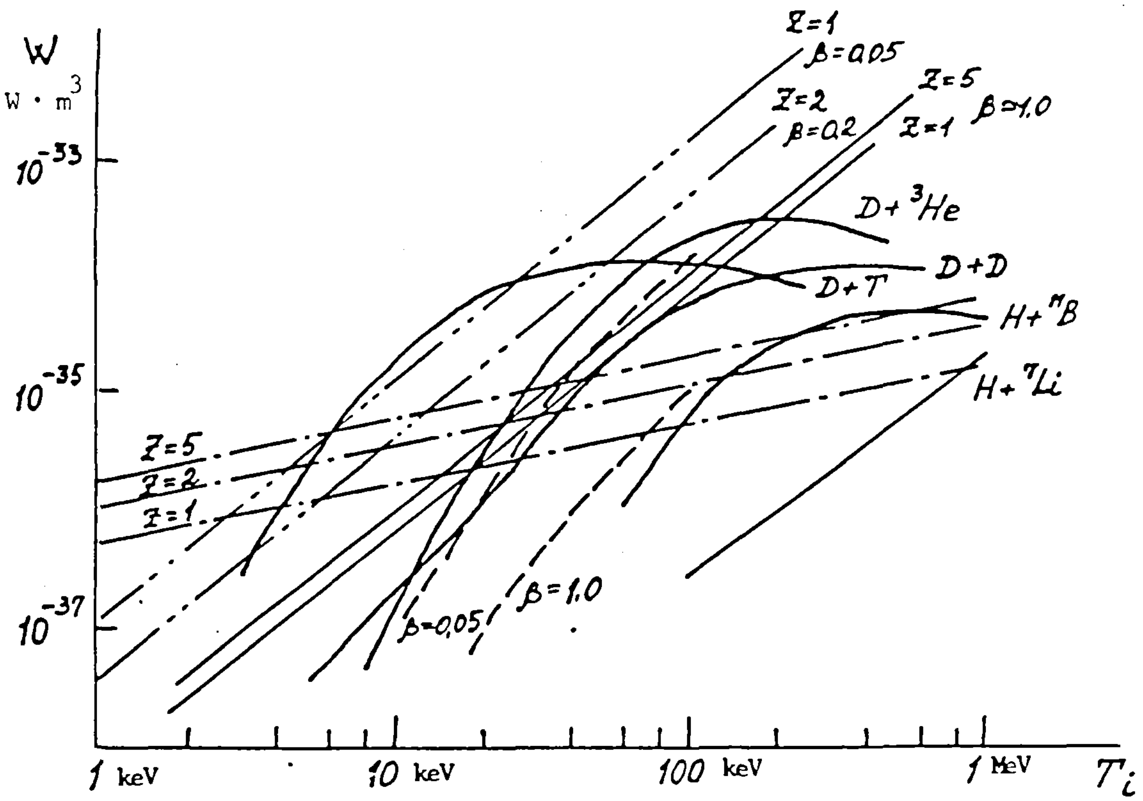
The minimum ratio of bremsstrahlung losses to the power intensity in the plasma is achieved for a particular composition of the reacting mixture. Thus, for example, at $Z_1 = Z_2$ we have $f_1 = f_2$, at $Z_1 = 1$ and $Z_2 = 2$, $f_1 = 0.74$, $f_2 = 0.26$, whereas at $Z_1 = 1$ and $Z_2 = 5$, $f_1 = 0.91$, $f_2 = 0.09$. The relative gain as a result of changing from equimolar ($f_1 = f_2 = 0.5$) to the optimal mixture composition, is approximately 10% at $Z_1 = 1$ and $Z_2 = 2$ and exceeds 300% at $Z_1 = 1$ and $Z_2 = 5$.

The graph shows the curves for power intensity, bremsstrahlung losses and magnetic bremsstrahlung losses (taking into account the blocking of radiation in the plasma volume) for the following reactions which are those of greatest practical interest.

$$D + T \rightarrow {}^4He + n,\ D + D \rightarrow T + p,\ D + D \rightarrow {}^3He + n$$

(taking into account the subsequent "burnup" of the tritium and ^3He formed),

$$p - {}^7Li \rightarrow 2\,{}^4He, \quad p + {}^{11}B - 3\,{}^4He.$$



Plasma energy balance components:

— W_{TN} —·—W_{BS} —··— W_{MB} — — — W_{MB} with allowance for blocking

It follows from the data used that in reactors with magnetic plasma containment at relatively high values of β(0.05-0.1), which is typical of Tokamak and Stellarator reactors, reliable sustainment can be expected only in the D + T → ^{4}He + n reactions. In systems with more effective plasma containment (β ~ 1), we can expect the D + D and D + ^{3}He reactions to be sustained as well. It can be expected that all the reactions shown on the diagram, with the exception of p + ^{7}Li, will be ignited in reactors with inertial plasma containment, where there is no magnetic bremsstrahlung. The "thermonuclear ideal" has to do with significantly reducing the radiation danger by using nuclear fusion reactions without neutrons and tritium[*].

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[*] Reactions "without neutrons" accompany reactions with neutrons and radioactive products, for example D + D - ^{3}He + n, D + D → T + p in the mixture D^{3}He, p + ^{11}B → ^{11}C + n and the secondary reaction ^{4}He + ^{11}B → ^{14}N + n in the H^{11}B mixture.

However, even the reaction D + ^3He, which has the highest cross-section, requires not only a higher plasma temperature, pressure and "containment parameter" ($p\tau_E$) than the reaction D + T, but also suppression of the accompanying reaction D + D, which leads to the formation of tritium and neutrons. Research is also required into methods of obtaining ^3He, which is present in negligible quantities on the earth. Non-neutron reactions are therefore still problematic.

Use of the T-reaction to produce energy is currently being considered. Although the total radioactivity of such a reactor cannot be significantly decreased from the level of 10^{10}/GWt, which is typical of fission reactors, the absence of a physical rapid excursion mechanism, and also the possibility of reducing the quantity of volatile and long-lived activation products by using special low-activity construction materials, warrants the assumption of a substantial reduction in radiation danger for this type of reactor too.

For these reasons, it is very important in developing promising fusion reactors to measure the activation cross-sections (with neutron energies of $E_N \leq 14$ MeV) of the construction material components (Fe, Ni, Cr, Mo, Al, Nb, V, Ti and ceramic materials based on C, Si and O), taking into account the impurities which they contain. The activation value of impurities in the construction materials used is significant. For austenite, they were as follows:

$^{52}Cr(n,2n)^{51}Cr$, $^{54}Fe(n,p)^{54}Mn$,

$^{54}Fe(n,\alpha)^{51}Cr$, $^{56}Fe(n,2n)^{55}Fe$,

$^{55}Mn(n,2n)^{54}Mn$, $^{59}Co(n,\gamma)^{60}Co$,

$^{58}Ni(n,p)^{58}Co$, $^{60}Ni(n,p)^{60}Co$.

For promising vanadium alloys:

$^{48}Ti(n,\alpha)^{45}Ca$, $^{50}Ti(n,\alpha)^{47}Ca$,

$^{45}Sc(n,p)^{45}Ca$.

For coil materials:

$^{63}Cu(n,p)^{63}Ni$, $^{63}Cu(n,\alpha)^{60}Co$,

$^{65}Cu(n,2n)^{64}Cu$.

For many isotopes data are available in the 14 MeV region, but are insufficient in other parts of the spectrum.

Thermonuclear reactors will probably go through several stages of development, as did fission reactors as well. A natural first stage would be hybrid fusion-fission reactors, with a nuclear fusion reaction chamber surrounded by a blanket of depleted uranium or thorium. The $D + T - {}^{4}He + n$ reaction is of particular interest as a source of neutrons with 14.1 MeV energy. For these neutrons, the fission cross-section of the isotopes ^{238}U and ^{232}Th is several times higher than for fission spectrum neutrons, so that considerable additional power can be obtained in the uranium blanket and the neutron yield per nuclear fusion event can be substantially increased. Most of the neutrons formed can be used to produce ^{239}Pu and ^{233}U isotopes suitable for use as fuel in fast-neutron fission reactors. The energy obtained from burning, in fission reactors, fuel produced by a hybrid reactor is several times higher than the energy generated in the hybrid reactor blanket (approximately 4-6 times higher for WWER reactors). Although a hybrid reactor cannot demonstrate the full potential of nuclear fusion, it is easier to set up than a thermonuclear reactor on account of its better energy balance.

Taking into account the practicable requirements for the blanket of a hybrid reactor based on the DT-reaction, a single fusion event can be expected to produce 1.2-1.6 plutonium nuclei in it. Approximately 0.6 ^{238}U fission events will take place in the depleted uranium blanket, and one ^{6}Li neutron capture event, with the breeding of a tritium nucleus. Plutonium regeneration is only slightly dependent on the type of fuel and the content of the blanket construction materials, and represents 0.8-1.0 T/GW(t) per year at the beginning of the cycle and 0.6-0.8 T/GWt(t) per year with a plutonium accumulation of up to 10 kg/T.

It would be interesting to set up a hybrid reactor with inertial plasma containment based on reactions in purely deuterium plasma. In this case the number of 14.1 MeV neutrons per single DD-fusion reaction would be twice as low as for the D + T reaction, but there would be less need to breed tritium, which gives one additional neutron per fusion event in the case of fissile isotope regeneration. As a result, fissile isotope regeneration per unit of blanket thermal capacity is increased (approximately up to 1.4 T/(GW•year) for a uranium blanket and up to 3 T/(GW•year) for a thorium blanket at the beginning of the cycle [3]).

The radiation safety of a hybrid reactor can be increased by a blanket with "suppressed fission", in which Th is mixed with Li, and neutron multiplication is achieved by the (n,2n) reaction in Be or Pb.

Neutron data for calculating neutron transport are very important in thermonuclear reactor design, as are also calculations of the functionals which determine the fissile isotope and tritium production, energy release, gas release and radiation damage to the blanket materials.

The 9-16 MeV neutron energy range is the most important one from the point of view of data refinement. The accuracy requirements for calculating the most important functionals (e.g. the rates of plutonium and tritium regeneration) are about 1%, for which data are needed on the total cross-sections with an accuracy of 3% and on the energy angular distributions of secondary neutrons with an accuracy of 10-20%. As the requirements indicated are not entirely met by any existing evaluated data library, first priority should be given to the measurement and evaluation of these data, a task requiring broad international co-operation.

Accuracy in calculating the recovery rate of fissile isotopes and tritium in the blanket with suppressed fission is largely dependent on the data for Be and Pb. The Be(n,2n) reaction cross-section must be known with an accuracy of ~ 3% in order to calculate the regeneration rate of ^{239}U and T with an accuracy of ~ 1%.

The most immediate and essential practical step towards achieving an industrial-scale fusion reactor is the development and construction of an experimental energy-producing fusion reactor. In addition to the development of the international INTOR design, a number of countries are also working on national designs for experimental tokamak reactors. Work of this kind is also being done in the USSR (on the EFR/experimental fusion reactor/project) [1].

The development priorities for the EFR design are as follows:

(1) To achieve the combustion of an intensive thermonuclear reaction;

(2) To achieve prolonged plasma combustion (of the order of several hundred seconds);

(3) Prolonged operation at the rated power level with a neutronic first-wall loading (neutron importance) of up to 5 MW/m^2;

(4) Breeding in the reactor of tritium burned by it;

(5) Attainment of a specific neutronic first-wall loading of not less than 0.8-1 MW/m^2 with neutron fluxes of approximately 10^{15} cm^{-2} s^{-1}.

BASIC CHARACTERISTICS OF THE EXPERIMENTAL FUSION REACTOR

Geometry	
Major plasma radius	6.3 m
Plasma radius (a)	1.5 m
Mean plasma elongation (K)	1.5
Aspect ratio (A)	4.2
Plasma volume	420 m^3
Plasma	
Mean temperature ($\langle T_i \rangle$)	8 keV
Mean electron density ($\langle n_e \rangle$)	1.7×10^{20} m^{-3}
Power confinement time (τ_e)	1.9 s
Field on the plasma axis (B_T)	5.8 T
Plasma current (I_p)	8.0 MA

Operating regime	
Thermonuclear power (P_{th})	500 MWt
Time of reaction combustion (t_{burn})	600 s
Dwell time (t_{dwell})	880 s
Number of combustion cycles	3×10^5
Load coefficient	up to 0.5
First wall	
Maximum heat flux	25 W/cm^2
Construction material	Austenitic steel
Thickness	6 mm
Coolant	Water
Maximum wall temperature	250°C
Mean neutron load	0.9 MW/m^2
Neutron fluence ($E \geq 0.1$ MeV)	5×10^{22} n/cm^2
Blanket	
Number of sectors	12
Neutron multiplier	Pb, $Pb_{83}Li_{17}$
Breeding material	$Pb_{83}Li_{17}$, ceramics
Construction material	Austenitic steel
Coolant	H_2O
Thickness (including first wall)	0.9 m
Tritium-breeding coefficient	1.05
Containment	
Containment materials	Boronated steel
	H_2O, Pb
Thickness of inner containment	0.9 m
Thickness of outer containment	1.5 m
Coolant	H_2O
Maximum temperature	100°C
Dose rate at external surface 24 hours after shutdown	2.5 mbar/hour

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NEUTRON LEAKAGE SPECTRA FROM Be, Pb and U SPHERES AT 14 MeV ENERGY

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Experiments based on studying the spectra of neutron leakage from homogeneous spherical assemblies with a neutron source at the centre are included among the so-called "benchmark" experiments [1]. The main purpose of such research is to test nuclear physics constants and calculation methods. If the experimental geometry is kept relatively simple (almost spherically symmetrical), corrections connected with the modelling of specific features of the experimental installation can be taken into account, or disregarded, in the analysis.

This paper contains measurement and calculation results for neutron leakage spectra for Be, Pb and U from spherical assemblies with a 14 MeV neutron source at the centre. Experiments at this energy are of particular interest in connection with thermonuclear reactor problems. There has been intensive research in this field in the United States [2, 3], Japan [4], and preliminary measurements have been performed in the USSR [5] and the German Democratic Republic [6].

The experiment and the measurement results

Neutron leakage spectra were measured using the time-of-flight method in a spectrometer using an IPPE KG-0.3 accelerator [7]. Neutrons of approximately 14 MeV energy were obtained by bombarding a TiT-target with 250 keV deuterons. The target was placed at the centre of the assemblies in question, which were hollow spheres with a Ø 5 cm cylindrical aperture under the ion guide (see Table 1).

The scattered neutrons were recorded by a scintillation detector in the shielding on time base of ~ 3.3 m. An iron cone was placed between the

Table 1

Element	R Int. cm	R Ext. cm	Wall thickness cm	λ_{mfp}[*]	Mass, kg
Be	6	11	5	0.9	8.59
Pb	3	12	9	1.6	80.85
^{238}U (0.4% ^{235}U)	4	12	8	2.2	130.34

[*] λ_{mfp} wall thickness, expressed in units of mean free path of source neutrons.

sphere and the detector to measure background. The total time resolution of the spectrometer was 3 ns.

Measurements were made at angles of 8°, 30°, and 60°. The neutron flux from the target was monitored by an all-band detector. A DKP_s-25 semiconductor silicon detector was used to calibrate the all-band detector and determine the absolute number of neutrons escaping from the target; it also recorded the alpha-particle spectrum from the T(d,α)n reaction . A sample source of ^{238}Pu alpha-particles was in turn used for absolute calibration of the DKP_s-25 detector. The source was placed in the ion guide on the site of the tritium target.

The effectiveness of the neutron detector in the energy interval from the threshold 0.2 MeV to 8 MeV was measured in relation to ^{252}Cf spontaneous fission neutron spectrum [8], by means of a fast ionization chamber [9]. At a neutron energy of 14 MeV, the effectiveness of the detector was measured in relation to the neutron yield from the target, which was calibrated using the DKP_s-25 counter.

After subtraction of the background, conversion to the energy scale, corrections for detector efficiency, and absolute normalization to 1 neutron leaving the target, neutron leakage spectra were obtained in the 0.4-14 MeV energy range (see Table 2). The error of the experimental data is 6-7%.

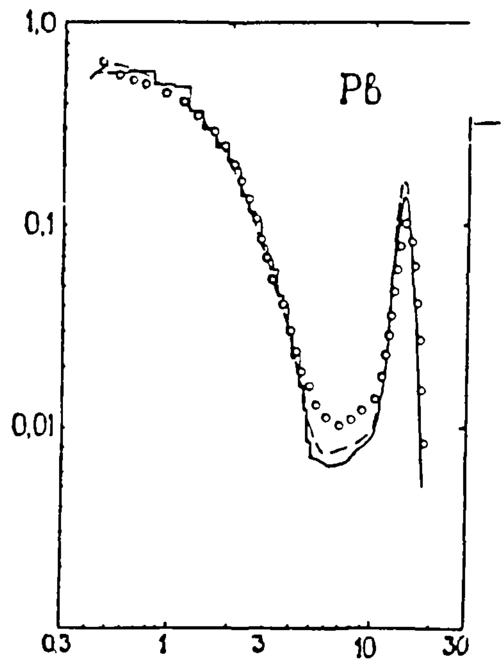
Table 2

Nucleus	E, MeV	Experiment, 10^{-2}n/av. 1n source 8°	30°	60°	A	f_1	f_2	Experiment n/1n source	C/E BRAND NEDAM	C/E BLANK ENDF
Ядро	E, МэВ	Эксперимент, 10^{-2} n/ср·1n ист 8°	30°	60°	A	f_1	f_2	Эксперим. n/1n ист	C/E BRAND NEDAM	BLANK ENDF
	0,4- I	0,94	0,90	0,88	0,02	I,08	I,05	0,I20	0,96	0,80
	I- 3	I,2I	I,I9	I,I3	0,02	I,03	I,03	0,I57	I,07	0,80
Be	3-I0	I,54	I,54	I,42	0,05	I,0I	0,97	0,I78	0,95	I,08
	I0 -I5	5,89	5,66	5,57	0,05	I,0I	0,99	0,690	0,85	0,99
	0,4-I5	9,58	9,29	9,00				I,I45	I,00	0,92
	0,4- I	2,54	2,53	2,59	0,0I	I,06	I,07	0,36I	0,86	0,90
	I-3	3,64	3,65	3,74	0,02	I,04	I,04	0,492	0,90	0,8[illegible]
Pb	3-5	0,53	0,52	0,56	0,03	I,02	I,0I	0,068	0,94	0,67
	5-I0	0,48	0,46	0,49	0,04	I,0I	I,00	0,059	0,6I	0,68
	I0-I5	3,64	3,59	3,55	0,05	0,99	0,98	0,44I	I,07	I,04
	0,4-I5	I0,8	I0,8	I0,9				I,42I	0,92	0,91
	0,4- I	5,73	5,II	5,53	0,02	I,02	I,09	0,774	I,I0	I,02
	I- 3	2,93	2,62	2,78	0,02	I,02	I,05	0,368	I,57	I,1[illegible]
U	3- 6	0,84	0,74	0,79	0,03	I,00	I,02	0,099	I,7I	I,07
	6-I0	0,37	0,34	0,35	0,04	0,99	0,98	0,043	0,76	0,6[illegible]
	I0-I5	2,67	2,65	2,44	0,04	0,98	0,98	0,303	I,04	I,0[illegible]
	0,4-I5	I2,5	II,5	II,9				I,58I	I,2I	I,04

Analysis of experimental results, comparison with calculations

Neutron leakage spectra for spherical assemblies were calculated on the basis of the BRAND [10] and BLANK [11] programs. BLANK uses a one-dimensional approximation, which holds good if there is spherical symmetry. The BRAND program, on the other hand, can be used to model a more realistic three-dimensional experimental geometry, the energy and angular distribution of the neturon flux from the target and the time-of-flight method for neutron recording. It is therefore initially advantageous to use BRAND to evaluate the influence of the factors mentioned, as a means, by introducing appropriate corrections, of arriving at a spherically symmetrical problem.

One of the reasons for the angular anisotropy of the problem is the dependence of the source neutron yield and energy on the angle of emergence Θ from the target [12]. Assuming that the angular distribution



o Experimental

___ BRAND calculation

- - - BLANK calculation

Figure

Neutron leakage spectra from Pb spheres

can be approximated by the expression $N(E,\theta) = N(E)(I + A \cos \theta)/4\pi$, the anisotropy coefficients A were evaluated for the assemblies studied. Asymmetry is also caused by the presence of a cavity under the ion guide in the spheres. To take this into account, the ratio $f_1(E)$ of neutron leakage spectra for spheres with and without such a cavity were calculated. Measurement of neutron leakage spectra by the time-of-flight method in conditions where the dimensions of the assembly in question are not negligible in relation to the time base, means that the procedure for unfolding the energy spectrum from the time may introduce systematic errors. The time-of-flight experiment and the data-processing procedure were modelled according to the BRAND program in order to evaluate this correction, $f_2(E)$.

Taking account of the above corrections and the angular anisotropy on the basis of experimental data measured at 8°, 30°, and 60°, the neutron leakage energy spectrum integral was obtained, corresponding to an isotropic source and a spherically symmetrical assembly. The experimental data are compared in Table 2; for the case of a lead sphere, they are shown on the diagram.

Neutron leakage spectra were calcuated according to the BRAND and BLANK programs by using evaluated nuclear data files from the NEDAM [13] and ENDF/BIV [14] libraries respectively. A correction is made to the spectrometer resolution function in the calculated neutron leakage spectra, the direct neutron flux spectrum from the target being used for the purpose.

Comparison of the calculated and experimental data (their ratio C/E is given in Table 2) shows discrepancies which usually arise from the experimental error (~ 6%). The most significant deviations are observed for Pb and U in the 6-10 MeV energy range, where the neutron leakage spectrum is underestimated in the calculation. This is a consequence of underestimating the cross-section in the high-energy region of the emission spectra when the 14 MeV neutrons interact with Pb and U nuclei. In order to verify this assumption, we replaced the emission spectra for U in the NEDAM library with a spectrum obtained in Ref. [15]. The ratio C/E then became 1.31 in the 3-6 MeV group and 0.94 in the 6-10 MeV group. If any general conclusion can be drawn on the reliability of the neutron data libraries tested in our work, then it is that NEDAM library describes the experimental data for Be and the ENDF/BIV library the data for U more accurately, and both are equally accurate in describing Pb.

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