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ABSTRACT

The work is devoted to analyses of neutron multiplication integral experiments on beryllium, carried out at Southwest Institute of Nuclear Physics and Chemistry (SWINPC), Chengdu, China.

Four parts are included. Part II gives a brief description of the SWINPC beryllium experiments. In Part III, Monte Carlo transport code MCNP is used to simulate the experiments in every detail, so as to analyze the questions concerning the experimental method and procedure, including validity of measurement by an ideal U–235 fission chamber detector, influence of the impurities in beryllium samples and D–T source structure materials, anisotropic effect of the experiment system, calculational factors and their errors. Part IV puts emphases on perturbation of the detecting system to measured values, caused by the in–system detecting procedure. ANISN–MORSE coupled method is developped to deal with the problem. Calculational results give a disappointing conclusion on the detecting system. Up to 6% integral deviations occur to the measured results, becoming experimental system errors. Corrections need to be made. In Part V, the beryllium experiments are simplified to a one–dimensional benchmark problem. The final experimental neutron leakage multiplications are compared with the benchmark calculations by using ENDF / B–IV, ENDF / B–VI, JENDL–3 and LASL–Sub beryllium data. Large differences are found between the neutron leakage multiplications by experiment and calculation, far beyond the experimental errors reported by experimenters, suggesting the inefficiency of the current beryllium evaluated data. Further studies on beryllium nuclear data are recommended.

Ⅰ. INTRUDUCTION

Nuclear data are essential in determining the neutronic behaviors in fusion reactor. With the fusion research moving forward to plasma energy breakeven, accurate nuclear data are becoming an urgent need for the fusion reactor design. Recent years have seen great progress in nuclear data research, indicated by the release of up–to–date evaluated data libraries, e.g. ENDF / B–Ⅵ, JENDL–3 and FENDL. As a result, for many materials involved, the nuclear data are satisfactory for fusion application. However, Nuclear data for some isotopes are still in poor situation. Beryllium, known as the best neutron multiplier in fusion reactor blanket, is among this list[1] . Differences are found between different evaluated beryllium data, and the evaluated Be–9 cross sections do not agree with the measured ones. Therefore, more efforts are motivated to put to beryllium nuclear data research.

In addition to differential nuclear data measurement and evaluation, integral experiment can play an important role in improving the quality of nuclear data. Several integral experiments[2,3,4,5] were conducted to test the effectiveness of the present beryllium evaluations. To meet the further needs for integral experiments on beryllium, neutron multiplication integral experiments were proposed and carried out, at Southwest Institute of Nuclear Physics and Chemistry, Chengdu, China, emphasizing on higher accuracy, thicker Be sample and symmetric geometry.

Analysis work is done to support the experiments. MCNP simulation, calculation of the detector perturbation and beryllium nuclear data test by the experiments are the main parts.

Ⅱ. SWINPC BERYLLIUM EXPERIMENTS

This part will give a description of the SWINPC integral experiments on beryllium. Experimental method, set–up and procedure are presented. Necessary parameters about the experiments are provided.

A Experimental Principle

The integral response in the experiments is the neutron leakage from a beryllium sphere sample for one 14 MeV source neutron, located at the center, expressed by Eq. (1),

$$M_L = \frac{L}{S} \tag{1}$$

where L is the neutron leakage from the Be sample, S represents the neutron source intensity. Obviously, because of the Be–9(n,2n) reaction, the neutron leakage multiplication M_L is greater than 1.0 and the integral response will reflect the nature of Be–9(n,2n) cross section to a great extent.

Total absorption method is adopted to measure both neutron leakage L and source intensity S. The experiment set–up is illustrated in Fig.1. A large volume of polyethylene sphere is designed to act as neutron moderator and absorber. Neutron leakage L and D–T neutron source intensity S are then determined by measuring the H–1(n,γ) absorption rates in the polyethylene region, when the Be sample is placed (with–Be system) and is removed (no–Be system), respectively.

From the conservation of neutron number, we have,

$$L = A_H^M + A_C^M + L_{PE}^M \tag{2}$$

for with–Be system,

$$S = A_H^0 + A_C^0 + L_{PE}^0 \tag{3}$$

for no–Be system,

where, A_H,A_C,L_{PE} are total absorption of H and C in the polyethylene region, and neutron leakage from the polyethylene sphere, superscripts "M" and "0" denoting with–Be system and no–Be system.

Then, the leakage multiplication M_L is achieved by:

$$M_L = \frac{A_H^M + A_C^M + L_{PE}^M}{A_H^0 + A_C^0 + L_{PE}^0} \tag{4}$$

or,

$$M_L = M_{app} \cdot F_C \tag{5}$$

here,

$$M_{app} = \frac{A_H^M}{A_H^0} \tag{6}$$

$$F_C = \frac{1 + \dfrac{A_C^M}{A_H^M} + \dfrac{L_{Be}^M}{A_H^M}}{1 + \dfrac{A_C^0}{A_H^0} + \dfrac{L_{Be}^0}{A_H^0}} \tag{7}$$

M_{app} , called apparent multiplication, is obtained by measuring the H–1(n,γ) rates in polyethylene sphere, for both with–Be and no–Be systems, while F_C , the calculational factor, is provided by calculation.

B Neutron Source, Beryllium Samples and Moderator

Source neutrons are generated by bombardment of D^+ beam (E_d^0 = 150 keV) on the TiT target (1.0 g / cm^2) through reaction:

$$D + T \rightarrow {}^{4}_{2}He + n + 17.6MeV \tag{8}$$

Source structure and dimensions are detailed in Fig. 2.

Certainly, there are some differences between the actual source and the ideal mono–energic and isotropic source in source characteristics.

The beryllium spheres used in the experiments are combined by seven shells from U.S.A. and from P.R.C.. Their parameters are listed in Table 1. The geometry is ideal spherical except a hole of diameter 3.0 cm (see Fig.3), providing a channel for deuteron duct.

Table 1. Beryllium shell parameters

	inner–outer radius (cm)	density (g/cm^3)	content (wt%)
U.S.A.	2.5–5.7, 12.8–13.4, 13.4–15.2, 15.2–17.35	1.84	Be–98.82, O–0.72, Fe–0.13, C–0.11, Al, Si, Mg, Ni
P.R.C.	5.7–6.9, 6.9–9.7, 9.7–12.8	1.82	Be–98.9, O–0.5, Fe–0.6

Leakage multiplications for three thicknesses of beryllium samples (inner–outer radius, 12.8–17.35, 6.9–17.35, 2.5–17.35) have been measured.

Polyethylene is chosen as the moderator, because of its strong neutron moderation and absorption capability, and the good knowledge of H–1 and C–12 nuclear data. Three layers of polyethylene with total diameter up to 138 cm are employed in attempt to decrease the leakage out of the polyethylene sphere. There exists a cylinder void to accomodate the D^+ beam duct (see Fig.4).

Table 2. Moderator parameters

	inner–outer radius (cm)	composition	density (g
cm^3)	region 1	20.0–21.7	CH_2
0.92	region 2	21.7–28.0	CH_2
0.56	region 3	28.0–69.0	CH_2

C Detector and Measurement

As pointed out above, the quantities to be measured are the total H–1(n,γ) absorption rates over the whole polyethylene sphere.

In the experiments, U−235(n,f) response distribution R(r,θ), r the radius and θ the direction angle, is first ascertained, using U−235 fission chamber detector, by moving the detector along five detecting channels at 0 ° , 40 ° , 80 ° , 120 ° and 150 ° angles to D^+ beam direction. In this way, the detector responses $R(r_i,\theta_i)$ are measured as the first−hand experimental data. In order to obtain response distribution R(r,θ), an assumption is made that the distribution obeys linear interpolation between measured points. For the variable r, the assumption can be assured by making radius differences of adjacent measured points small enough. However, as to the direction angle θ, the interpolation needs to be verified, because only at five directions measurements are carried out.

From the known detector response function R(r,θ), the total H−1 absorption rate can be relatively determined by:

$$A_H = C \cdot \left[n_1 \int_{region\ 1} R(r,\theta)dV + n_2 \int_{region\ 2} R(r,\theta)dV + n_3 \int_{region\ 3} R(r,\theta)dV \right] \tag{9}$$

The factors n_1,n_2,n_3, atomic densities of H−1 in the three polyethylene regions, are introduced considering that detector medium U−235 is of the same atomic densities while the H−1 atomic densities are different in the three regions. "C" is a constant dependent only upon the detector efficiency.

Eq.(9) implys that H−1(n,γ) reaction can be relatively represented by U−235(n,f) reaction. This will be tested in the following analyses.

Therefore, apparent multiplication M_{app} is achieved by:

$$M_{app} = \frac{n_1 \int_{region1} R^M(r,\theta)dV + n_2 \int_{region2} R^M(r,\theta)dV + n_3 \int_{region3} R^M(r,\theta)dV}{n_1 \int_{region1} R^0(r,\theta)dV + n_2 \int_{region2} R^0(r,\theta)dV + n_3 \int_{region3} R^0(r,\theta)dV} \tag{10}$$

where $R^M(r,\theta)$ and $R^M(r,\theta)$ are detector response distributions for with−Be and no−Be systems.

Ⅲ. MCNP SIMULATION OF THE EXPERIMENTS

In this part, the Monte Carlo transport code MCNP[6] is used to simulate the experiments in detail, so as to provide answers for issues concerning experimental method and procedure.

A MCNP Model

Four MCNP input files are prepared to carry out simulations, corresponding four measured systems, i.e. no−Be and with−Be (12.8−17.35, 6.9−17.35, 2.5−17.35).

The geometry and compositions are rigorously modelled, by using MCNP's flexible input function. Nuclear data are in pointwise ACE format, processed from different evaluation libraries (Be, C from LASL–Sub; H,O,Al,Fe,U from ENDF / B–IV; Mo from ENDL–73). In thermal energy region, free gas model treatment is adopted.

Subroutine SOURCE is modified and embedded into MCNP system, so as to simulate the source neutron generation process, because the actual D–T neutron source is not a standard MCNP source. Considering the moderation of D^+ beam in TiT target and the two–body reaction kinematics, the source intensity distribution can be expressed as,

$$S(\vec{r},\vec{\Omega},E_n)d\vec{r}d\vec{\Omega}dE_n = n_t I_d \sigma_{D-T}(E_d)\left(\frac{dE_d}{dZ}\right)^{-1} dE_d dX dY \cdot \frac{d\vec{\Omega}_c}{4\pi} \cdot \delta(E_n - E_n^*)dE_n \qquad ......(11)$$

where, $\vec{r}$—position vector where D(T,n)α occurs,

$\vec{\Omega}$—secondary neutron direction vector in lab system,

E_n—secondary neutron energy,

E_d—D^+ energy at which D(T,n)α occurs,

X,Y,Z—coordinate values, equivalent to $\vec{r}$,

$\vec{\Omega}_c$—secondary neutron direction vecter in CM system,

n_t—tritium atomic density in TiT target,

I_d—D^+ beam intensity,

$\sigma_{D-T}(E_d)$—D(T,n)α reaction cross section[7],

$\frac{dE_d}{dZ}$— D^+ energy slow–down rate in TiT target[7],

E_n^*—secondary neutron energy, determined by kinematics[7].

In Eq.(11), the D(T,n)α secondary neutron angular spectrum in CM system is assumed to be isotropic. Random sampling according to the source distribution is made in subroutine SOURCE to determine source neutron parameters $\vec{r},\vec{\Omega},E_n$.

Output tallies are carefully designed to obtain quantities of interest. "Flag" technique is used to discriminate the contribution of neutrons reflected from the polyethylene region to the process rates in beryllium samples. The whole polyethylene sphere is divided into pseudo sub–zones, in order to investigate anisotropicy of the system. U–235(n,f) cross sections are introduced as the flux multiplier (response function) to imitate the measurements in polyethylene of an ideal U–235 fission detector.

The simulations are conducted on SUN Work Station.

B Results

After sufficient source neutrons are tracked to guarantee the statistic deviations of interest quantities, calculational results are obtained and are shown in Table 3 and Fig. 5,6,7,8. The results are normalized to one source neutron.

Table 3 lists process rates in all regions for four measured systems."Flagged" means the contribution of reflected neutrons. U−235(n,f) detector responses is weightedly summed over three layers of polyethylene according to Eq.(9). The negative values of Mo absorption indicate that multiplication process occurs to Mo nuclide.

Fig. 5,,6,7,8 depict the anisotropicy of the H−1(n,γ) reaction in the polyethylene region for four measured systems. Step−jump curves are presented, because countings are made in every pseudo zone, divided by equal direction consine.

Table 3. Calculational results for the experiments by MCNP

region	process	No−Be	With−Be 12.8−17.35	With−Be 6.9−17.35	With−Be 2.5−17.35
source structure	Al absorption	1.770e−3	2.194e−3	2.715e−3	3.340e−3
	Mo absorption	−9.260e−3	−9.102e−3	−8.996e−3	−8.780e−3
beryllium sphere	total Be(n,2n)	—	3.669e−1	7.680e−1	9.864e−1
	flagged Be(n,2n)	—	9.803e−3	7.923e−3	6.153e−3
	total Be(n,x)	—	4.240e−2	1.122e−1	1.571e−1
	flagged Be(n,x)	—	9.301e−3	2.704e−2	3.804e−2
	C absorption	—	4.632e−5	3.509e−5	5.924e−5
	O absorption	—	3.949e−4	7.101e−4	8.643e−4
	Fe absorption	—	4.212e−4	4.498e−3	7.580e−3
	leakage	1.000	1.327	1.651	1.812
polyethylene sphere	H−1(n,γ)	8.572e−1	1.217	1.580	1.761
	C absorption	6.782e−2	5.370e−2	3.973e−2	3.263e−2
	U5(n,f) response	1.739e+4	2.467e+4	3.201e+4	3.564e+4
	leakage	8.243e−2	5.672e−2	3.773e−2	2.804e−2

C Discussions and Conclusions

From the MCNP simulation results, issues concerning experimental method and procedures are discussed.

① In the experiments, H−1(n,γ) reaction rates in the polyethylene are relatively measured by U−235 fission chamber detector. The accuracy of the assumption is of primary concern. Table 4 lists the comparison of H−1(n,γ) absorption rates and ideal U−235(n,f) detector responses. As to relative values, very good agreements are shown. This results from the same 1 / V cross section law that H−1(n,γ) and U−235(n,f) reactions obey in thermal energy region, where most of these two reactions occur in the polyethylene sphere.

Table 4. Comparison of H−1(n,γ) and U−235(n,f) rates in polyethylene

process		no−Be	with−Be 12.8−17.35	with−Be 6.9−17.35	with−Be 2.5−17.35
H−1(n,γ)	value	8.572e−1	1.217	1.580	1.761
	relative	1.000	1.420	1.843	2.054
U−235(n,f)	value	1.739e4	2.467e4	3.201e4	3.564e4
	relative	1.000	1.419	1.841	2.049

② From Table 3, we can conclude that the presence of source structure (Al,Mo) and the impurities (C,O,Fe) in the beryllium samples has little influence on the leakage multiplications. Less than 1% neutrons are absorbed by these nuclides.

③ The polyethylene sphere was designed as a device to measure the neutron leakage multiplications from beryllium samples. However, the actual multiplications, measured by this method (total absorption method), will be deflected from what is expected for bare samples. Neutrons reflected by the polyethylene sphere may enter to beryllium again, therefore change the process rates in the beryllium sphere.

The contributions of the reflected neutrons are labelled with "flagged" in Table 3. Be(n,2n) rates are only slightly increased, while Be(n,x) reaction suffers from remarkable changes, as a consequense of the soft energy spectrum of the reflected neutrons. Overall, the neutron leakage multiplications will decrease by −0.0005, 0.020 and 0.032, for 12.8−17.35, 6.9−17.35 and 2.5−17.35 beryllium samples respectively, from the values of bare beryllium spheres.

④ As stated in Part II, after measurements are conducted at five directions, 0 ° ,40 ° ,80 ° ,120 ° ,150 ° angles to D^+ beam direction, linear interpolation scheme is employed to acquire the total hydrogen absorption in the whole polyethylene sphere. This assumption is verified by Fig. 5,6,7,8, which show that the anisotropic effect is not strong and can be well represented by linear interpolation between these measured directions.

⑤ The main task of the experimental analyses is to convert the measured "apparent multiplications" to neutron leakage multiplications, by providing calculational factor F_c in Eq.(5). Based on Table 3, we calculated the factors by Eq.(7). In addition, the errors of the calculational factors are estimated by presuming the calculational errors of each term in Eq.(7) to be 10%, according to error propagation rule.

We can see from the following table that, though the final results of the experiments depend on calculation, the dependence is very weak and the calculational errors will not overshadow the experimental accuracy. This is because most of the neutrons leaking from the beryllium samples are absorbed by hydrogen in polyethylene, which are measured.

Table 5. Calculational factors and their error estimation

Be sample	F_c	error
12.8–17.35	0.929	1.2%
6.9–17.35	0.893	1.1%
2.5–17.35	0.880	1.1%

Ⅳ. PERTURBATION OF THE DETECTING SYSTEM

In the previous part, detailed simulation of the experiments produced positive evaluation on the experimental method and procedure, given the detector system is ideal infinite small. However, in fact, the detecting system is far from ideal. The measurements are carried out along detector duct in the polyethylene sphere and detecting system is of certain size. In this case, the in–system measurement will give rise to neutron flux perturbation around the detector region. Consequently, measured values may be distorted. This part will analyze the perturbation.

A Detecting System

Fig.9 is a schematic picture of detecting system.

In front of the U–235 media foil, there is a layer of cupper, 1mm thick. When detector was moved outside, the void before the detector is filled by polyethylene cylinder of density 0.92 g/cm^3 , although the polyethylene sphere consists of three layers with different densities.

Obviously, neutron flux tends to be perturbed, caused by perturbation of composition distribution around the duct.

B Analysis Method

In order to ascertain the perturbation of neutron flux arround the detector, one–dimensional calculation is not enough. However, for such a large experiment system (138cm in diameter), it is not economical and efficient to make three–dimensional Monte Carlo modelling, even impossible to reach satisfactory accuracy. Taking advantage of the small size of disturbed region, ANISN[8] coupling MORSE[9] method is developped. For the unperturbed system, suppose there is no detector duct, neutron flux is computed by 1–D Sn transport code ANISN. Then, MORSE, 3–D Monte Carlo transport code, adopting the same format nuclear constants as ANISN, is chosen to study the flux behavior in the perturbed region, while the boundary conditions of MORSE calculation are determined by ANISN flux.

①Nuclear constants

25–group, P_3 constants, processed by NJOY[10] from ENDF / B–Ⅵ.

②MORSE boundary

MORSE boundary is a hypothesized cylinder, enclosing the detector duct. Determination of the cylinder dimension is a tradeoff between calculation accuracy and expenses. The cylinder should be large enough, so the boundary flux can be regarded as unperturbed, on the other hand, as small as possible so as to save calculation costs.

③ANISN–MORSE coupling

The surface source intensity on the MORSE boundary is expressed as:

$$dI = \varphi_g(r,\mu)\ |\vec{\Omega} \cdot \vec{n}|\ dS\ d\vec{\Omega}$$

$\varphi_g(r,\mu)$ is the neutron flux given by ANISN, $\vec{\Omega}$ representing the neutron direction vector, $\vec{n}$ the normal vector for surface cell dS. After discretation, the source neutron probability distribution versus group number, position and direction is obtained. The subroutine SOURCE in MORSE is revised to accomplish the source particle sampling from the probability distribution. Thus, ANISN and MORSE are coupled.

C Results and Conclusions

ANISN calculations were first conducted for no–Be sample system and for with–Be sample systems. Then ANISN fluxes were used to prepare source neutron probability distribution for SOURCE in MORSE. Similar to the experiment measurement procedure, at every detected point in the polyethylene sphere, MORSE was run to compute the perturbed detector responses, i.e. U–235 fission rates. Unperturbed and perturbed detector responses at detected points are obtained. The perturbation of integral values of interest is then calculated by integration and is shown in Table 6.

Table 6. Calculated perturbation of apparent multiplications

Be sample	unperturbed M_{app}	perturbed M_{app}	ratio
12.8–17.35	1.433	1.390	1.031
6.9–17.35	1.895	1.827	1.037
2.5–17.35	2.154	2.033	1.060

For the most important values to be measured — apparent multiplications, Analyses indicate that unnegligible perturbations took place, because of the in–system measurement. In order to correct this measurement system errors, a simple approach is adopted. By assuming the ratioes in Table 6 hold true for actual measurements, the measured values, perturbated, are corrected to unperturbed ones in Table 7.

Table 7. Correction of measured apparent multiplications

Be sample	Measured M_{app} [11]	Corrected M_{app}
12.8–17.35	1.371	1.414
6.9–17.35	1.711	1.774
2.5–17.35	1.832	1.942

We can conclude that the perturbations have large negative effect to the reliability of the experimental results, because of the fatal defect of the detecting system. It is necessary to make perturbation correction to the raw experimental data.

The experimenters have also tried to investigate this perturbation by measurements.

V. TEST OF BERYLLIUM NUCLEAR DATA

The main goal of this experiments is to establish a benchmark to check the beryllium nuclear data. Several beryllium evaluations are processed for this benchmark calculation. Comparisons between calculations and the experiments are made. Remarks on beryllium nuclear data are drawn.

A One–Dimension Model of the Experiments

If accurate calculation for the experiments is intended, three dimensional description for the experiments must be given, because the actual geometry and neutron source have some deviation from ideal one–dimension. For the purpose of benchmark test, it is appropriate to model this experiments with simplified 1–D description.

Another MCNP calculation is carried out, with the experiments being described with simple 1–D benchmark model, in which, the deuteron beam duct and the D–T source structure are omitted to form an exact spherical geometry and the source is substituted by an isotropic point source, neutron energy evenly distributed between 13.5 and 15.0 MeV.

When compared with the results of exact simulation in Part II, the feasibility of the model simplification is examined in Table 8. Thanks to the experimenters' efforts to establish one–dimensional benchmark, this 1–D benchmark model can well represent the

SWINPC Be experiments.

Table 8. Neutron multiplications by 1–D and 3–D MCNP calculations

Be sample	accurate 3–D	simplified 1–D	ratio
12.8–17.35	1.327	1.332	1.004
6.9–17.35	1.651	1.667	1.010
2.5–17.35	1.812	1.840	1.015

B Benchmark Test of Beryllium Data

Final experimental results are obtained from the corrected measured apparent multiplications in Table 7 and the calculational factors in Table 5.

Current JENDL–3, ENDF / B–Ⅳ, ENDF / B–Ⅵ and LASL–Sub evaluations are used to interpret the beryllium experiments, by ANISN or MCNP code, with help of the one–dimensional benchmark model.

Results of experiments and calculations are compared in Table 9.

Table 9. Beryllium neutron leakage multiplications of SWINPC experiments

Be sample	experiment	calculation			
		ENDF / B–Ⅳ ANISN	ENDF / B–Ⅵ ANISN	JENDL–3 ANISN	LASL–Sub MCNP
12.8–17.35	1.314	1.361	1.323	1.340	1.332
6.9–17.35	1.584	1.751	1.667	1.710	1.667
2.5–17.35	1.709	1.948	1.849	1.907	1.840

Taking the experimental results as standard, all these beryllium evaluations over–estimate the neutron multiplication capability of beryllium. Although the ENDF / B–Ⅵ and LASL–Sub calculations give closer results to the experimental ones, the disagreements are still far beyond the experimental errors, reported to be 3%. This may suggest the inefficiency of these beryllium nuclear data.

There is a confusing situation that the SWINPC experiments contradict with INEL beryllium experiments[5], carried out in U.S.A. about at the same time, which provided rather consistent results with LASL–Sub calculations.

Ⅵ. SUMMARY

Putting aside problem involving the detecting system, detailed MCNP simulation verified the experimental procedure and method. The U−235 media can relatively measure the hydrogen absorption in polyethylene sphere with perfect accuracy. The source structure and impurities in beryllium samples barely influence the beryllium neutron multiplications. The linear interpolation assumption about the anisotropic effect of H−1(n,γ) reaction distribution brought about no errors than tolerable. Though the experimental results need assistance of calculation, the experiment design assured that the result errors come mainly from measurements.

The in−system measurement scheme proved to be of fatal defect. Analyses by ANISN−MORSE coupling method showed that large perturbations occur to the measured values. To rectify the shortcoming, corrections should be made to meausred apparent multiplications.

By using the corrected experimental results, benchmark tests of different beryllium nuclear data are conducted. It is shown that no beryllium evaluations can predict the experimental results satisfactorily. The current beryllium evaluations produce too high neutron multiplication, for 2.5−17.35 Be sample, ENDF / B−Ⅵ 8% higher and JENDL−3 11% higher.

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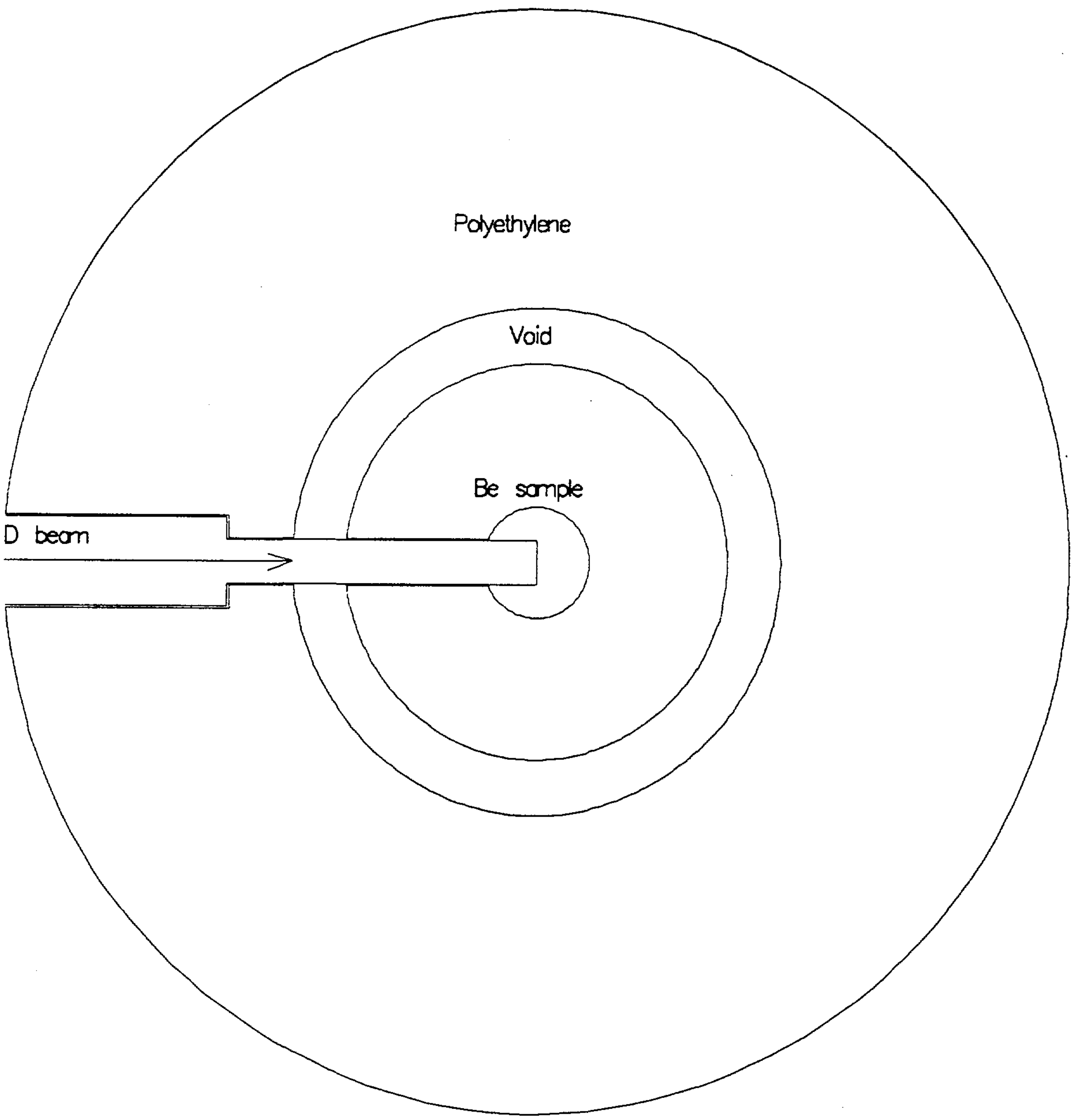

Fig. 1. SWINPC experiment scheme

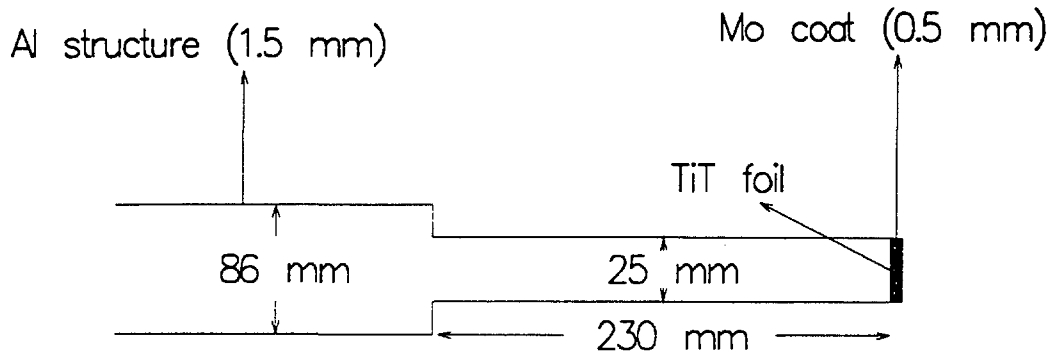

Fig. 2. D–T neutron source in SWINPC experiments

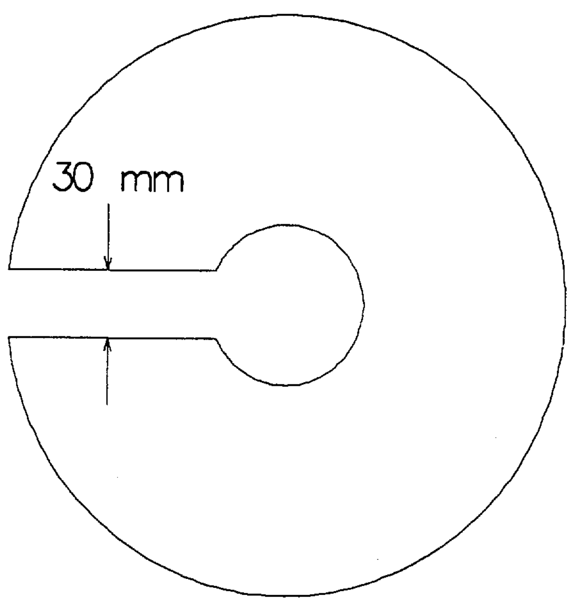

Fig. 3. Beryllium sample in SWINPC experiments

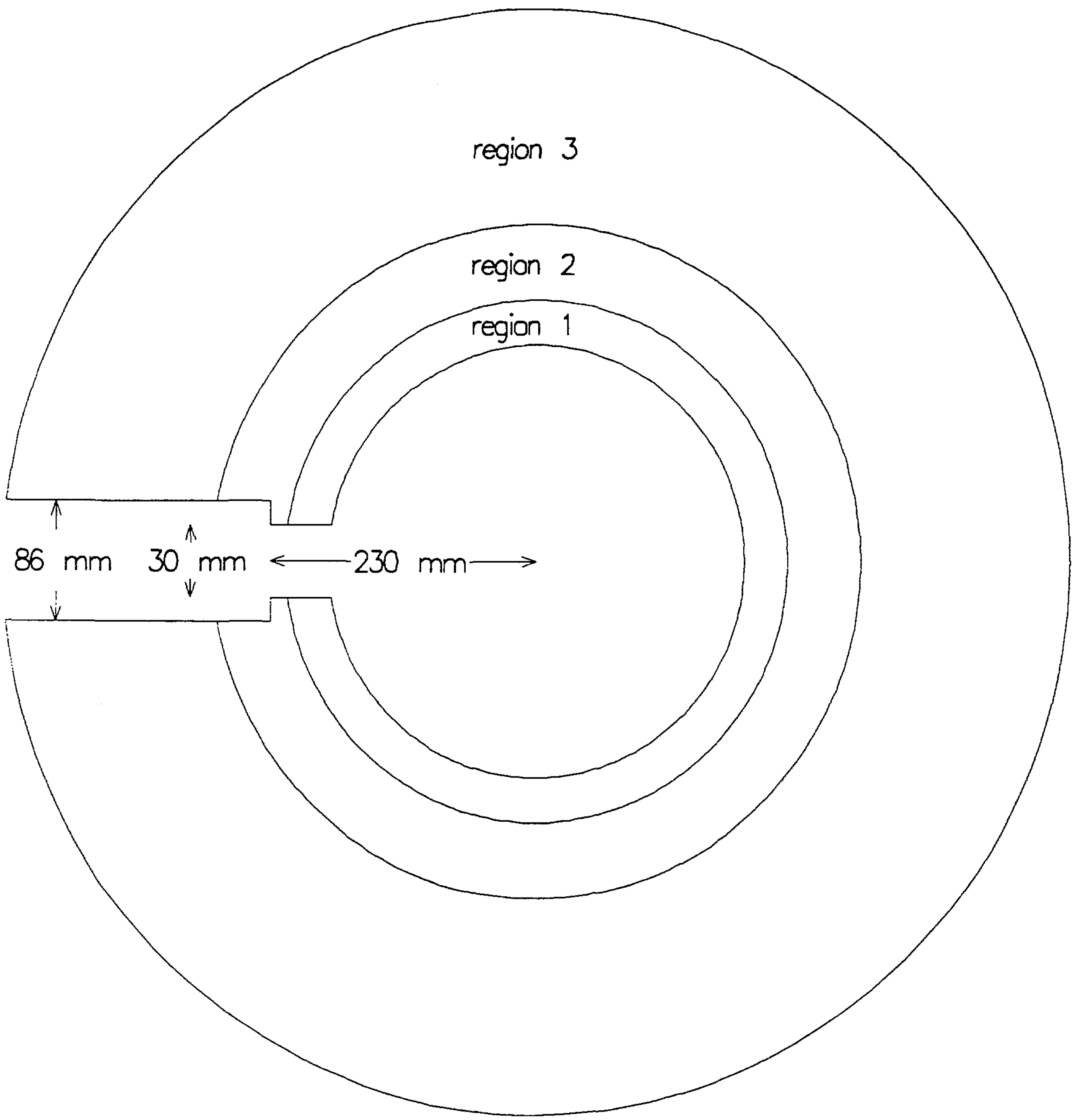

Fig. 4. Polyethylene moderator in SWNPC experiments

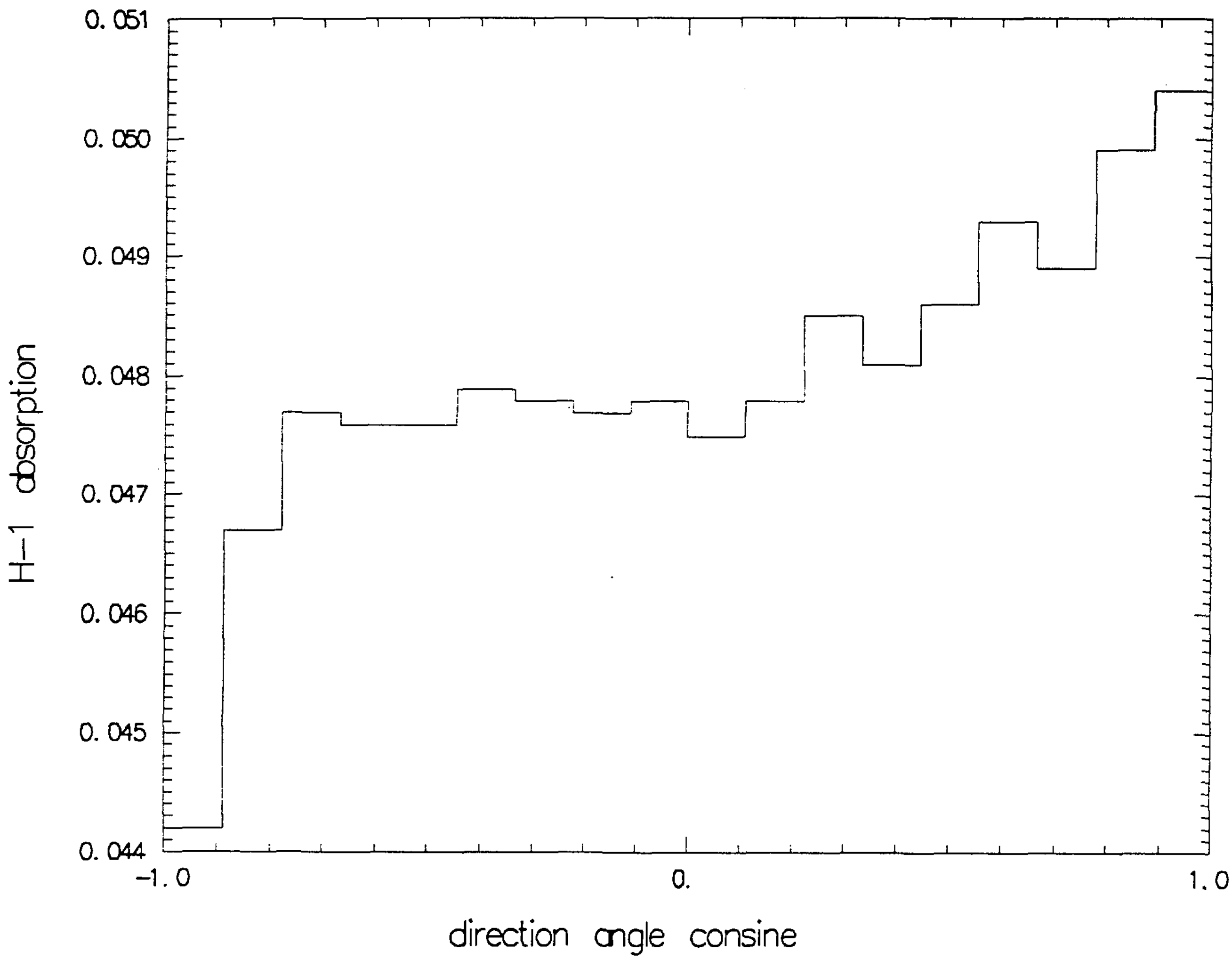

Fig. 5. Anisotropicy of H-1(n,r) in polyethylene for no-Be system

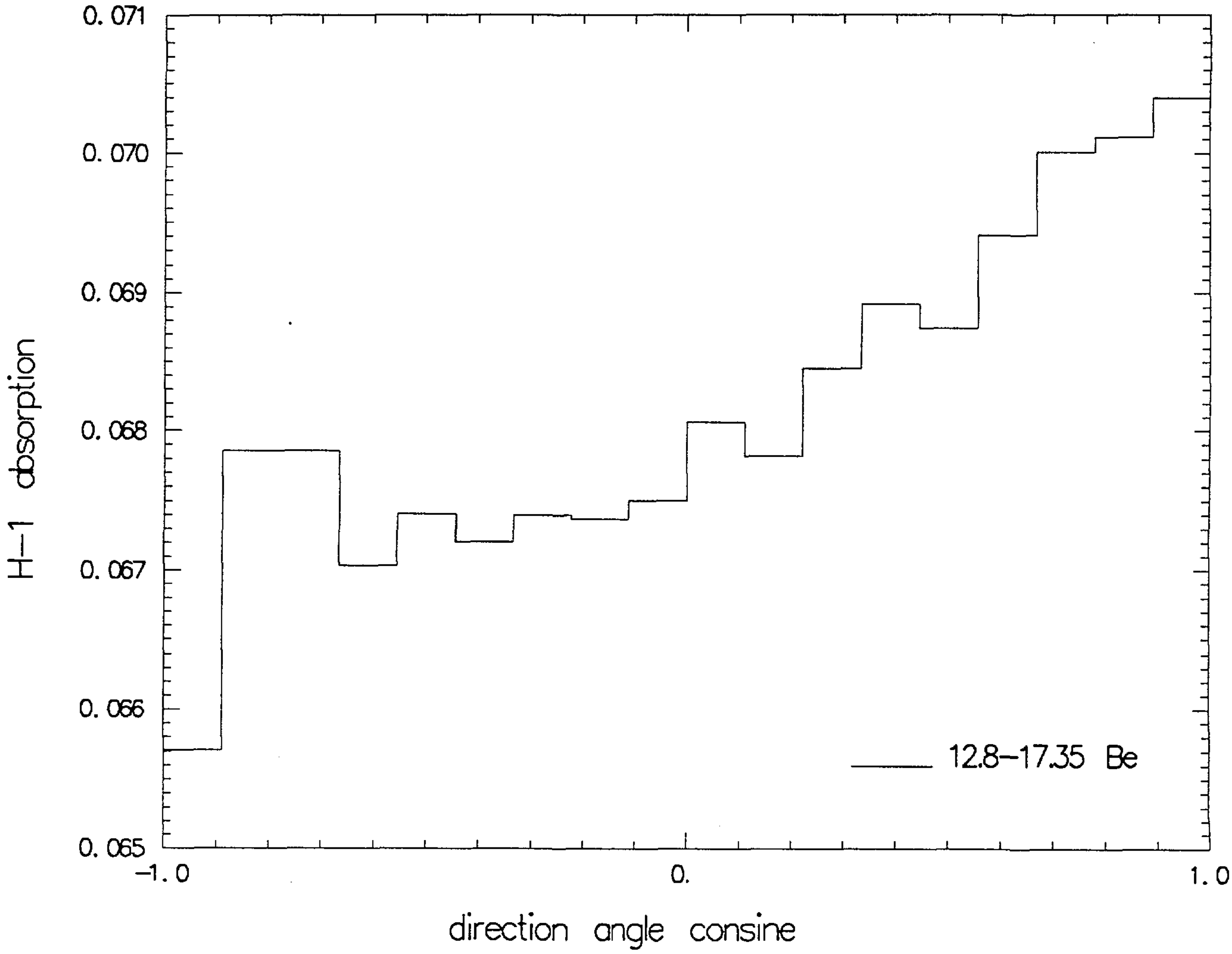

Fig. 6. Anisotropicy of H−1(n,r) in polyethylene for with−Be system

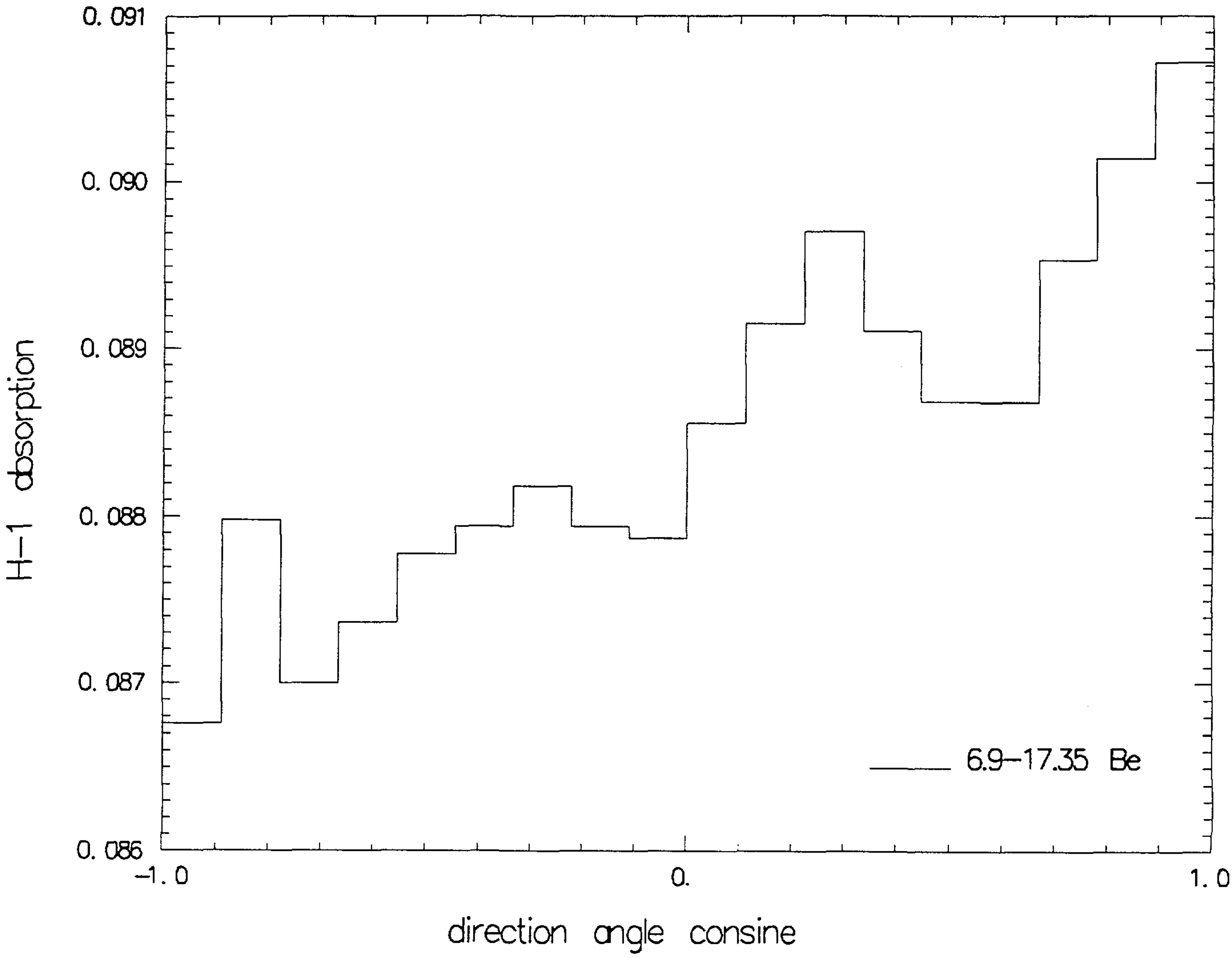

Fig. 7. Anisotropicy of H-1(n,r) in polyethylene for with-Be system

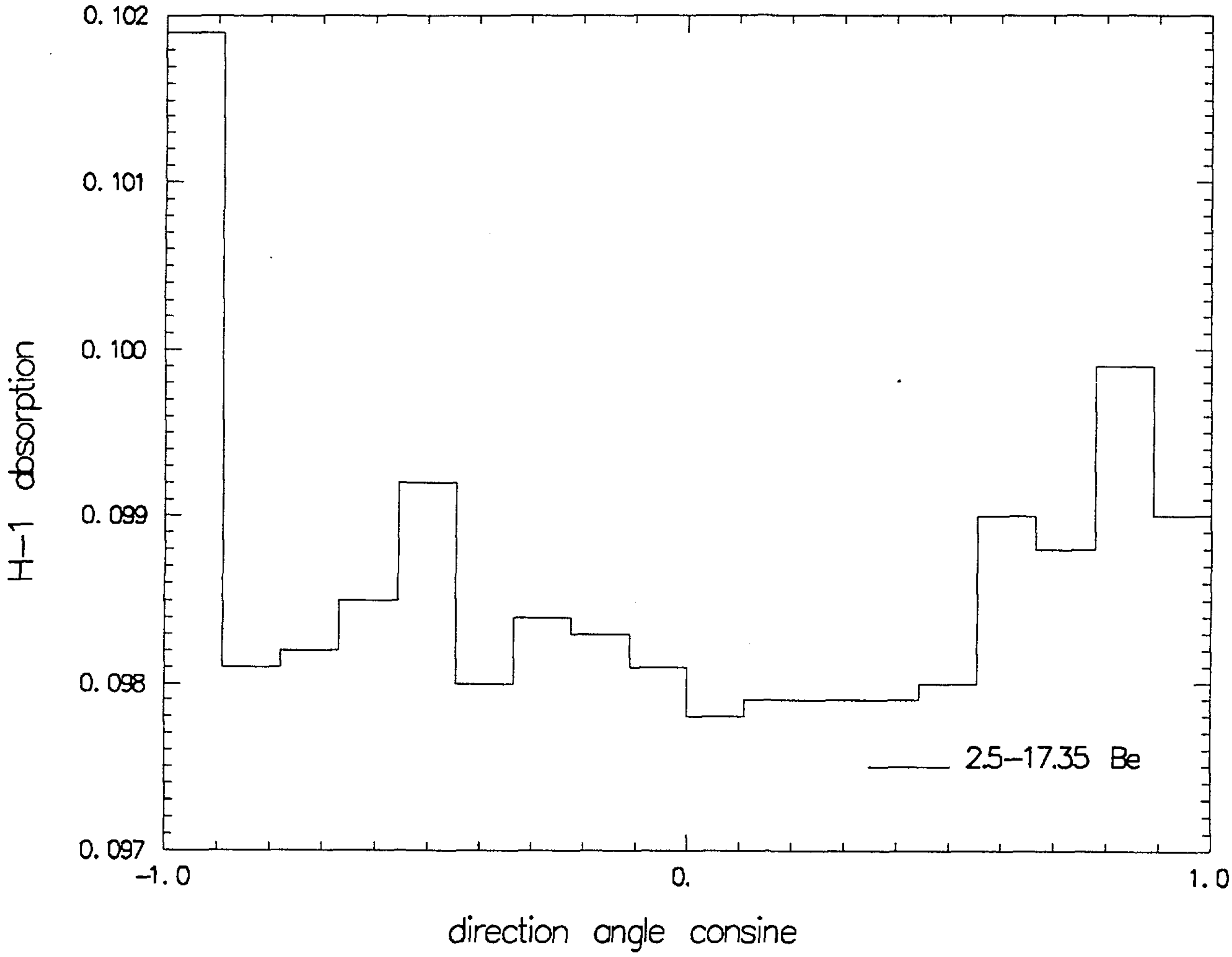

Fig. 8. Anisotropicy of H–1(n,r) in polyethylene for with–Be system

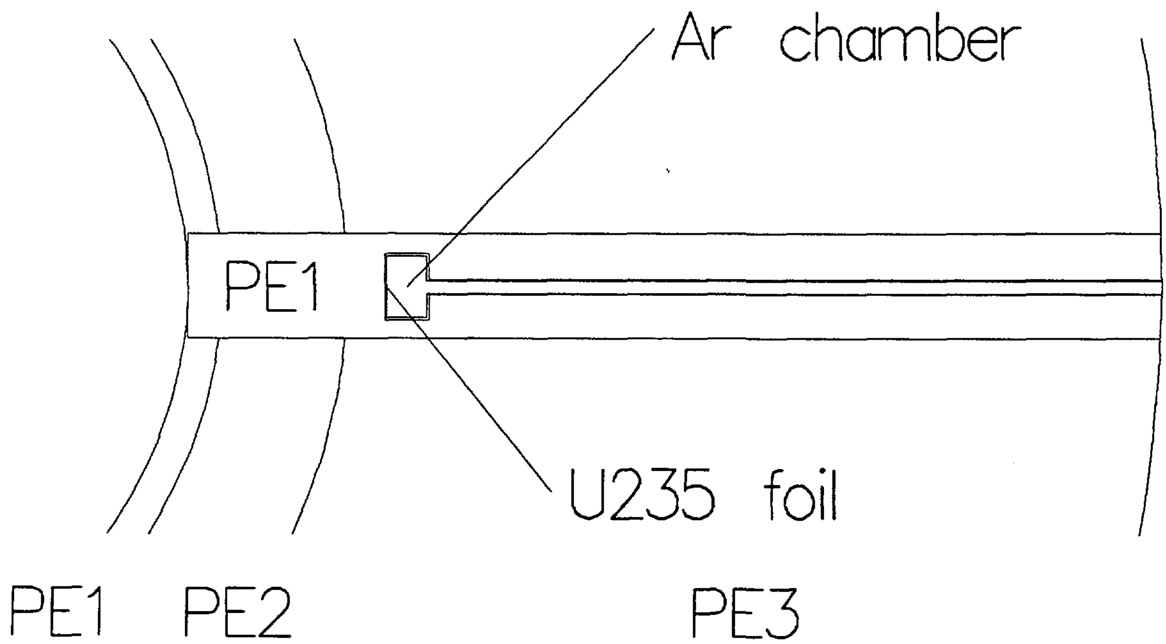

Fig. 9. Detecting system of SWINPC experiments