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Abstract:

A neutron detector is described which measures the neutron dose rate in rem/h independently of the energy of the neutrons from thermal to 15 MeV.

The detector consists of a BF_3 proportional counter surrounded by a shield made of polyethylene and boron plastic that gives the appropriate amount of moderation and absorption to the impinging neutrons to obtain rem response. Two different versions have been developed. One model can utilize standard BF counters and is suitable for use in installed monitors around reactors and accelerators and the other model is specially designed for use in a portable survey instrument. The neutron rem counter for portable instruments has a sensitivity of 2.4 cps/mrem/h and is essentially nondirectional in response. With correct bias setting the counter is insensitive to gamma exposure up to 200 r/h from Co^{60}.

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1. Definitions of concepts

The quantity used as a measure of radiation damage for radiation protection purposes is the dose equivalent and the unit of dose equivalent is the rem. The International Commission on Radiological Units and Measurements (ICRU) has recommended the use of this quantity and given the definitions (1). The International Commission on Radiological Protection (ICRP) uses the rem in its radiation protection guides. The dose equivalent in rem is defined as the absorbed dose in rad multiplied by certain quality factors (QF). The QF are used to express the modification of biological effect due to linear energy transfer (LET), relative damage factor (n) and other conditions. ICRP has furnished specific values of the QF to be used in different regions of LET.

The concepts given above are also applied to neutron irradiation of tissue. The radiation damage is described by the dose equivalent distribution in the exposed volume. W. S. Snyder and J. Neufeld, Oak Ridge, USA, have calculated the absorbed dose distribution and the LET distribution in an infinite slab of tissue, 30 cm thick, irradiated by a broad beam of monoenergetic neutrons. The dose equivalent distribution in $rem/n/cm^2$ in the slab could then be determined. Their results have been used to establish a Radiation Protection Guide (RPG) for neutrons, which is at present widely followed. The calculations and the RPG are presented in US National Bureau of Standards Handbook 63 (1957). We have reproduced in Fig. 1 the curve showing the maximum dose equivalent in $rem/n/cm^2$ in the tissue slab as a function of neutron energy. This curve is fundamental for the work described in this report. Table I gives the neutron flux densities at different energies that produce 2.5 mrem/h. These figures are usually referred to as the RPG for neutrons or the maximum permissible neutron flux densities. They follow from the curve in Fig. 1.

2. Measurement of neutron dose equivalent

The measurement of dose equivalent in the case of neutron irradiation appears to have involved considerable difficulties. It is not really possible to design an instrument with a physical response mechanism that is in accordance with the definition of the rem unit. The dose equivalent can be calculated by use of the information in Fig. 1 if the neutron fluence and the neutron energy spectrum are known. But since neutron spectrum measurements are usually very complicated and time-consuming, it is not a procedure that can be applied in ordinary radiation protection work. H. H. Rossi, Columbia University, New York, USA, has developed a device for measuring the distribution of absorbed dose in LET, from which the dose equivalent can be calculated (2). But for the purpose of practical health physics it is necessary to have a more simple routine.

A convenient method of measuring dose equivalent is to use an instrument that has been given artifically a neutron sensitivity, e.g. in counts/n/cm^2, that varies with neutron energy in accordance with the curve shown in Fig. 1. The number of counts from such a detector is proportional to the dose equivalent in rem, and knowledge of the energy distribution of the neutrons is superfluous.

This approach to the problem of measurement of neutron dose is commonly used. Detectors that have correct dose response over certain energy regions have been described in the literature. Hurst, Ritchie and Wilson (3) designed a proton-recoil proportional counter whose response to unidirectional neutrons is such that the number of counts produced is approximately proportional to the first-collision tissue dose for neutron energies between 0.2 and 10 MeV. Dennis and Loosemore (4) have described a proton-recoil proportional counter essentially nondirectional in response and with an energy response curve that follows the 1954 recommendations of the International Commission on Radiological Protection over the energy range 0.1 - 15 MeV. De Pangher (5) has shown that a paraffin moderator and a BF_3 counter can be arranged in such a way that the count rate is approximately proportional to the dose rate in rem/h in accordance with the RPG in NBS Handbook 63 over a range from 0.2 - 5 MeV. Basson (6) has presented a neutron detector consisting of a 6.3 cm diameter sphere surrounding a boron-loaded scintillation counter with a response proportional to within ± 12 % of the dose rate in rem/h for energies between 1 ev to 10 keV.

A thermal neutron detector surrounded by moderating material is sensitive to neutrons of all energies. The composition and shape of the moderator has a large influence on the sensitivity as a function of neutron energy. In this report we shall describe an investigation made to find out whether it was possible to design a detector according to this principle and having an approximately correct dose response over the energy range from thermal to 15 MeV neutrons. We also took into consideration that the detector should be suitable for use in a portable instrument for radiation protection survey.

R. L. Bramblett, R. I. Ewing and T. W. Bonner, Rice University, USA (7), D. E. Hankins, Los Alamos, USA (8), and K. D. Androsenko and G. N. Smirenkin, USSR (9), have described counters that consist of a polyethylene or a paraffin shield around a thermal neutron detector, which are useful as neutron dosimeters over a wide energy range. By using a more elaborate shield around the thermal neutron detector we hoped to get improved approximation to the ideal dose response and a considerable reduction in weight.

A progress report of this work was presented at the IAEA Symposium on Neutron Dosimetry in Harwell 1962 (10).

3. Description of detector arrangement

The detector design that was selected for investigation is shown in Fig. 2. A BF_3 proportional counter is surrounded by two cylindrical layers of polyethylene separated by a 5 mm layer of boron plastic. Circular disks of the same materials and thicknesses make up the ends of the cylinders. The boron plastic layer contains 200 mg/cm^2 of boron. It is made of a 2.5 mm sheet of the material* wrapped

* Delivered by Firma N. Örne, Banérgatan 81, Stockholm, Sweden

twice. The cylindrical polyethylene layers are made by winding 1 mm sheets of polyethylene. The adjustment of the neutron sensitivity as a function of energy to fit the rem dose curve in Fig. 1 was to be carried out by variation of the three following parameters:

1. the thickness (d_1) of the inner polyethylene layer
2. the thickness (d_2) of the outer polyethylene layer
3. the number of holes drilled through the boron plastic layer.

In the first series of experiments it was intended to have the neutrons incident normally to the axis of the counter.

4. Theory

It was felt to be necessary to develop at least a crude theory for the purpose of guiding the experimental work. In a geometry as complex as that of our counter, Monte Carlo calculation seemed to offer the best approach for accurate results. But as it was felt that in any case the final adjustments must be made by trial and error, this expensive approach was not tried.

A simplified theory, originally used for guiding the initial experiments, explains the principle of the counter and is therefore given below. The theory was based on the following model. All neutrons entering the counter and making their first collision in the outer polyethylene layer were considered as being absorbed in the boron shield, e. g. lost for the purpose of detection. On the other hand, all neutrons making their first collision in the polyethylene region inside the boron shield were regarded as counted in the detector with a probability k.

With these assumptions the probability of obtaining a pulse from a neutron of energy E incident normally to the axis of the counter is

$$k\,e^{-\Sigma(E)d_2}\left[1 - e^{-2\Sigma(E)d_1}\right],$$

where $\Sigma(E)$ is the macroscopic total scattering cross-section and d_1 and d_2 are as given in Fig. 2.

The values of d_1 and d_2 should be chosen to give the function the same energy dependence as the rem dose curve. One would also like to have d_1 as large as possible to obtain a high sensitivity and $d_2 + d_1$ as small as possible to get a counter which is easy to handle. With the simplified theory the optimum solution of this problem is $d_1 = 16$ mm and $d_2 = 32$ mm. The calculated dose curve is then followed to within ± 30 % over the energy range 1 eV to 10 MeV.

The theory finally used for interpolation between experimental points was somewhat more refined. In an earlier report [11] a method of calculation of the neutron sensitivity of a polyethylene-moderated BF_3 counter has been described. A multigroup diffusion programme, developed for bulk-shielding calculations, has been utilized with astonishingly good results; this encouraged us to use a similar approach for improving the theoretical treatment of the case in hand. The original programme is not able to handle the perforated boron layer between the two polyethylene layers. Therefore the problem was considered as consisting of two separate cases. The output current in the forward direction of the first polyethylene layer was split into two parts. The part penetrating the boron layer was attenuated exponentially, and the sum of this part and the contribution leaking through the holes made up the input of the inner region of the counter. The calculation was carried out for $d_1 = 16$ mm, $d_2 = 30$, 50, 65 and 67 mm and 0, 3.0 and 11.3 % of holes in the boron layer. An approximate dose response of the counter could be expected if the values of the parameters were $d_1 = 16$ mm, $d_2 = 70 - 75$ mm and the percentage of holes in the boron layer = 10 %.

5. Measurement of neutron response

A counter arrangement of a design that is outlined in Fig. 2 was built for experimental investigations. The BF_3 counter that was used is shown in Fig. 3. The thickness of the inner polyethylene layer was made 16 mm. We started with no holes in the boron shield and it was arranged that the thickness of the outer polyethylene layer could be changed between d_2 = 30, 50 and 67 mm. In a first series of measurements we wanted to determine the neutron sensitivity as a function of energy for the three different values of d_2 with the neutrons incident at right angles to the axis of the counter.

Monoenergetic fast neutrons were produced by the charged particle reactions Li^7 (p, n) Be^7 and H^3(p, n)He^3 using the 5.5 MeV Van de Graaff machine at the AB Atomenergi Research Laboratory at Studsvik. The radioactive neutron sources Sb-Be and Pu-Be were used to carry out measurements at 35 keV and 4 MeV respectively. A thermal neutron source was prepared by placing a Sb-Be source in a cavity with 11 cm thick walls of polyethylene. The fast neutron fluence was measured by a Hanson-McKibben long counter. In the Van de Graaff experiments the long counter and the neutron dose counter were placed at the same distance, 170 cm, from the target and at the same angle, 40^o, but on opposite sides relative to the direction of the proton beam. The counters were exposed simultaneously. The pulses delivered by the counters were registered in scalers. The absolute long counter efficiency was determined by the use of calibrated Ra-Be and Pu-Be sources to an accuracy of $\pm$ 5 %. The relative long counter efficiency as a function of energy was taken from Allen et al. [12]. The flux density from the thermal neutron source was measured by a calibrated BF_3 counter.

With these monoenergetic neutron sources of known intensity we determined the sensitivity of the dose counter arrangement in counts per n/cm^2 at three different values of d_2. The results are shown in Fig. 4. The ideal rem response curve is given by the dashed line in the figure. It appears that a value of d_2 slightly less than 67 mm would give a sensitivity curve that follows the rem response curve fairly well in the region 35 keV to 5 MeV. These results were in reasonable agreement with the multigroup diffusion calculations. Below 35 keV, the theory predicts a sensitivity which is considerably lower than that required to obtain rem response, and at thermal energies there is practically no sensitivity because the neutron are effectively absorbed in the boron shield. In order to cope with this problem we drilled a number of holes in the boron shield, thus introducing a third parameter. This will obviously increase the response for thermal and epithermal neutrons. The sensitivities at thermal energy and at 35 keV (Sb-Be) were now measured as a function of the number of holes in the boron shield. The holes had a diameter of 7 mm and they were uniformly distributed. For 3.0 % and 8.3 % of the area of the boron shield covered with holes, the sensitivities were also determined at different values of d_2 around d_2 = 67 mm. It was then possible to select a combination of percentage of holes and thickness of the outer polyethylene layer, d_2, to obtain the correct sensitivities at thermal and 35 keV energies. A hole area of 11.3 % and d_2 = 65 mm were finally accepted and a counter with these data was manufactured. This is the counter that is actually shown in Fig. 2. A new calibration was made and the results are presented in Fig. 5. The absolute sensitivity of the neutron dose counter is 7.6 cps/mrem/h. The deviation from the ideal rem response is given by the numbers in the figure. The sensitivity measured with Sb-Be has been referred to a neutron energy of 40 keV. This will include the correction for the Sb-Be neutron spectrum and the detector response. In the region between thermal energy and 40 keV, where there are no experimental data, the shape of the response curve is taken from the multigroup diffusion theory. The measured sensitivities are estimated to be correct within ± 10 %. In the region around 5 keV the accuracy is expected to be ± 30 %.

The response curve in Fig. 5 should be obtained for any BF_3 counter or other thermal neutron detector that may be inserted into the shield. The dose sensitivity can be estimated by use of the relation

$$D = 0.82 \times N$$

where D is the dose sensitivity in cps/mrem/h and N is the thermal neutron sensitivity in count/n/cm^2 of the thermal neutron detector itself. The shield can be lengthened to make it possible to use longer BF_3 tubes without serious effect on the response curve. If the length of the shield is reduced it will affect the response curve. This is discussed in greater detail in the next chapter.

If one wants to make use of BF_3 tubes of diameters above 30 mm it will probably be necessary to change the design of the shield to some extent. The values of d_1 and d_2 that are given in Fig. 2, 16 mm and 65 mm, mean that the cylinders consist of 16 and 65 layers, respectively, of 1 mm thick polyethylene. Because the cylinders do not fit perfectly into each other and the layers are not wrapped absolutely tightly, the total diameter of the counter, 207 mm, has been slightly larger than the sum of the thicknesses of the different layers and the diameter of the BF_3 tube. The real dimensions are important because it is both the geometric extension and the mass of the material around the thermal neutron detector that affect the response.

The dose counter we have described has an approximately correct rem response over the energy range 0.025 eV to 15 MeV. It is characterized by high sensitivity, flat bias plateau and very good discrimination against gamma radiation. The counter is unaffected up to 100 r/h from Co^{60}. The directional dependence (Fig. 6) and the weight, 15 kg, make the counter most suitable for use in installed monitors, for example around reactors and accelerators.

6. A neutron rem counter for a portable survey instrument

A neutron dosimeter for radiation protection survey should be nondirectional in response and it should be easy to handle. The dose counter we have presented has not very good properties in these two respects, but if the length of the counter is reduced, improved conditions could be expected. After experiments with different arrangements the construction shown in Fig. 7 was finally adopted. This was considered an optimum solution to the problem of getting high sensitivity, reasonable weight and good directional properties. We call the new counter the Neutron Rem Counter, NRC III. Several modifications have been introduced in the design. The BF_3 tube has a sensitive length of 60 mm. The thickness of the outer polyethylene disks at the ends is reduced to 46 mm to increase the sensitivity for neutrons incident parallel to the axis of the counter. The total length of the counter is further shortened by having the pump tube entering into the shield. The hole area in the boron shield is 22 % to get the best fit to the ideal rem response curve. It will compensate for the higher thermal neutron leakage in the smaller volume and slightly larger diameter of NRC III.

A final calibration was carried out. The energy region 35 keV to 4 MeV was covered by neutrons from Sb-Be, $Li^7(p, n)Be^7$ and Pu-Be and a long counter as a flux density monitor. The R2-0 reactor at Studsvik was used for the thermal neutron calibration. The measurements were made in a cavity in a large volume of heavy water which was placed close to the tank of the reactor. The thermal neutron fluence was determined by the Mn-foil activation technique. The response of NRC III for thermal neutrons was obtained as the difference in counts at irradiation with the counter bare and placed in a container of 1 mm thick cadmium. The sensitivity at 14.7 MeV was measured by neutrons from the $T(d, n)He^4$ reaction with the 300 kV accelerator at the National Defence Laboratory at Ursvik. The flux density was determined by a stilbene scintillation recoil spectrometer.

The results of the calibration are shown in Fig. 8. The response was measured both for neutrons incident at right angles and parallel to the axis of the counter. The absolute sensitivity is 2.4 cps/mrem/h. The deviation from ideal rem response is indicated in Fig. 8 and in Table II. The counter should be useful as a dosimeter in the energy range 0.025 eV to 15 MeV. The angular responses for Pu-Be and

Sb-Be neutrons are shown in Fig. 9. The uniformity is considerably improved and it should be sufficient for most applications. The weight of the counter is 8.5 kg. Fig. 10 shows a photo of NRC III and Fig. 11 displays the details of which it is made up.

The formula for calculation of the dose sensitivity gives a good result even for the shorter shield. The thermal neutron sensitivity of the BF_3 tube in NRC III is 2.8 counts/n/cm^2. The formula then gives $0.82 \cdot 2.8 = 2.3$ cps/mrem/h, which is close to the measured value 2.4 cps/mrem/h.

NRC III is used in a portable instrument for radiation protection survey (Fig. 12). The transistor electronics is mounted directly on one end of the counter. The instrument has a logarithmic display over the range 0.03 - 100 mrem/h. With correct bias setting there is no influence from gamma ray exposure up to 200 r/h of Co^{60}.

The neutron rem counter NRC III is manufactured commercially by 20th Century Electronic Ltd., Croydon, England.

Acknowledgements

The authors are indebted to M. Leimdörfer for valuable discussion regarding the theoretical aspects of this work, and to R. Fräki for performing the numerical calculations. We wish to express thanks to S. Malmskog for his contribution in the preliminary experiments, to B. Brunfelter for his assistance and advice in the experiments at the National Defence Laboratory at Ursvik, and to E. Lindahl for building the counters and for his assistance in the calibrations.

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TABLE I

MAXIMUM PERMISSIBLE NEUTRON FLUX
From NBS Handbook 63 (1957)

Neutron energy (MeV)	100 mrem/40 h ($n/cm^2 sec$)
Thermal	670
0.0001	500
0.005	570
0.02	280
0.1	80
0.5	30
1.0	18
2.5	20
5.0	18
7.5	17
10.0	17
10.0 - 30.0	10

TABLE II

Response of Neutron Rem Counter NRC III

Neutron energy	Sensitivity of NRC III relative to an ideal rem response of 2.4 cps/mrem/h	
MeV	at right angles to the axis	parallel to the axis
15	0.60 ± 10 %	0.65 ± 10 %
10	0.78 "	0.78 "
7.5	0.82 "	0.80 "
5	0.98 "	0.89 "
2.5	1.20 "	0.97 "
1	1.00 "	0.73 "
0.5	1.11 "	0.77 "
0.1	1.06 "	0.93 "
0.040	1.19 "	1.12 "
0.020	1.27 ± 20 %	1.27 ± 20 %
0.005	1.60 ± 30 %	1.60 ± 30 %
0.0001	1.12 "	1.12 "
Thermal	0.94 ± 20 %	0.94 ± 20 %

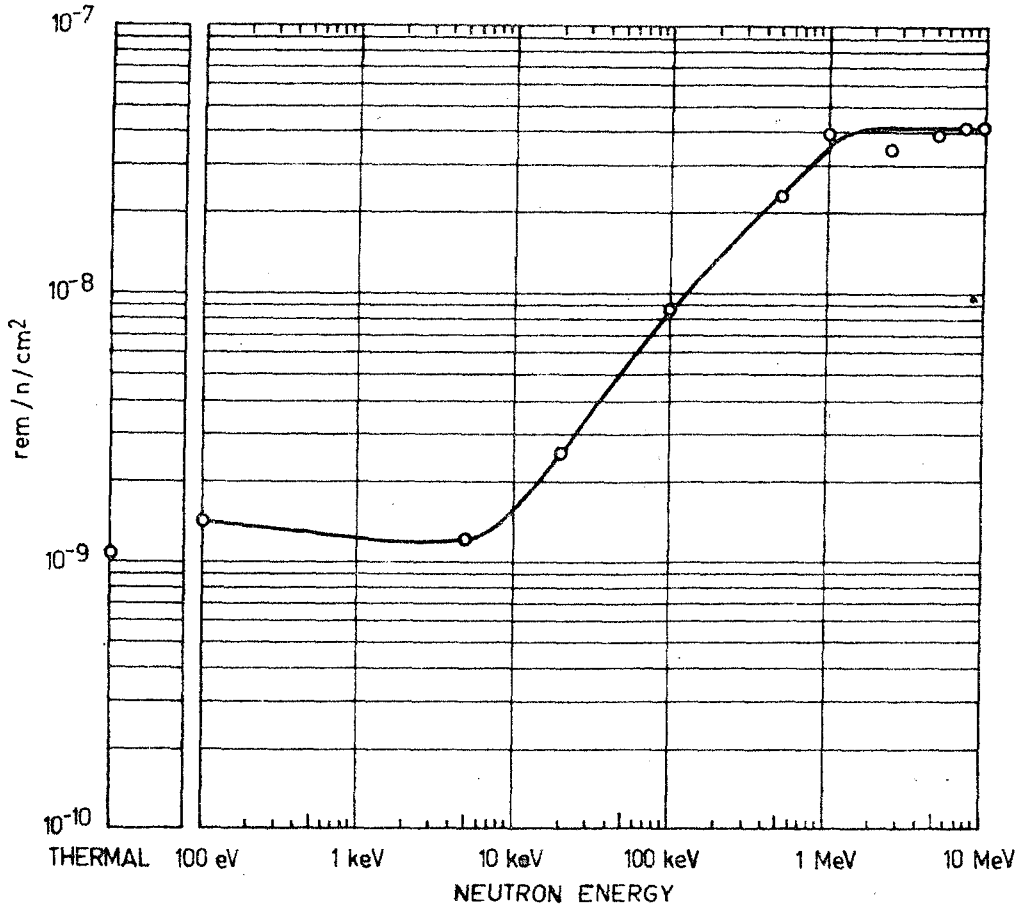

Fig. 1. Neutron dose in rem/n/cm^2 as a function of neutron energy.
From NBS Handbook 63, (1957).

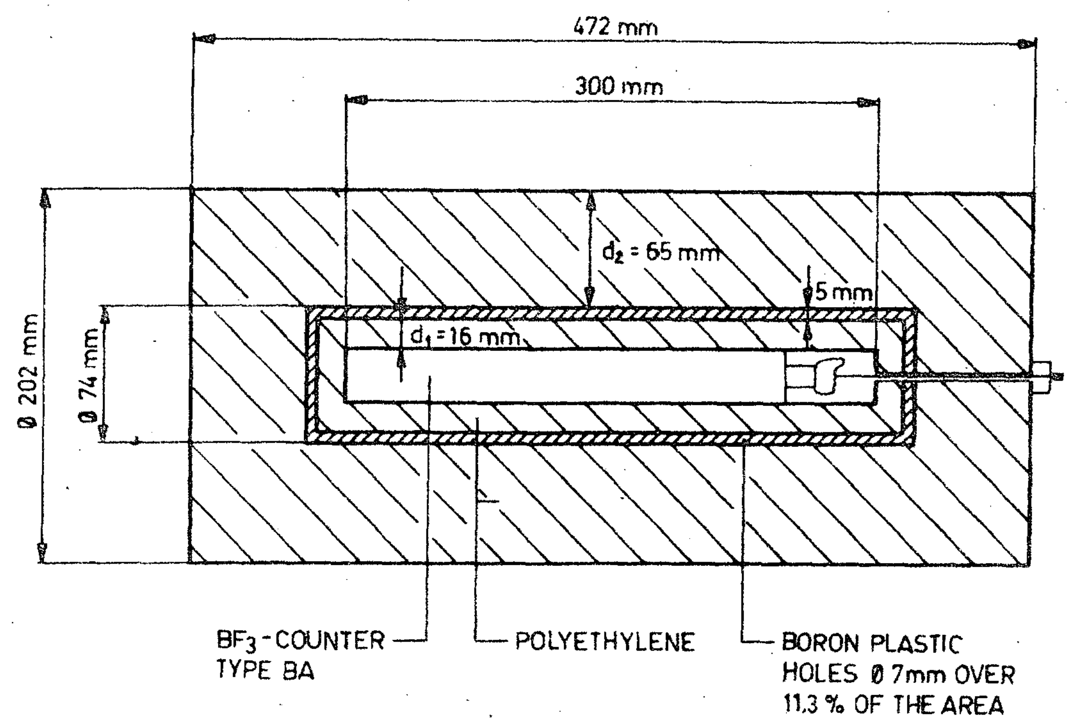

Fig. 2. Construction of neutron rem counter.

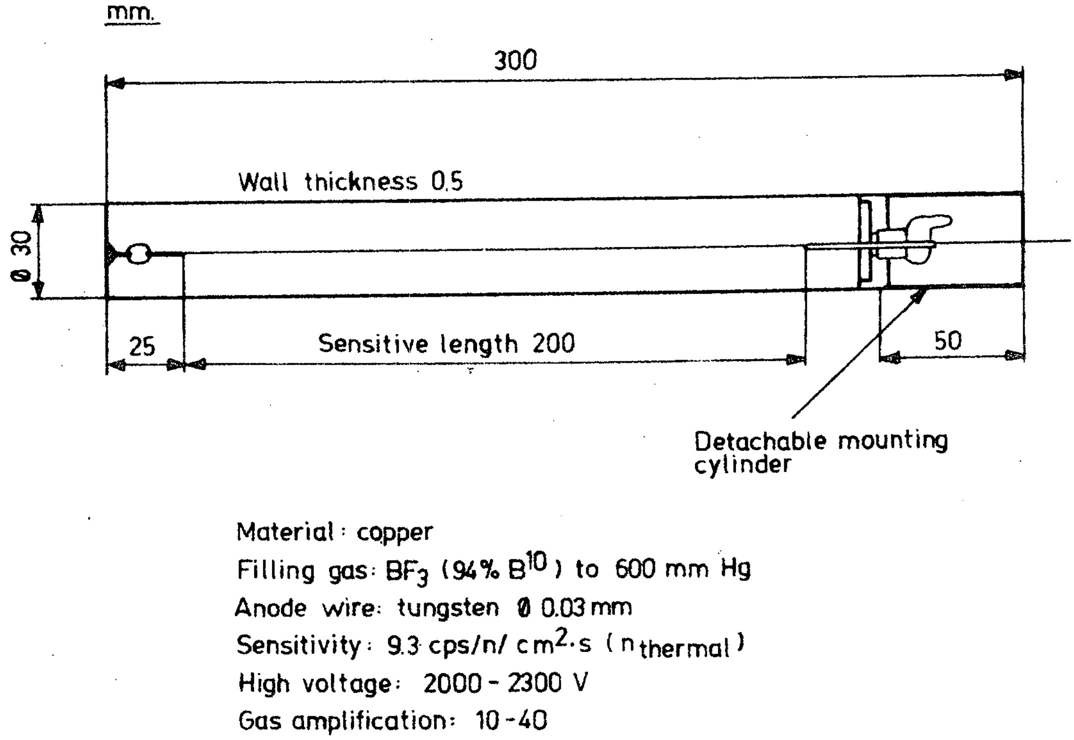

Fig. 3. BF_3-counter type BA.

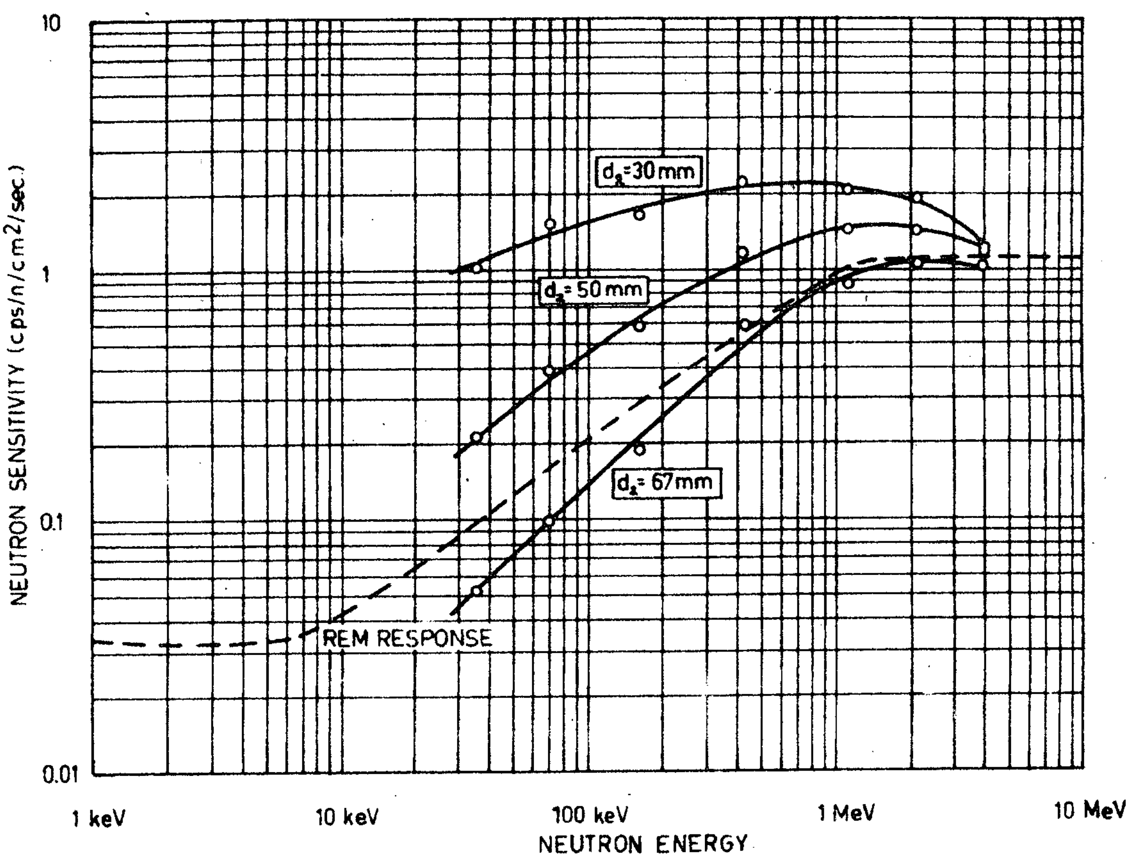

Fig. 4. Neutron sensitivity as a function of energy for different thicknesses (d_2) of the outer polyethylene layer.

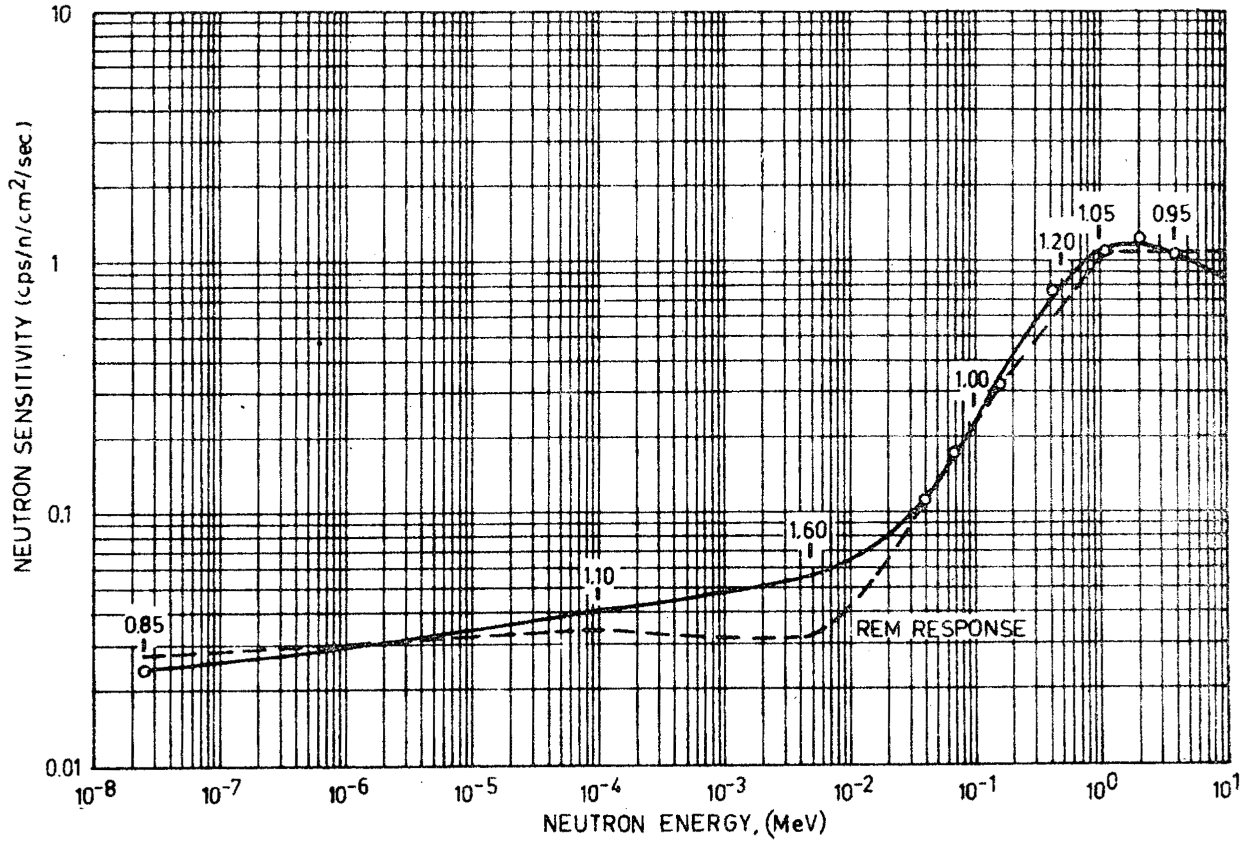

Fig. 5. Neutron sensitivity as a function of energy for neutron rem counter shown in Fig. 2. The figures give the sensitivity relative to an ideal rem response of 7.6 cps/mrem/h.

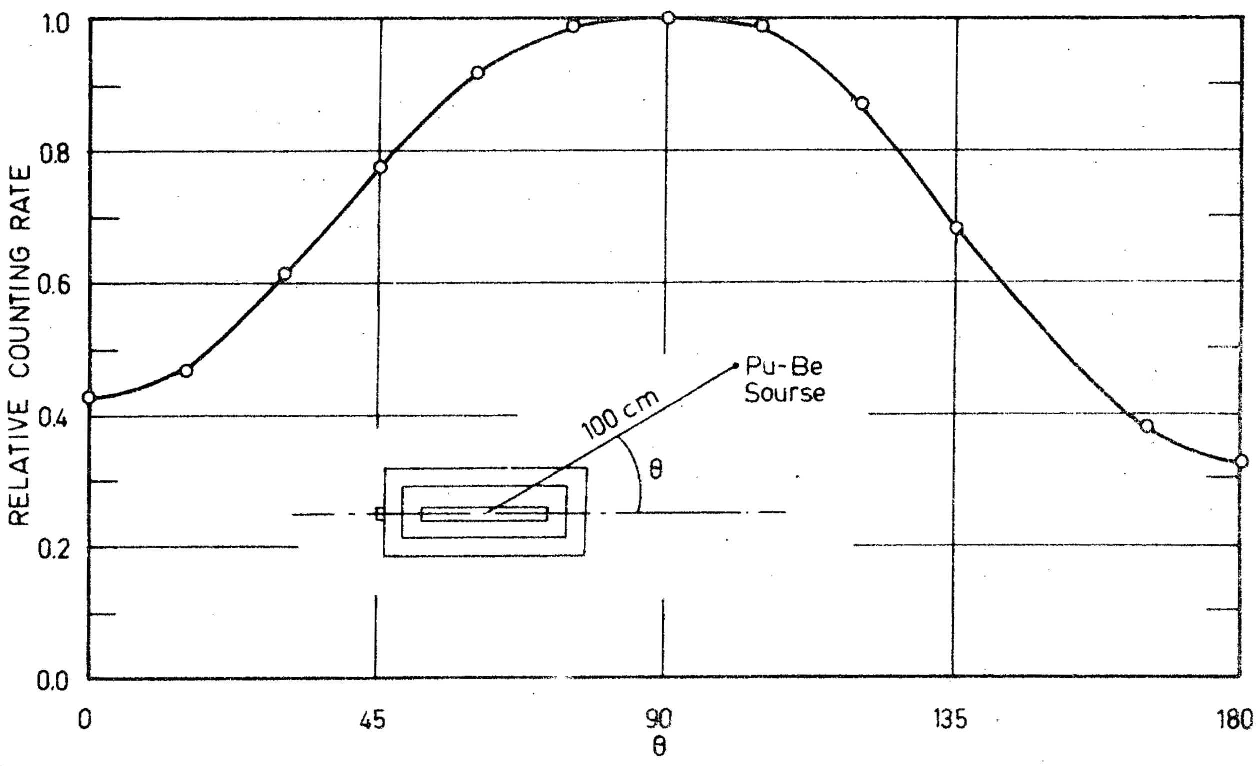

Fig. 6. Directional dependence of the neutron rem counter for Pu-Be neutrons.

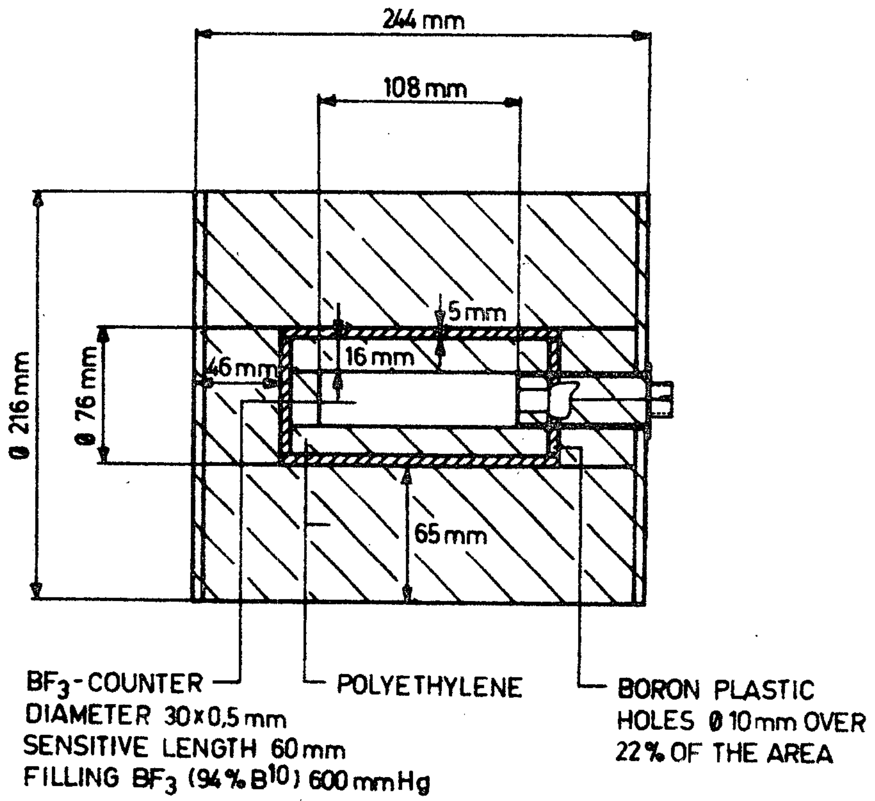

Fig. 7. Neutron rem counter NRC III.

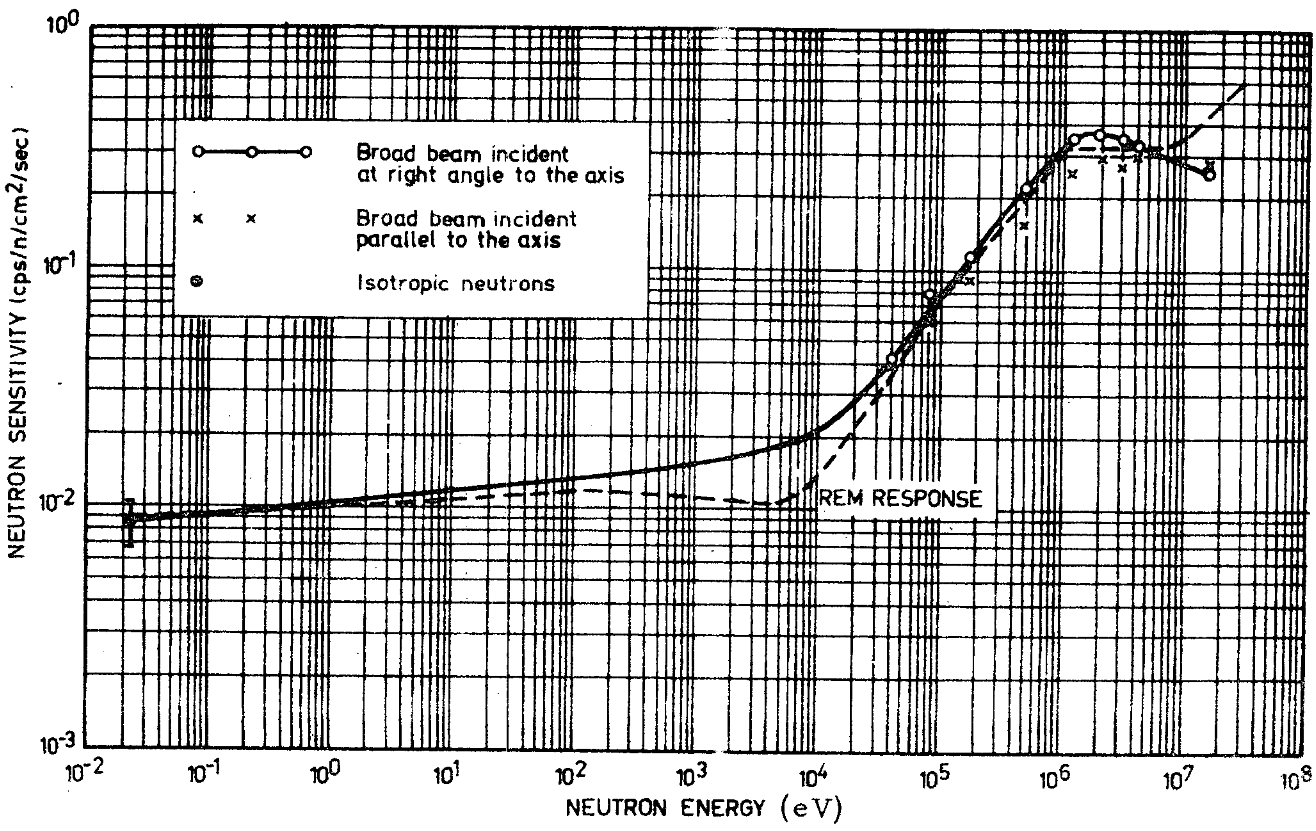

Fig. 8. Neutron sensitivity as a function of energy for the neutron rem counter NRC III.

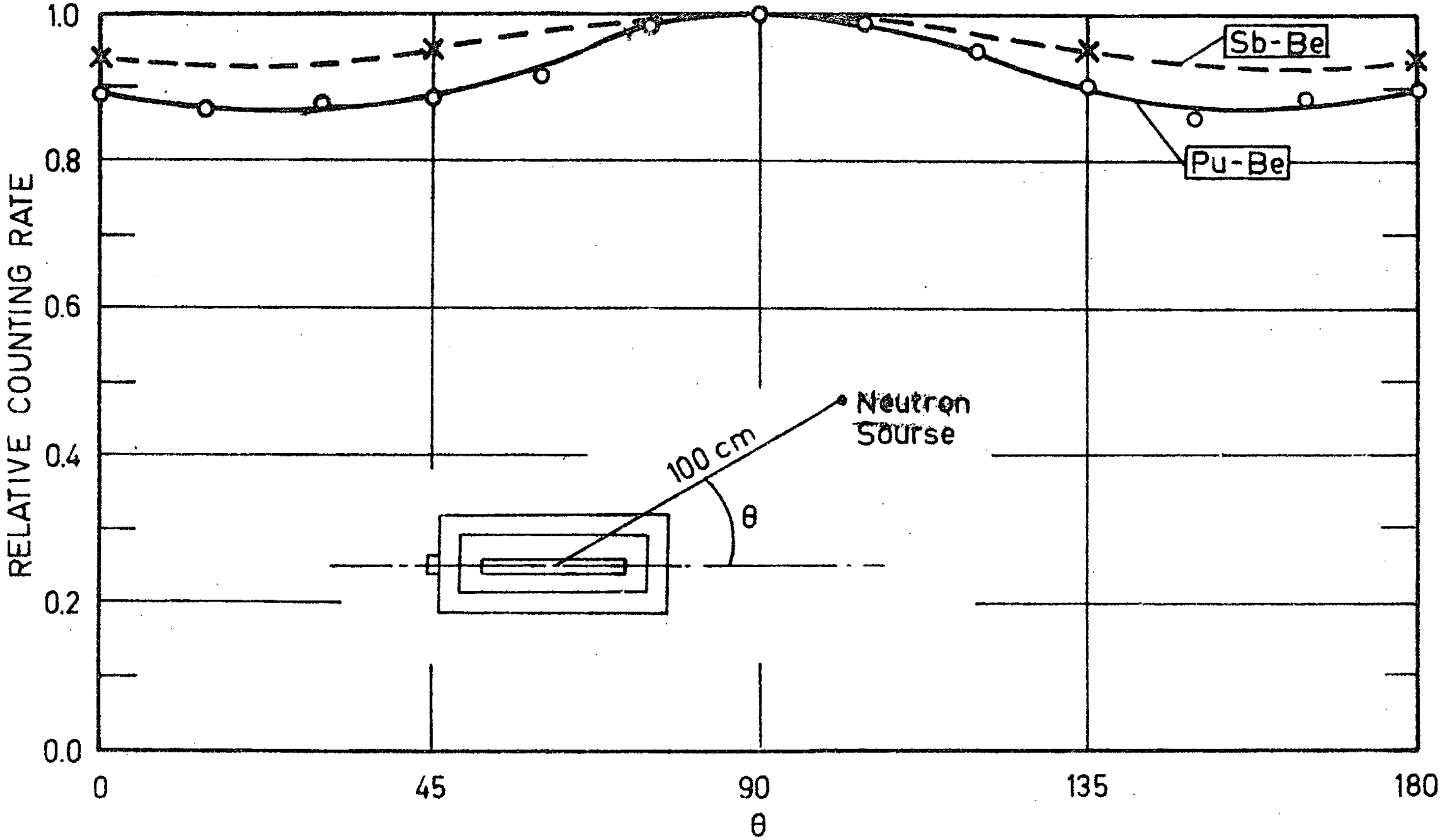

Fig. 9. Directional dependence of the neutron rem counter NRC III for Sb-Be and Pu-Be neutrons.

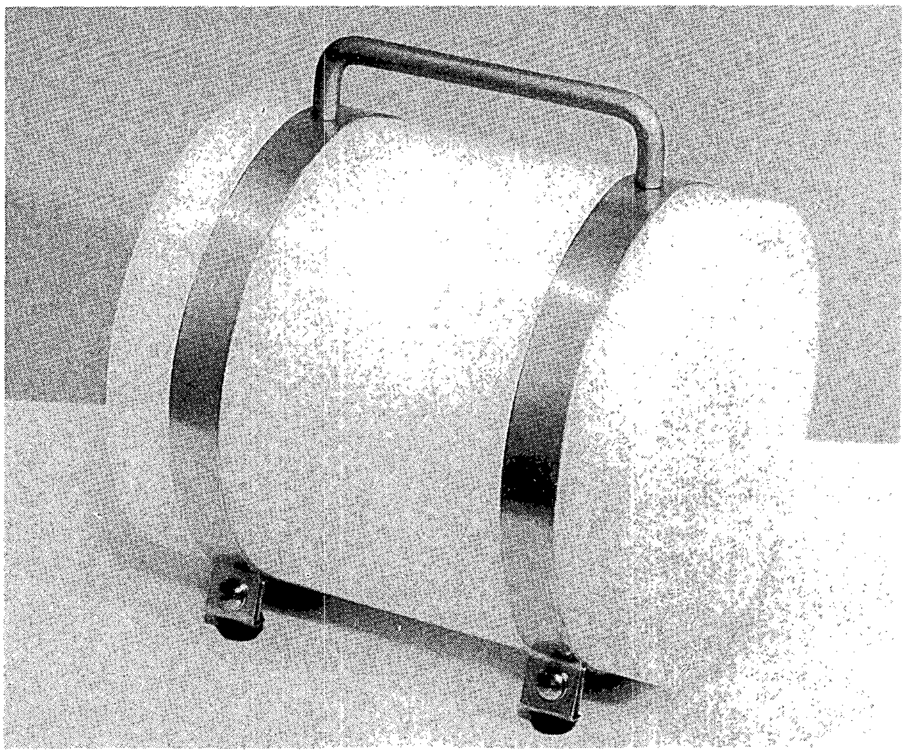

Fig. 10. Neutron Rem Counter NRC III

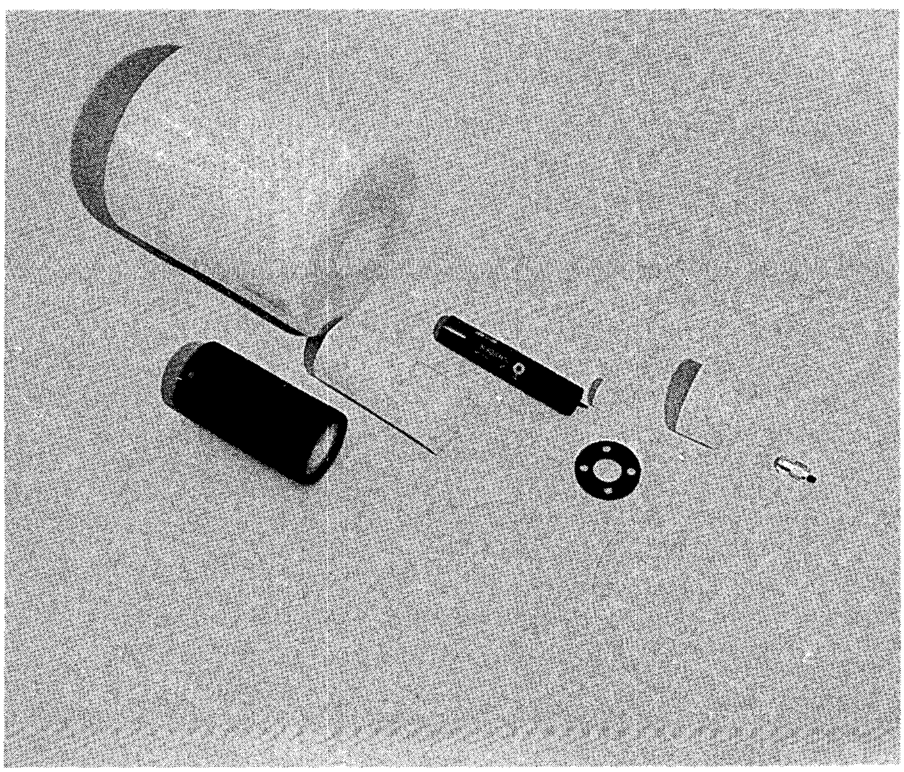

Fig. 11. Neutron Rem Counter NRC III. Details

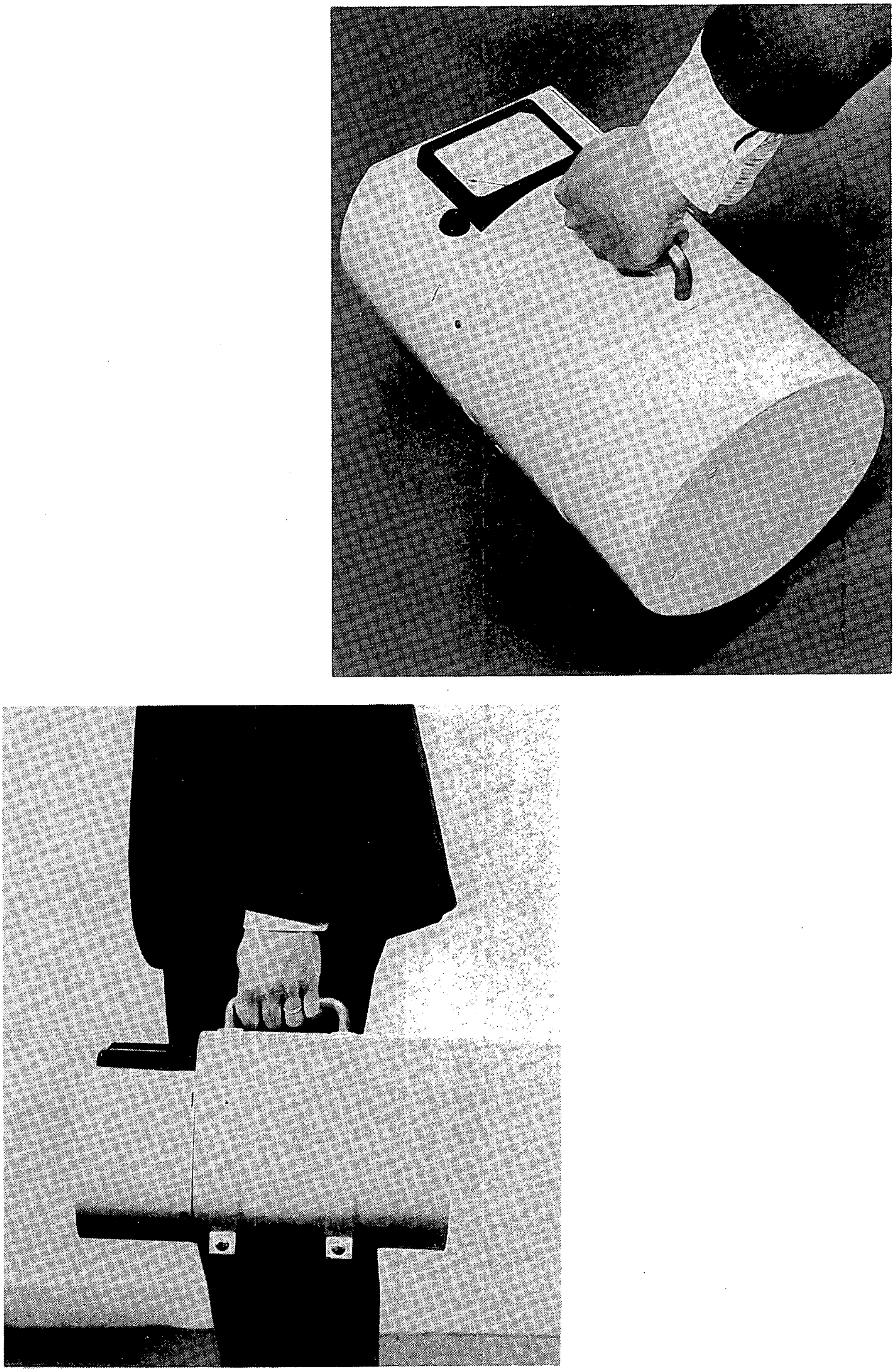

Fig. 12. NRC III in a portable neutron dosimeter

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