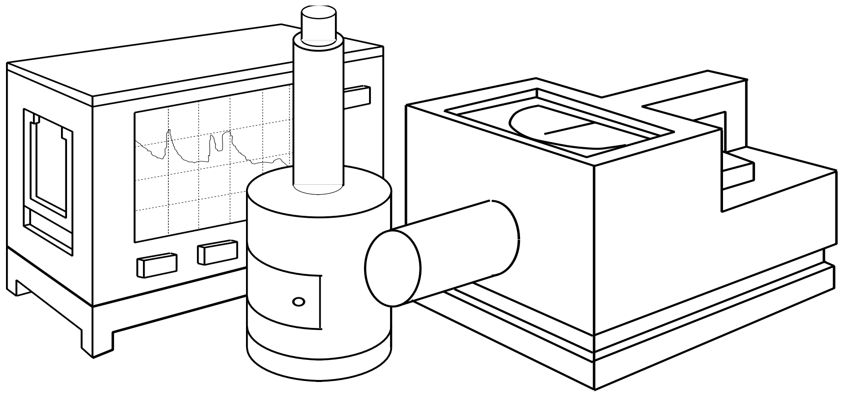
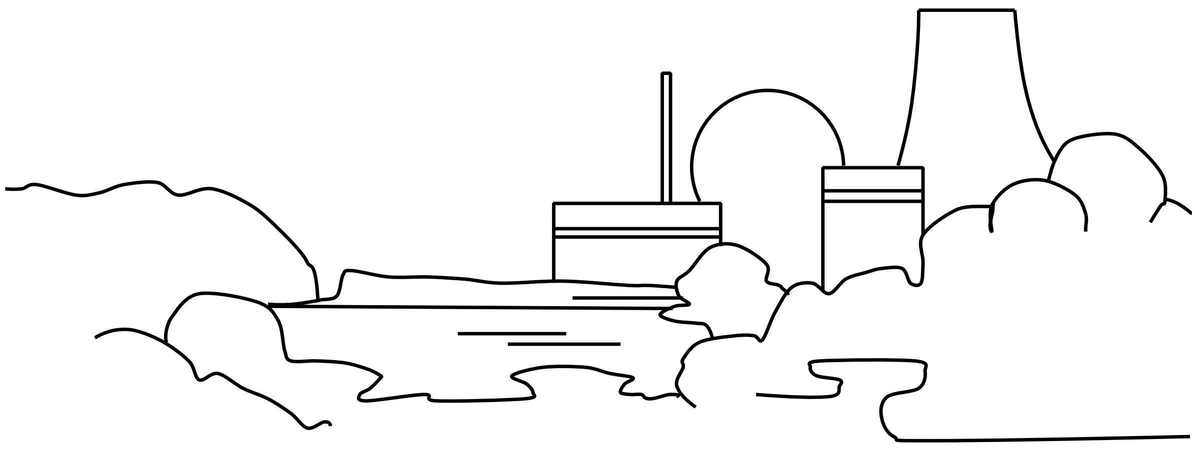
TECHNICAL UNIVERSITY DRESDEN
Institute of Power Engineering
Training Reactor



Reactor Training Course

Experiment

"Gamma Dosimetry and Dose Rate Determination"



Instruction for
Experiment "Gamma Dosimetry and Dose Rate Determination"

Content:

Figures:

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1. Motivation

The experiment aims on familiarising with the methods of calibrating different detectors for determination of the dose and the dose rate. Furthermore, the dose rate and the activity in the vicinity of an enclosed source of ionising radiation (Cs-137) will be determined taking into account the background radiation and the measurement accuracy. Additionally, the experiment focuses on the determination of the dose rate of a shielded source as well as on the calculation of the required thickness of the shielding protection layer for meeting the permissible maximum dose rate. The dose rate at the open reactor channel of the AKR-2 will be measured.

2. Theoretical Background

2.1. Properties of Ionising Radiation and Interactions of Gamma Radiation

The radiation emitted along with the transformation of radionuclides is the outward characteristic of radioactivity. The radiation is categorised as α-radiation (He-4 nuclei), $ß^+$- and $ß^-$-radiation (positrons and electrons, respectively) and γ-radiation. As a by-product of such transformations or as a result of them conversion electrons and X-rays can be observed. In some cases of nuclear reactions, also neutrons are released that will interact with matter, too.

Gamma radiation is high-energy electromagnetic radiation that are emitted in form of photons with a discrete energy distribution along with a transition of an atomic nucleus from an excited state into a lower or the ground state. Simply speaking, the origin of gamma radiation corresponds to the origin of visible light, which, in contrast, is the result of transitions between states of the atomic shell.

The interaction of gamma radiation with matter differs significantly from that of charged particles with matter. Whereas α- and ß-radiation lose energy by ionisation or radiation slowing-down processes, γ-rays lose energy mainly by the photo effect, Compton scattering, and pair production.

The attenuation of gamma radiation, i.e. the intensity I after penetration of a layer with the thickness x consisting of absorber material relative to the intensity I_0 without any absorber, can be described via the attenuation equation $I = I_0 \cdot \exp(-\mu x)$ with μ being the linear attenuation coefficient. This coefficient is the arithmetic sum of those coefficients which quantify the three independently appearing kinds of absorption interactions of gamma radiation, i.e. $\mu = \mu_{co} + \mu_{Ph} + \mu_{Pa}$ (Fig. 1).

The quantity μ_{co} represents the Compton scattering (also called: Compton effect), which is the transfer of energy from a photon to an electron of the atomic shell by collision. The gamma quant is redirected, i.e. scattered. This process goes along with a defined loss in energy dependent on the angle (being between 0 ... 180°). The emitted electron is called a Compton electron. Compton electrons have a continuous energy spectrum.

The quantity μ_{Ph} represents the photo effect, which is the transfer of the entire energy of a photon to an electron of the atomic shell and hence, the photon vanishes. The electron (photo electron) is emitted out of the atomic shell with the total energy of the extinguished photon minus the ionisation energy.

The quantity μ_{Pa} represents pair production, which is the transformation of a photon to an electron-positron pair. This effect appears only at photon energies higher than 1.02 MeV (two times the rest mass of an electron = $2 \cdot 0.51$ MeV).

The percentage of each of these effects depends essentially on the energy of the photon and the atomic number of the absorber (Fig. 1).

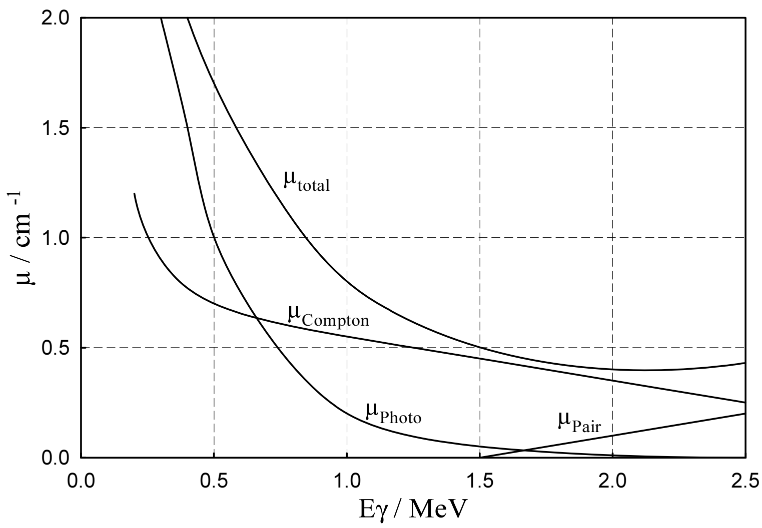


Fig. 1, Composition of the attenuation coefficient μ of γ-radiation in lead

The dependence on the atomic number is:

for the photo effect	proportional to Z^4 ... Z^5
for Compton scattering	proportional to Z
for pair production	proportional to Z^2

From these relations, the suitability of a special absorber material for certain gamma energies can be assessed. The choice for a particular material is determined not only by radiation protection reasons but also by economic, technical and low-weight criteria (e.g. in case of protective clothing).

The electrons which are released by each of the three processes carrying different amounts of energy cause ionisation of the matter which they penetrate. This phenomenon is the basis of devices for the determination of the γ-energy. It is also the reason for the biological effectiveness of γ-radiation.

2.2. Detection of Ionising Radiation

Radiation measuring devices recognise the existence of ionising radiation and determine the kind, the energy, the intensity, and the direction of the radiation. For this purpose, a variety of measurement devices have been developed. The training experiment at AKR-2 focuses on dose and dose rate measurement devices. The ionisation chamber is an important detector for this purpose. It has a simple concept, can be applied in various geometries and dimensions, and can be adjusted according to a particular radiation in a wide range.

Design and method of operation of an ionisation chamber:
In the simplest case, an ionisation chamber is a cylinder filled with air or an inert gas. It is equipped with two insulated plate electrodes which are connected to a power supply over a resistance R (Fig. 2).

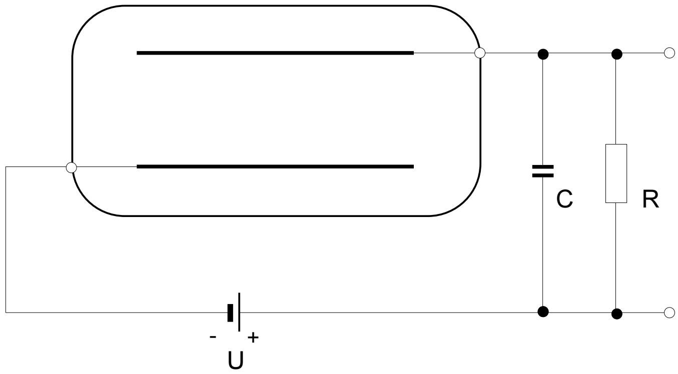


Fig. 2, Design of an ionisation chamber

Inside the chamber, a charged particle generates pairs of ions, whose number per unit length depends on the energy and the kind of the particle, on the type of the gas, and on the pressure of the gas. Due to the applied electric field these charge carriers migrate to the electrodes and cause a chamber current i, which increases for increasing chamber voltage. For large numbers of particles, an average chamber current appears whose magnitude is a measure for the number of incoming particles. Such an ionisation chamber that works on the current is used especially for detection of beta and gamma radiation.

2.3. Quantities and Units of Dosimetry

Although the unit of the **energy dose D** (unit: 1 Gray = 1Gy) was introduced respecting the biological effectiveness, exposures of a certain energy dose by various kinds of radiation can have widely differing biological effects. This property of the radiation can be quantified by the **radiation weighting factor w_R** (Tab. 1). The values of the radiation weighting factor w_R depend on the type and quality of the external radiation field or on the type and quality of the radiation emitted from an incorporated radionuclide. In this context, the quantity of the **dose equivalent H** is introduced, whose unit is Sievert [Sv].

Type and energy range	Radiation weighting factor w_R
X-, ß- and γ-radiation	1
Protons	2
Neutrons (energy dependent)	ca. 2.5 - 20
α-radiation, fission products	20

Tab. 1, Radiation weighting factor w_R (StrlSchV Annex 18, Part C)

The dose equivalent H and the energy dose D are related as follows:

$$H = D \cdot w_R$$

Because for X-, γ- and ß-radiation the radiation weighting factor is $w_R = 1$, for these (and only these!) kinds of radiation is:

$$1\ \text{Gy} = 1\ \text{Sv}$$

All of these relations are independent on the time of irradiation.

The **effective dose E** is defined as the sum over all organs T

$$E = \Sigma\,(w_T \cdot H_T)$$

with H_T - organ absorbed equivalent dose
w_T - tissue weighting factor for the respective organ or tissue T

The values of the tissue weighting factors for particular organs or tissues are given in Tab. 2.

tissues or organs	tissue weighting factors w_T
red bone marrow, colon, lung, stomach, chest	each 0,12
gonads	0.08
bladder, oesophagus, liver, thyroid gland	each 0.04
skin, bone surface, brain, salivary gland	each 0.01
other organs or tissues (together)	0.12

Tab. 2, Tissue weighting factors w_T (StrlSchV Annex 18, Part C)

The most important quantities in radiation protection are summarised in Tab. 3.

The application of scientifically reasonable measures in radiation protection requires profound knowledge about the interaction between radiation and biological tissue. For the handling of sources of ionising radiation, the Radiation Protection Act (German: Strahlenschutzgesetz, StrlSchG) and the Radiation Protection Ordinance (German: Strahlenschutzverordnung, StrlSchV) define maximum doses for certain time periods. The values of these dose limits vary for various organs, parts of the body, and for whole body, as well as for occupationally radiation exposed people, particular groups of the population, and the whole population. Moreover, as a general principle the radiation exposure has to be kept **a**s **l**ow **a**s **r**easonably **a**chievable (ALARA principle). This principle has to be kept also for radiation exposures below the legal limits.

According to the German Radiation Protection Act the individual personal exposure to radiation per one calendar year must not exceed the following limits (in mSv):

Values in mSv	Occupationally exposed person		Individual members of the general public (due to the work with radioactive substances)
	Category A	Category B	
Effective dose (whole body)	20 [1)]	6	1 [2)] or 0.3 [3)], resp.
Eye lens	20	15	15
Skin, hands, feet, forearms	500	150	50

1) exceptionally an effective dose of 50 mSv for a single year can be permitted, where 100 mSv for five consecutive years may not be exceeded, in maximum 400 mSv during the whole professional life

2) direct irradiation **and** discharge of radioactive substances (air and water)

3) only discharge of radioactive substances (air and water)

Moreover:

- people under 18 years	1 mSv/a	
but: trainees and students	6 mSv/a	insofar as this is required for the purpose of their training
- women of child-bearing age	2 mSv/month	at the uterus
- during pregnancy altogether	1 mSv	due to the occupation of the mother

Through the Radiation Protection Ordinance, the legislative body requires that the rooms being used for handling sources of ionising radiation have to be classified into radiation protection areas. These are

a) supervised areas,
b) controlled areas,
c) exclusion areas

Fig. 3 shows the permissible thresholds of the various radiation protection areas.

Value	Definition	Legal unit	Former unit	Conversion
Activity	number of radioactive transformations per time unit	Bequerel 1 Bq = 1 s^{-1}	Curie	1 Ci = $3.7 \cdot 10^{10}$ Bq
Energy dose	total amount of absorbed energy per mass element divided by its mass	Gray 1 Gy = 1 J / kg	rad	1 rad = 0.01 Gy
Dose equivalent	energy dose multiplied by the radiation weighting factor of the respective kind of radiation	Sievert 1 Sv	rem	1 rem = 0.01 Sv
Energy dose rate	energy dose per time unit	Gy / s	rad / s	1 rad / s = 0.01 Gy / s
Dose equivalent rate	dose equivalent per time unit	Sv / s	rem / s	1 rem / s = 0.01 Sv / s

Tab. 3, Dosimetric quantities and units

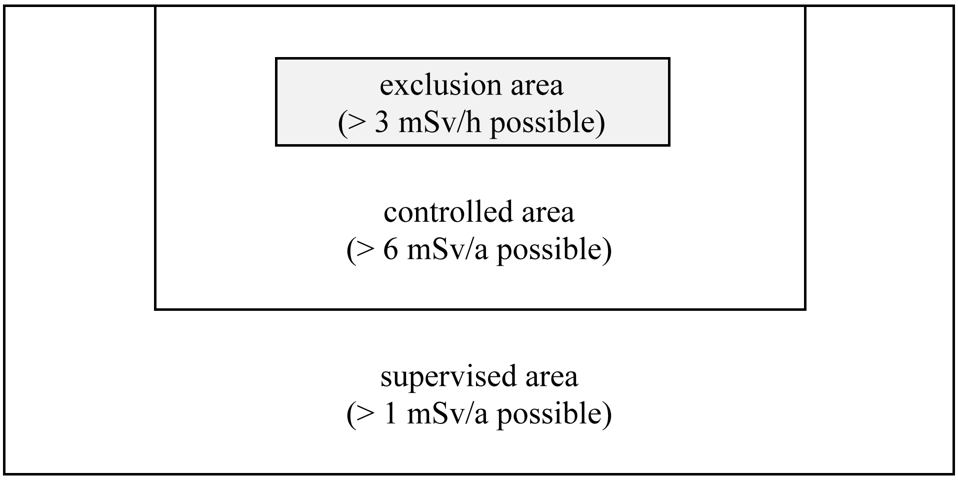


Fig. 3, Classification and legal limit values of radiation protection areas

The definitions of the limits of controlled areas and exclusion areas base on a 40-hours work-week with 50 workweeks per calendar year, as far as no other causal details about the duration of stay are given.

3. Procedure of the Experiment

For handling of radioactive substances or of sources of ionising radiation, the following principles of radiation protection need to be kept with care!

1. No one is allowed to be subject of uncontrolled radiation exposure (personal dosimetry control).

2. The group of people being exposed to radioactivity needs to be kept as small as possible.

3. The duration of exposure needs be kept as short as possible (work organisation).

4. The distance from the radiation source needs to be kept as large as possible (manipulators, pincers).

5. Use shieldings as far as possible.

6. Tidiness and cleanness

7. Exceptional events have to be reported to the radiation protection officer.

3.1. Commissioning and Calibration of the Dosimeter Thermo FH40G

1. Switch-on the device.
2. Wait until the end of the self-test.
3. The device is ready in case of no error messages.

3.2. Commissioning and Calibration of the Dosimeter Berthold LB 133-1

Screw the counter tube into the face area of the device until the end stop is reached. Press the large red push-button. At normal natural background radiation, the device indicates a dose rate of (0.1 ... 0.2) µSv/h.
Then, the battery status has to be checked. For this purpose, press the respective button below the handle bar, the pointer should stand in the area with black background. Fully charged batteries allow service for about 8 hours.
For changing the measuring range use the respective buttons below the handle bar. By pressing the button "Zeitkonstante" the standard time constant can be increased by a factor of 10 from 2 s to 20 s. For locating radiation fields and for overview measurements, the smaller time constant is recommended, whereas the large time constant is used for more precise measurements and for the measuring in the most sensitive range "x1".
The device has an adjustable alarm threshold. When exceeding the previously set threshold, the device gives an acoustic alert over a time of about 4 s. The alert is reactivated only after the display value has fallen below the threshold. Though, the indicator lamp blinks as long as the threshold is exceeded. Note that the value for stopping the alert is about 20 % lower than the threshold's scale value (hysteresis). The acoustic alert can neither be deactivated nor be changed in its volume.

3.3. Commissioning and Calibration of the Dosimeter STEP RGD 27091

1. Switch-on the device.
2. A check of the battery status is not necessary, because the device performs a self-test and automatically reports a low battery status (optically).
3. Put the switch to zero adjustment ("Nullabgleich").
4. Put the switch "Messbereich" to 20 and set the zero-balance to a display value <0.5 using a potentiometer.

3.4. Setup of the Experiment

The setup of the experiment is given in Fig. 4.

Fig. 4, Setup of the experiment

After calibrating the devices, the three detectors are placed at the intended spot on the measurement table. Only shortly before the measurement, the source is taken out of the shielded lead container (use pincers!) and mounted onto the slide (the distance between slide and measurement table is about 3 m).

For the shielding investigations the shielding material has to be placed directly in front of the source.

3.4.1. Measurement of Dose Rate in Various Distances from the Radiation Source

The measurements are carried out with all three measuring devices. The radiation source has to be placed in those distances from the detectors that are given in the example protocol (Tab. 4). The opening of the source has to point into the direction of the detectors.
The measurements start with the largest distance. The whole series of measurements will be repeated afterwards. For positioning of the measurement devices high accuracy is required.

<u>Getting measured values:</u>

Thermo FH40G:	direct reading on the display
LB 133-1:	choose the larger time constant and the largest measuring range, wait until the setting time is over (about 30 s)
STEP RGD 27091:	direct reading on the display

3.4.2. Measurement of Dose Rate behind Radiation Shielding

The radiation attenuation coefficient μ has to be determined for the shielding materials heavy concrete, light concrete, and lead. Its impact has to be discussed with relation to radiation protection. For this purpose, the shielding of the respective material has to be placed between the source and the detector. From the measured values, the attenuation coefficient can be calculated (Tab. 5).

3.5. Measurement of Dose Rate at the open Reactor Channel

1. Mount the spherical chamber on the device
2. Switch-on the dosimeter
3. Mount the spherical chamber in such a way that the middle of the chamber is positioned according to the distance tick marks
4. Insert the measurement stick as far as possible into the reactor channel (attention: Do not stand directly in front of the channel opening!)
5. Pull-out the stick stepwise according to the respective tick marks
6. The distances which have to be set for measurement are given in the example protocol (Tab. 6)

4. Evaluation of Measuring Results

1. Plot the function P_x in dependence on distance r using the values measured with the dosimeters (use log-log scales!)

2. Calculate the function $P_x = K \cdot \frac{A}{r^2}$ from the given activity of the source Cs-137 and compare the result with the measured values

   with P_x - dose rate [mSv/h] due to a point-like gamma emitter in air

   A - activity [Bq]
   A (Febr. 1993) = $0.26 \cdot 10^9$ Bq ($T_{1/2}$ = 30 a)

   K - dose constant [$mSv \cdot m^2$ / ($h \cdot GBq$)]
   for Cs-137 is K = 0.0925 $mSv \cdot m^2$ / ($h \cdot GBq$)

   r - distance to the source [m]

3. Discuss the appropriateness of each of the detectors and also the reasons for the differences in the results

   Some specifications of the measurement devices:
   - **STEP RGD 27091:**
     Indication error < 5 %, the detector is an ionisation chamber
   - **Thermo FH 40G** and **Berthold LB 133-1:**
     Indication error < 15 %, very low energy dependence, the detector is an ionisation chamber

4. Calculate the linear attenuation coefficient from the values of the dosimeters for all three shielding layer thicknesses using $P_x = B_D \cdot P_{x0} \cdot e^{-\mu x}$ with P_{x0} being the dose rate at the place of measurement without any shielding

5. Calculate the half-value thickness $x_{1/2}$ [cm] with $x_{1/2} = \frac{\ln 2}{\mu}$

6. Calculate the mass-attenuation coefficient μ' [cm²/g] using $\mu' = \frac{\mu}{\rho}$

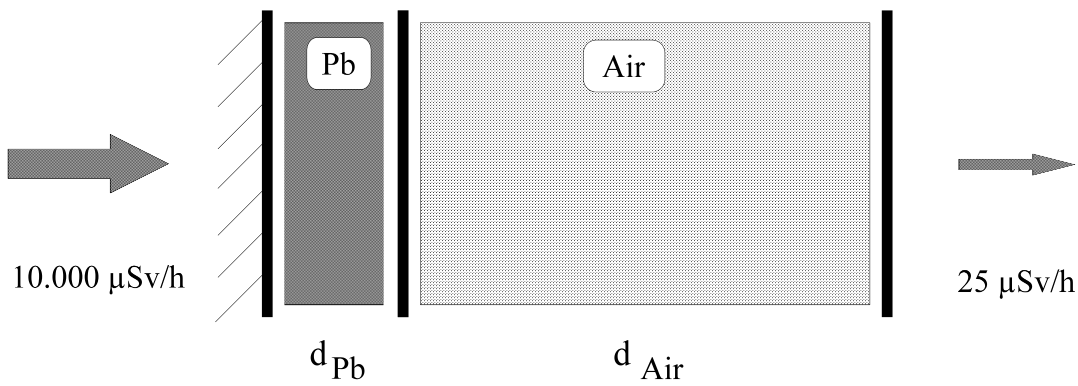
   with $\rho = 0.6$ g/cm³ (light concrete)
   $\rho = 2.5$ g/cm³ (heavy concrete)
   $\rho = 11.7$ g/cm³ (lead)

Remarks regarding the dose build-up factor B_D:
The dose build-up factor is defined as the ratio between the sum of the dose rate of all photons (scattered and non-scattered) and the non-scattered photons.
It depends on the energy of the photons, the shielding layer thickness, the radiation protection material, and the measurement geometry. By means of the build-up factor, calculations become possible also for wide spreaded beams as well as for extended or thick absorbers via the attenuation factor.
For the shielding measurements in the experiment, the dose build-up factor B_D can be neglected (i.e. $B_D \approx 1$), because the Compton quants are scattered away (from the detector) and do not contribute to the dose rate behind the shielding in the used setup.

7. Calculate the required shielding layer thickness to meet a gamma dose rate of 50 mSv/a (or more precisely, of about 25 mSv/h) in a distance of half a meter from the source (use the mean value of µ!). Use lead for the shielding material!

$$D = D_0 \cdot e^{-\mu_{air} \cdot d_{air}} \cdot e^{-\mu_{Pb} \cdot d_{Pb}}$$

$$d_{total} = d_{Pb} + d_{air} = 0{,}5 \text{ m}$$



8. Discuss the impact of the shielding material on the design of the radiation protection facilities

9. Plot the dose rate over the length of the AKR-2 tangential channel

Distance [m]	RGD27091 1. meas. [µSv/h]	RGD27091 2. meas. [µSv/h]	FH40G 1. meas. [µSv/h]	FH40G 2.meas. [µSv/h]	LB 133 1.meas. [µSv/h]	LB 133 2.meas. [µSv/h]
0.05						
0.1						
0.2						
0.3						
0.4						
0.5						
0.6						
0.8						
1.0						
1.2						
1.5						
2.0						

Tab. 4, Investigations regarding the distance law for various detectors

	RGD27091 1. meas. [µSv/h]	RGD27091 2. meas. [µSv/h]	FH40G 1. meas. [µSv/h]	FH40G 2. meas. [µSv/h]	LB 133 1. meas. [µSv/h]	LB 133 2. meas. [µSv/h]
light concrete (20 cm)						
heavy concrete (20 cm)						
heavy concrete (40 cm)						
lead (5 cm)						

Tab. 5, Protocol for the determination of the mass attenuation factor (example)

Distance [m]	Gamma dose rate [μSv/h]
1.3	
1.25	
1.2	
1.1	
1.0	
0.9	
0.8	
0.7	
0.6	
0.5	
0.4	
0.3	
0.2	
0.1	
0.0	

Tab. 6, Protocol for the determination of the gamma dose rate at the open reactor channel (example)