IN SITU GAMMA SPECTROMETRY (1995-2010)

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CONTENTS

- Few words about our research group
- some theory about in situ gamma spectrometry
- Presentation of the major results

ARISTOTLE UNIVERSITY OF THESSALONIKI

Greek and Roman monuments

FACULTIES

Biggest University in Greece

90 000 undergraduates

8 000 postgraduates

teaching and research staff: 2 304
Covers all disciplines

Independent Schools

- School of Pharmacy
- School of Physical Education and Sports Sciences
- School of Physical Education and Sports Sciences
- School of Journalism and Mass Media Studies

FACULTIES

Schools of:

- Civil Engineering
- Architecture
- Urban-Regional Planning and Development Engineering
- Rural and Surveying Engineering
- **Mathematics, Physics and Computational Sciences**
- Chemical Engineering
- Mechanical Engineering
- Electrical and Computer Engineering

School of Electrical and Computer Engineering

Admission criteria

Undergraduate : national competition (numerus clausus)

students admitted to this school have the 2nd largest marks over all Greek univ. schools

Postgraduate : selection among graduates with diploma mark > 7/10

Divisions

Laboratories of:

Electrical Machines

Power systems

High Voltage

Nuclear Technology

Electro technical materials

Laboratories of:

Electronics

Automation and Robotics

Information Processing and Computing

Computing Systems Architecture

Laboratories of:

Telecommunications

NUCLEAR TECHNOLOGY LABORATORY

Nuclear Techn

http://nestoras.ee.auth.gr

Staff 1 Professor , 1 assistant Professor, 1 Lectrurer ,

Professor

2 Research collaborators

Research: Environmental Radioactivity (measurements, models)

1 Lecturer Fellow

Interactions of fast ions with matter

Nuclear reactor safety and diagnostics

Publications in international journals > 180

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Electromagnetic spectrum

In situ γ-spectroscopy

METHODS FOR THE DERIVATION OF GAMMA DOSE RATE IN AIR FROM AN IN SITU GAMMA RAY SPECTRA

- **"spectrum stripping" method**
- **"peak area" method (can give also under certain conditions radionuclide concentrations)**

COMPARISON BETWEEN THE 2 METHODS

Spectrum stripping

DELIVERABLE

- Total absorbed gamma dose rate
- Total flux energy distribution

ADVANTAGES-DISADVANTAGES

- Contribution of each radionuclide to the total dose rate (not possible)
- Detector's geometry knowledge is needed
- Source geometry is not needed

peak area method

DELIVERABLE

- Total absorbed gamma dose rate **and** contribution of each radionuclide to the total gamma dose rate

ADVANTAGES-DISADVANTAGES

- Contribution of each radionuclide to the total dose rate (possible)
- Detector's geometry knowledge is not needed
- Source geometry is needed

SPECTRUM STRIPPING METHOD

- A count registered by the Germanium detector can be caused by the full or partial absorption of an incident photon or by the passage of a cosmic ray producing a charged particle.
- In order to convert to gamma dose rate the spectrum must be stripped from the partial absorption and cosmic ray events leaving only the events corresponding to the full absorption of a gamma ray.
- The events corresponding to partial absorption in the detector are determined by Monte Carlo simulations for different incident photon energies and angles.Therefore a complete knowledge of the detector geometry is needed.
- Knowing the computed shapes of partial absorption continuum deduced by the Monte Carlo simulation for different incident photon energies and angles ,it is therefore possible to convert the measured in-situ spectra to total incident flux spectra by applying the full absorption efficiency curve of the detector ,which is determined by calibrated point sources and Monte Carlo simulations.
- Having calculated the flux energy distribution ,the absorbed dose rate in air due to gamma radiation is easily deduced .

$$\dot{D} = \sum_{E=0}^{E=E_{\max}} \phi(E) \cdot E \cdot \frac{\mu_\alpha}{\rho}(E)$$

Example of the stripping method procedure

Object: conversion of measured spectrum to energy flux photons

Need for stripping

example

X

Simulation of the HPGe detector

Manufacturer's draft scheme

HPGe as was simulated by Monte Carlo

Efficiency of the HPGe detector

$$\varphi = \frac{S \cdot e^{-\mu r}}{4\pi \cdot r^2}$$

ϕ = unscattered flux incident to the detector (γ cm^{-2} s^{-1})
S = intensity of the source (γ s^{-1})
r = distance between source and the detector (cm)
μ = attenuation coefficient in air for the specific energy

summary for the spectrum stripping method

Spectrum stripping → Energy flux distribution

Multiply by $E \cdot \frac{\mu_\alpha}{\rho}(E)$

Absorbed dose rate energy distribution

Total absorbed dose rate

$$\dot{D} = \sum_{E=0}^{E=E_{\max}} \phi(E) \cdot E \cdot \frac{\mu_\alpha}{\rho}(E)$$

PEAK AREA METHOD

- This method was introduced by Beck et al.(HASL 258,1972) Based on this method, Helfer and Miller (Health Physics 55,15,1988) derived simple calibration factors (for the outdoor measurements) which convert the measured full absorption peak count rate to activity in the soil and dose rate in air. The only parameters that are needed are the efficiency and the crystal dimensions of the Ge detector used.

- Over the years, a number of investigators have adopted and modified the technique (for a review, see ICRU report 53. *Clouvas et al.Health Physics 79,274,2000* extended this technique for measurements in an indoor environment and particularly in case of masonry structure.

GAMMA DOSE RATE DETERMINATION BY THE "peak area method"

- The procedure starts with the measurement of an indoor or outdoor gamma spectrum.

- What is **directly deduced** by the in situ gamma spectrometry measurement is the **number N** of counts in each photopeak per unit of time (in counts per minute)
- For each photopeak the dose rate in air due to unscattered photons (primary photons of **energy *E***) is $\dot{D}_p(E) = E \times (N/\varepsilon) \times \mu_a/\rho$
- $\mu(E)$ is the mass absorption coefficient
- ε is the efficiency :peak count rate (in counts per minute) per unit uncollided flux (photons cm^{-2} s^{-1}) for a parallel beam of gamma rays of energy *E* that is incident-normal to the detector face.

"peak area" method (calibration factors for outdoor environment)

Calibration equation

$$\frac{N_f}{A} = \frac{N_f}{N_0} \cdot \frac{N_0}{\phi} \cdot \frac{\phi}{A} \qquad \frac{N_f}{\dot{D}} = \frac{N_f}{N_0} \cdot \frac{N_0}{\phi} \cdot \frac{\phi}{\dot{D}}$$

N_f : Total absorption peak count rate (cpm) in the spectrum at the energy of a particular nuclide gamma transition

N_0 : peak count rate (cpm) for a parallel beam of gamma rays of the same energy incident-normal to the detector's face

ϕ : Unscattered flux at a specific energy 1 m above ground

A : Concentration of the radionuclide in soil (Bq/kg)

$\dot{D}$: Absorbed dose rate in air (nGy/h) due to gamma radiation of a specific radiocuclide

Angular correction

- In an in-situ outdoor or indoor - spectrometry measurement the incident radiation is not just a parallel flux normal to the detector's face but has all angles of incidence.
- Angular response of the detector is not a critical factor at least up to 120° of incidence

Detector efficiency ε (E)

- As in the stripping method ε(E) is determined by point sources or/and Monte Carlo simulations.
- Alternatively, with a good approximation the generic factors of Helfer and Miller (Health Physics 55,29,1988) can also be used.

Unscattered flux equation for uniform distribution of radionuclides in soil

$$\phi = -S_0 \cdot \frac{\displaystyle\int_0^1 \frac{e^{-\left(\frac{\mu_\alpha}{\rho_\alpha}\rho_\alpha\right)\frac{h}{\omega}} \cdot \left(e^{-\left(\frac{\mu_s}{\rho_s}\rho_s\right)\frac{z}{\omega}} - 1\right)}{\left(\frac{\mu_s}{\rho_s}\right)} d\omega}{2 \cdot \rho_s}$$

where

- S_0 : Intensity of the source in photons/cm^3 s
- z : depth till the source is distributed
- ω : cosθ
- ρ_s : soil's density
- μ_s/ρ_s : mass attenuation coefficient in soil
- ρ_α : air's density
- μ_α/ρ_α : mass attenuation coefficient in air
- h : detector's distance from soil

Section of an external space in spherical coordinates

Simulation of 2π geometry

Simulation of 2π geometry (MCNP code). On the right, zoom in the detector's area can be seen

Build up factor definition

❖ The total absorbed dose rate in air from a radionuclide can be expressed as a sum of the absorbed dose rate due to unscattered photons and scattered photons in the indoor or outdoor environment.

$$\dot{D}_t = \dot{D}_p + \dot{D}_s \quad (1)$$

❖Relation (1) can be rewritten as

$$\dot{D}_t = \dot{D}_p \times (1 + \dot{D}_s / \dot{D}_p)$$

❖The expression within the parenthesis is termed the dose buildup factor *B*.

$$\dot{D}_t = \dot{D}_p \times B$$

- *The dimensionless build up factor B can be calculated if the geometry of the outdoor or indoor environment is known*
- *Required: model of outdoor or indoor environment*

2π geometry conclusions

✓ The methodology used for the derivation of the gamma dose rates from the *in situ* gamma ray spectra is the one introduced by Beck et al. Based on this method, simple calibration factors (for the outdoor measurements) were deduced which convert the measured full absorption peak count rate to activity in the soil and dose rate in air.

✓ The only parameters that are needed are the efficiency and the crystal dimensions of the Ge detector used.

✓it can be considered that the Ge detector has a uniform response over angles at least up to 120° of incidence

✓Alternatively, with a good approximation the generic factors of Helfer and Miller can also be used. The only parameters that are needed are the efficiency and the crystal dimensions of the Ge detector used.

Simulation of 4π geometry

A' way of simulation of an indoor space (massive brick)

B' way of simulation of an indoor space (brick with holes: 7cm material, 6 cm air, 7 cm material)

Element	Contribution to weight composition (%)
O	51.35
Al	14.3
Fe	0.9
Si	32.25
Ti	1.2

Weight composition of brick.

Element	Contribution to weight composition (%)
O	53.1
Al	3.4
Fe	1.4
Si	33.8
Ca	4.4
K	1.3
Na	1.6
H	1.0

Weight composition of concrete.

Build up factors for 4π geometry

Dose build up factor *B* for three different wall thickness (10 cm, 20 cm and 30 cm). The dimensions of the room used for this calculation were 4X4X3 m and the thickness of floor and ceiling was 0.2 m.

Dose build up factor *B* calculation for the same room dimensions as before. The wall thickness is 20 cm. In the upper curve the building material of the wall is only brick and in the lower curve the wall is modeled a sum of three slabs (7 cm brick + 6 cm air + 7 cm brick).

Build up factors for 4π geometry

Dose build up factor *B* for different room dimensions

Influence of the density of the bricks (ρ_b) to the build up factor *B*.

Build up factors for 4π geometry

Dose build up factor *B* for two different gamma emission fields. In the upper curve (circles) the photons are emitted only from the floor (the four walls and ceiling are just scattering medium). In the lower curve (squares) the photons are emitted from the four walls, floor and ceiling.

Comparison between the indoor dose build up factors calculated in the present work with those deduced for the outdoor environment by the calculations of Beck et al. (1972)

4π geometry conclusions

✓The build up factor *B* does not depend strongly on parameters such as a)dimensions of the rooms, b)the thickness of walls, c)the density of the building materials, and d)the gamma source geometry.

✓If only approximate results are desired it is possible to use those factors derived for a half-space geometry also for the indoor environment. In the case of a masonry structure this approach is adequate, due to the fact that the ratio of primary to scattered flux is approximately the same for the two environments

✓Total gamma dose rates can be directly deduced from in situ gamma spectra using a “spectral stripping method” which does not require any assumptions concerning the source geometry

Build up and Conversion factors

Photon energies used for the determination of the indoor and outdoor absorbed gamma dose rates, their associated radionuclides as well as the corresponding Build up factor *B* and the contribution *C* to the dose rate due to all photon energies of each radionuclide

Nuclide	Energy (keV)	outdoor		indoor	
		B	*C*	*B*	*C*
Uranium series					
^{214}Pb	351.932	3.21	0.57	2.87	0.57
^{214}Bi	609.312	2.57	0.19	2.35	0.20
Thorium series					
^{212}Pb	238.632	3.64	0.90	3.25	0.90
^{208}Tl	583.191	2.62	0.14	2.39	0.16
^{212}Bi	727.33	2.46	0.47	2.23	0.48
^{228}Ac	911.204	2.31	0.28	2.07	0.28
^{137}Cs	661.657	2.52	1.00	2.30	1.00
^{40}K	1460.83	1.99	1.00	1.80	1.00

Dose rate conversion factor *DRCF*

Nuclide	Energy (keV)	outdoor *DRCF* (nGy/h per gamma$_{unscattered}$ cm^{-2} s^{-1})	indoor *DRCF* (nGy/h per gamma$_{unscattered}$ cm^{-2} s^{-1})
Uranium series			
^{214}Pb	351.932	267	219
^{214}Bi	609.312	165	143
Thorium series			
^{212}Pb	238.632	368	288
^{208}Tl	583.191	342	287
^{212}Bi	727.33	1443	1212
^{228}Ac	911.204	332	281
^{137}Cs	661.657	28	26
^{40}K	1460.83	43	39

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Nuclear Technology Laboratory

Nuclear Technology Laboratory

TD 66.1±14.3
R 41.8-86.1
U 31.8%
Th 37.1%
K 27.2%
Cs 4.0%

TD 62.6±18.9
R 40.8-94.6
U 31.2%
Th 36.9%
K 26.7%
Cs 5.2%

TD 28.3±4.3
R 23.9-34.2
U 29.0%
Th 27.9%
K 33.8%
Cs 9.4%

TD 80.8±16.5
R 62.6-107.7
U 24.7%
Th 38.7%
K 33.9%
Cs 2.7%

TD 53.5±15.1
R 30.3-70.6
U 24.4%
Th 38.3%
K 34.2%
Cs 3.1%

TD 81.0±30.9
R 56.0-153.0
U 24.6%
Th 37.1%
K 37.6%
Cs 0.7%

TD 84.7±37.1
R 34.0-149.2
U 23.8%
Th 39.5%
K 36.6%
Cs 0.1%

TD 49.6±18.5
R 10.5-71.8
U 25.1%
Th 37.0%
K 41.5%
Cs 1.6%

TD 60.6±20.0
R 22.9-81.3
U 24.0%
Th 36.0%
K 37.6%
Cs 2.3%

TD 88.0±47.3
R 40.5-173.4
U 20.3%
Th 32.4%
K 34.0%
Cs 13.4%

TD 37.9±19.8
R 14.4-59.6
U 27.3%
Th 25.4%
K 28.4%
Cs 18.9%

TD 23.7±7.7
R 14.1-39.8
U 50.5%
Th 21.9%
K 25.5%
Cs 2.1%

TD 36.3±13.2
R 19.7-52.5
U 27.4%
Th 31.8%
K 34.1%
Cs 6.7%

TD 26.7±7.5
R 17.5-35.9
U 31.4%
Th 31.8%
K 29.3%
Cs 7.4%

TD 59.2±6.7
R 51.9-69.9
U 48.5%
Th 27.9%
K 23.6%
Cs 0.0%

TD 33.4±4.3
R 29.7-39.5
U 28.0%
Th 35.4%
K 33.5%
Cs 3.1%

TD 42.0±19.7
R 13.0-68.5
U 23.3%
Th 36.1%
K 37.5%
Cs 3.1%

TD 37.5±8.1
R 27.3-52.4
U 25.3%
Th 33.0%
K 37.1%
Cs 4.6%

TD 55.9±29.3
R 27.4-107.4
U 24.1%
Th 31.1%
K 41.3%
Cs 3.5%

TD 42.9±14.2
R 22.9-69.1
U 20.7%
Th 37.3%
K 40.1%
Cs 1.9%

TD 85.2±26.8
R 44.5-113.7
U 26.8%
Th 43.0%
K 29.0%
Cs 1.1%

TD 64.5±9.0
R 52.4-74.1
U 19.6%
Th 37.0%
K 19.4%
Cs 24.0%

TD 46.2±16.1
R 9.6-107.1
U 22.1%
Th 33.1%
K 31.1%
Cs 14.0%

TD 98.3±42.7
R 46.3-195.8
U 22.9%
Th 47.3%
K 29.5%
Cs 0.4%

TD 48.0±17.0
R 24.2-66.1
U 24.8%
Th 25.6%
K 29.5%
Cs 20.1%

TD 37.8±8.3
R 28.8-48.9
U 24.5%
Th 34.1%
K 35.7%
Cs 5.7%

TD 108.5±59.0
R 15.4-208.8
U 21.8%
Th 50.2%
K 27.0%
Cs 1.0%

TD 30.6±13.3
R 10.1-48.3
U 25.4%
Th 27.5%
K 36.7%
Cs 10.4%

TD 22.3±7.3
R 13.0-30.7
U 32.9%
Th 34.1%
K 25.7%
Cs 7.3%

TD 46.2±7.0
R 38.1-59.1
U 23.4%
Th 26.1%
K 33.6%
Cs 16.9%

TD 40.7±22.0
R 18.4-98.9
U 38.5%
Th 36.1%
K 24.2%
Cs 1.3%

TD 32.1±13.8
R 12.9-48.6
U 22.6%
Th 33.8%
K 33.4%
Cs 10.2%

TD 23.3±9.1
R 15.2-39.2
U 33.4%
Th 27.7%
K 29.3%
Cs 9.7%

TD 61.2±15.8
R 37.9-92.8
U 32.7%
Th 37.6%
K 28.6%
Cs 1.1%

TD 23.7±4.9
R 16.6-31.5
U 37.4%
Th 26.8%
K 33.8%
Cs 2.0%

TD 20.2±8.9
R 11.2-31.3
U 32.9%
Th 36.5%
K 36.1%
Cs 1.8%

TD 22.6±11.0
R 10.6-37.1
U 33.5%
Th 29.9%
K 34.5%
Cs 2.1%

TD 29.8±7.0
R 20.0-37.7
U 20.8%
Th 38.6%
K 36.1%
Cs 4.5%

TD 22.4±7.9
R 5.1-41.0
U 44.7%
Th 27.3%
K 26.6%
Cs 1.4%

TD 30.3±9.3
R 17.8-45.3
U 33.0%
Th 33.0%
K 27.1%
Cs 6.9%

TD 29.5±13.3
R 6.7-57.8
U 35.1%
Th 25.6%
K 34.4%
Cs 5.0%

TD 34.7±14.1
R 13.6-55.7
U 29.2%
Th 34.5%
K 30.4%
Cs 6.0%

TD 21.9±9.7
R 6.6-39.7
U 31.0%
Th 33.1%
K 31.9%
Cs 4.0%

TD 29.9±4.9
R 24.9-34.8
U 27.6%
Th 37.6%
K 31.7%
Cs 3.1%

TD 30.3±11.6
R 15.6-44.0
U 39.2%
Th 33.4%
K 23.3%
Cs 4.1%

TD 22.4±3.8
R 16.8-30.7
U 50.6%
Th 20.1%
K 22.0%
Cs 9.9%

TD 25.5±5.5
R 18.5-33.4
U 30.2%
Th 34.1%
K 33.5%
Cs 2.1%

TD 17.3±4.8
R 9.5-21.6
U 39.3%
Th 34.6%
K 26.1%
Cs 0.0%

TD 24.9±4.8
R 20.1-29.7
U 49.6%
Th 28.1%
K 18.7%
Cs 3.6%

TD 34.7±12.0
R 26.8-59.9
U 55.3%
Th 24.5%
K 19.8%
Cs 0.4%

TD 35.6±12.5
R 16.7-65.3
U 34.3%
Th 34.4%
K 30.1%
Cs 1.2%

TD 33.8±12.7
R 18.4-49.4
U 48.4%
Th 23.7%
K 26.9%
Cs 1.0%

TD 27.4±11.4
R 13.3-46.5
U 42.6%
Th 28.9%
K 25.9%
Cs 2.6%

TD 35.1±0.8
R 34.4-35.9
U 54.4%
Th 29.0%
K 13.0%
Cs 3.7%

TD 42.5±9.6
R 33.6-57.7
U 40.1%
Th 35.1%
K 22.0%
Cs 2.7%

	URANIUM SERIES		THORIUM SERIES		K-40		TOTAL	
dwellings	mean	range	mean	range	mean	range	mean	range
	nGy h^{-1}	nGy h^{-1}	nGy h^{-1}	nGy h^{-1}	nGy h^{-1}	nGy h^{-1}	nGy h^{-1}	nGy h^{-1}
26	15±4	9-26	21±5	12-34	22±4	14-31	58±12	40-87
60	13±3	9-22	20±3	12-31	23±4	14-35	56±9	41-80
225	15±4	5-28	21±5	6-36	23±5	5-35	59±13	18-99

Figure 4. Indoor dose rate due to ^{137}Cs as function of the construction year of the house or building.

Figure 5. Indoor dose rate due to ^{40}K as function of the construction year of the house or building.

Apartments in ground and first floor
Indoor radon concentration in Bq/m3
140
120
100
80
60
40
20
0
0
5
10
15
20
25
30
Gamma dose rate due to Uranium series in nGy/h

Apartments above third floor
RADON CONCENTRATION (in Bq/m3)
100
90
80
70
60
50
40
30
20
10
0
0
2
4
6
8
10
12
14
16
18
20
GAMMA DOSE RATE DUE TO URANIUM SERIES (in nGy/h)

Measurements in forest ecosystems

TAXIARXIS

OAK

C = 1

SANI

PINE

C = 0.82

From the combination of in situ gamma spectrometry measurements and Monte Carlo Simulations a conversion Factor C (nGy/h per kBq./m^2) for Cs-137 was deduced for the 2 forests.

Industrial application

Determining minimum alarm activities of radioactive sources in scrap metal using Gamma Spectrometry measurements and Monte Carlo techniques

Radioactive contamination of scrap metal

- Radioactive sources used in industry or medicine
- NORM radiation (naturally occurring radioactive material) from oil or chemical industries
- Contaminated metal from nuclear facilities

Dealing with the problem

- Installation of radiation portal monitors
  - Metal recycling industries
  - Border crossing

Comparison between measurements and simulations

- Homogenous load experiment
- Scrap load experiments
  - Old scrap E1
  - Shredded scrap
- Experiments' simulation

Homogenous load experiment

Minimum Alarm Activities Cs-137

…….Measurements abroad (Germany)

PTB

UDO

SOUL
gsf

Previous discharges of radioactivity from Mayak Production Association plant in Urals, Russia have resulted in considerable radionuclide contamination of the Techa River, and consequent high radiation doses during the late 1940s and 1950s to residents of villages along the Techa River. The most contaminated villages were evacuated in the period 1954 - 1962.

The aim of the SOUL project is to quantify risks of late health effects associated with low-dose rate exposure to plutonium, strontium and external gamma radiation. This will be done by improving, updating and analysing dosimetric and health data for the Mayak worker cohort (MWC), the extended Techa River cohort (ETRC) and the Techa River offspring cohort (TROC).

- **Dose conversion coefficients for film badges and tooth enamel for exposures at the early work places are evaluated ,by use of simulated photon spectra**

- **Due to differences in fuel parameters the exposure conditions at current work places differ from those of the early work places. However , measurements at current workplaces and the comparison with Monte Carlo simulated photon spectra at these workplaces are needed for validating calculations and simulation parameters.**

PHOTON SPECTRA IN MAYAK WORK PLACES

- More than 400 spectra were measured in the course of work, 17 points of them with angle distribution (7 spectra in each point) All of them were “unfolded” using the codes and procedure developed by the AUTH scientists

- Due to specific regulations it was not possible for AUTH scientists to perform measurements inside MAYAK. However, it was possible to perform in common with the MAYAK scientists measurements in the MAYAK test ground

Photon spectra at the Radiochemical Plant: chem_3

Validation :Comparison between measured and calculated photon flux energy distributions

- **MAYAK PA work place**
- (No1, No2, No3): The positions where the in situ gamma spectrometry measurements have been performed.
- 1: lead barrier.
- 2: 257 mCi 137Cs source.

RESULTS

	Unscattered Flux Measured	Unscattered Flux MCNP
No1	3.4 ± 0.5	3.9
No2	7.0 ± 1.0	7.7
No3	4.8 ± 0.7	4.2

Measured and calculated (MCNP) unscattered photon flux (photons per cm^2 per second) in the three locations

Experimental and simulated scattered photon flux energy distribution in position No2. A good agreement was found also in the other two positions.

POSITION No1

POSITION No3

.. Exotic Use of Germanium detector

Measurement of cosmic radiation (muon dose rate) with a Germanium detector

Typical spectrum of in situ gamma spectrometry

Typical spectrum of in situ gamma spectrometry

Spectrum of in-situ γ-spectrometry with attenuation gain of preamp and amplifier

Muon measurements with Germanium detectors

Shift of muon peak proportional to the crystal's dimensions

Spectra Simulated - Measured

HPGe 10% efficiency

HPGe 20% efficiency

HPGe 70% efficiency

Remark:

The simulated spectra in the lower energies are smaller then the measured ones

This is due to electron photon shower.

Dose calculation from a muon spectrum

$$\dot{D}(E_i) = N(E_i) \cdot \frac{E_i}{t \cdot V \cdot \rho}$$

where :

N(*Ei*): counts in energy *Ei*

t: measuring time

V: crystal's active volume

ρ: crystal's density

Total dose: $$\dot{D} = \sum_{i=4}^{i=120} \dot{D}(E_i)$$

Conversion factor C = $\text{Dose}_{(ICRU)}$ / $\text{Dose}_{(Ge)}$ = 1.33

CORRESPONDING PUBLICATIONS

- Clouvas et al Health Physics 74,216,(1998)
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