DESIGN OF THE GAMMA COUNTER LOCAL SHIELD FOR THE INSTALLATION INTENDED FOR IDENTIFICATION OF HIDDEN EXPLOSIVES AND DRUGS

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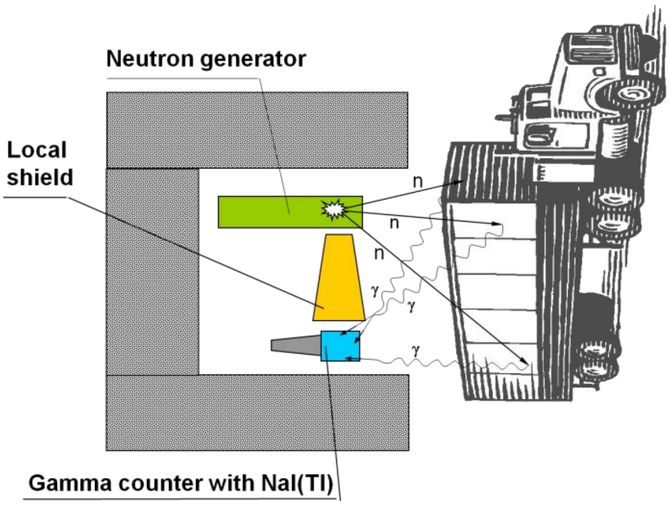
<u>The aim and tasks:</u> computational modeling of the local shielding of a gamma spectrometer against 14.1 MeV neutrons for choosing the optimal shielding material and design.

Identification of hidden explosives and drugs is the most urgent task for custom or security services. Prompt gamma-ray neutron activation analysis is now a proven method for the fast nondestructive check of storage container or luggage. Neutrons can easily penetrate the casing of a container and they excite the atomic nuclei of the chemicals inside. These nuclei promptly de-excite by emission of gamma-rays that also penetrate the container casing and can be detected by the external gamma-spectrometer. The shape of gamma-ray spectrum can be evidence of the presence of warfare or drug agents within the container.

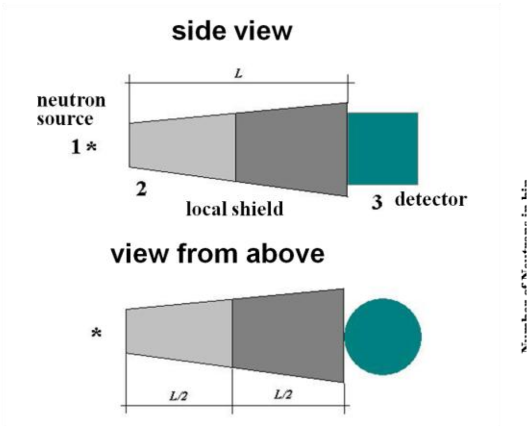
As a neutron source now usually uses a portable pulse-type neutron generator with neutron energy of 14 MeV and neutron yield to (2-5)·108 n/s. Such source is rather dangerous for the service personnel firstly and produces a large neutron-induced background of the gamma-spectrometer secondly. The effective radiation shield is necessary in order to avoid the unjustified exposure of personnel and reduce the gamma-spectrometer background. The calculations of the different variants of radiation shield between the neutron generator and gamma-spectro-meter have been done. Two types of shielding have been examined in this work: homogeneous and heterogeneous shields (Fe; CH2; Pb; Fe+CH2; CH2+Fe; CH2+Pb; Pb+CH2) for optimization of the shield mass and dimensions. The calculations have been carried out by the MC code MCNP4C.

Monte Carlo N-Particle Transport Code (MCNP) is a software package for simulating nuclear processes. It is developed by Los Alamos National Laboratory since at least 195 with several further major improvements. It is distributed within the United States by the Radiation Safety Information Computational Center in Oak Ridge, TN and internationally by the Nuclear Energy Agency in Paris, France. It is used primarily for the simulation of nuclear processes, such as fission, but has the capability to simulate particle interactions involving neutrons, photons, and electrons. "Specific areas of application include, but are not limited to, radiation protection and dosimetry, radiation shielding, radiography, medical physics, nuclear criticality safety, detector design and analysis, nuclear oil well logging, accelerator target design, fission and fusion reactor design, decontamination and decommissioning."

MCNPX (Monte Carlo N-Particle eXtended) was also developed at Los Alamos National Laboratory, and is capable of simulating particle interactions of 34 different types of particles (nucleons and ions) and 2000+ heavy ions at nearly all energies, including those simulated by MCNP.



Schematic view of a custom control post



Calculation geometry

This practical work will help students to master MCNP – a universal software package for the simulation of neutron and gamma ray transport in matter.

Recommended: http://en.wikipedia.org/wiki/Origin_(software)

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