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MCNP modelling of a neutron generator and its shielding for PGNAA in mineral exploration

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SUMMARY

This paper present results of Monte Carlo simulations of shielding design against neutron and gamma-rays from a D-D 2.5MeV neutron generator. The generator will be located in a restricted access laboratory at the Department Of Exploration Geophysics at Curtin University. To protect staff and students from radiation we need to calculate shielding characteristics needed to reduce the effective dose, from the generator, to safe limits.

Since operation facility is of limited dimensions, shielding needs to be optimised in terms of it thickness and the cost as well. Shielding calculations were made using the MCNP6.1 Monte Carlo code. We were required by Radiological Council of Western Australia to put sufficient shielding to achieve a conservative dose constraint for non-radiation workers of 0.5 mSv per year or 9.6 µSv in a week. The shielding was modelled as a hollow sphere of varying shielding thickness of borated polyethylene (BPE), concrete and lead (Pb). Our goal was to determine thickness of concrete needed to decrease the effective dose below prescribed limits. We already purchased 15cm thick BPE and 2.2mm Pb slabs. As a result, we concluded that 15cm thick concrete shielding will be enough to safely operate neutron generator.

Our neutron generator will be one of the main components of our proposed prompt gamma neutron activation (PGNA) logging-while-drilling (LWD) tool. This tool should be able to reliably identify the major elements of rock units, including the presence of metallic ores. The availability of such real-time information should improve almost every stage of mining and mineral processing.

Key words: MCNP, radiation shielding, neutron generator, PGNA, LWD.

INTRODUCTION

A prompt gamma neutron activation (PGNA) technique has wide application in many branches of industry, including mining and exploration (Borsaru, Zhou, Aizawa, Karashima, & Hashimoto, 2006; Charbucinski, Malos, Rojc, & Smith, 2003; Gozani, 1985; Lim & Sowerby, 2005; Naqvi, Abdelmonem, Al-Misned, & Al-Ghamdi, 2006; Seabury & Caffrey, 2004). PGNA is a near real-time, penetrative, non-destructive analytical method used for detecting and quantifying many elements in the sample simultaneously. Logging While Drilling (LWD) is a technique that involves gathering downhole data while drilling a well.

Our idea is to develop a prototype prompt gamma neutron activation logging-while-drilling tool that can reliably identify bulk elemental composition of rock units for identification of rock type. Rather than supplant later analytical laboratory analysis of core, the focus will be on implementing technologies that will be acceptable to industry in performance And cost, and without export or shipping restrictions. We propose to implement a novel borehole logging strategy using an instrumented core-barrel integrated with the commonly used diamond core drilling process. The prototype device will be viable example of a new down-hole geological logging tool to be used widely in the mining industry. The prototype will pave the way for automated geological logging in every diamond drilling hole. Many drill holes have little or no elemental analysis, and none have in-situ analysis creating inefficiency and waste in exploration and mining through a lack of timely information. Experienced geologists would be more productive in that they may remotely monitor drilling progress instead of being present in a remote area visually inspecting rock core.

The tool will have three principal components: the neutron generating source, the gamma-ray detector and the data acquisition system. The switchable neutron generator is central to the effective operation of the tool. It should be relatively inexpensive to operate; by having a sufficiently long lifetime, and be robust enough to operate in an LWD environment. Our preliminary research indicated that a D-D neutron tube is a suitable candidate for the diamond drilling LWD applications. We purchased D-D nGen-310 neutron generator from Starfire Industries to install it for our laboratory experiments (Starfire Industries), which is shown in Figure 1.

Figure 1. nGen-310-DD neutron generator, powered by Starfire Industries nGen technology.

Neutron generator testing will take place in a purpose built facility, equipped with appropriate radiation shielding and radiation monitor system. The set-up will be enclosed in a 2.5 m x 2.5 m partitioned area within a larger laboratory, which will also be equipped with swipe card access and be operated only by authorised individuals with a suitable level of radiation safety training working under the supervision of a radiation licence holder. Since the testing laboratory is in a campus building, we have to make sure that the shielding is adequate to protect the public and operators of the neutron generator from radiation. Due to the limited space of our laboratory and research budget, we need to optimise our shielding in terms of dimensions and shielding materials. For radiation shielding, we have to consider both neutron and photon interaction with matter. Many studies concerning neutron shielding and effectiveness of different shielding materials have been published in the past (Sharma, Alajo, & Liu, 2014). Considering these findings, we opted for borated polyethylene (BPE), lead (Pb) and ordinary concrete as our shielding materials. BPE is commonly used for absorbing neutrons in nuclear industry (Singleterry Jr & Sheila A, 2000). Borated polyethylene is made from high density plastic $(C_2H_4)_n$ with 5% boron content by weight. The high hydrogen content of the plastic moderates the neutrons which are subsequently captured by the boron, which has one of the highest neutron capture cross-sections. Concrete is the most commonly used radiation shielding material; it is cheaper than other shielding materials and it is suitable for both neutron and gamma-ray shielding. Lead (Pb) is a good shielding material for gamma rays, due to its high density of 11.35 g cm^{-3}. Many researchers utilize MCNP as well as other Monte Carlo simulation codes to determine the effectiveness of different shielding materials in reducing the dose rates (Calzada, Grünauer, Schillinger, & Türck, 2011; Osborn, Ersez, & Braoudakis, 2006; X. da Silva & R. Crispim, 2001).

Schedule I 1(3) of the Radiation Safety (General) Regulations 1983 ("Radiation Safety (General) Regulations 1983,") states that the effective dose for non-radiation workers must not exceed 1 mSv per year when averaged over 5 years or 5 mSv in a single year. The Radiological Council require a conservative dose constraint of 0.5 mSv per year or 9.6 µSv in a week. Personnel will not occupy the neutron generator room during operation. The laboratories around the neutron generator room will accommodate both radiation workers and non-radiation workers, so all calculations have been conducted with reference to the limits for non-radiation workers.

METHOD AND RESULTS

Modelling of the neutron doses and shielding requirements was conducted using MCNP6, a general-purpose Monte Carlo N-Particle code developed by Los Alamos National Laboratory (Goorley et al., 2012). Detector geometry and source characteristics are specified in input text files and the resulting fluences and doses automatically written to output text files. As an initial test, MCNP was used to model the effective dose from an unshielded source. This calculation provides an indication of the maximum possible level of exposure with no safety controls. With the neutron generator in the middle of the room, the distance to any wall is 1.25 m. The neutron flux on the surface of an imaginary sphere of radius 1.25 m centred on a source emitting 10^7 n/s was calculated to be 75.25 µSv/h. We also used analytical calculation to theoretically estimate the unshielded dose rate at any point (DE_0), distant R cm away from a neutron source of strength S (neutrons per second):

$$DE_0 = \frac{Sq}{4\pi r^2}$$

where q is the effective dose per neutron fluence in pSv cm^2 for mono-energetic neutrons incident in various geometries on an adult anthropomorphic computational model, given in table J1 of ICRP publication 119 (Appendix II) (Eckerman, Harrison, Menzel, & Clement, 2013). The similar result of 74.8 µSv/h was obtained.

To estimate the total effective dose we have to assume a maximum exposure time for laboratory workers, which is determined by the duty cycle of the neutron generator. The DD neutron tube in the nGen-310 has a limited operating lifetime and so it will not be operated for more than 2 hours/day or 10 hours/week. Therefore, the unshielded exposure dose rate, assuming this operational cycle, would be 752.5 µSv/week. This is more than 78 times higher than the legally allowed dose so it is clear that shielding will be required to ensure safe operation of the neutron generator. For a 10 hour/week operation, the public limit is 0.96 µSv/h.

We made a simplified shielding model, shown in Figure 2, assuming concentric spheres of selected shielding materials, placed in a hollow sphere filled with air and centred on the source. With fixed dimensions of BPE and Pb shielding, our goal was to determine thickness of concrete needed to decrease the effective dose below the prescribed limit. The neutron generator was modelled as an isotropic point source of 2.5 MeV neutrons, emitting 10^7 neutrons/second, the maximum rate achievable of the nGen-310.

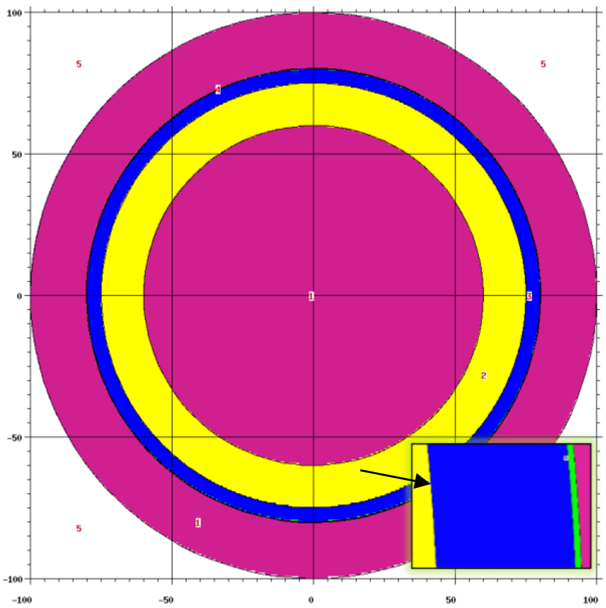

Figure 2. Shielding model simulated in MCNP6.1. Air filled sphere with the shielding inside is represented in pink colour. Fixed 15 cm of BPE is yellow coloured; varying thicknesses of concrete (5 cm in this figure) is depicted blue. Being too thin (only 0.22 cm), Pb layer is represented in enlarged image area (green coloured).

The effective dose due to both neutrons and gamma-rays was calculated by convolution of the simulated fluence with fluence-to-dose data provided in ICRP Publication 119. MCNP automatically computes this when the fluence-to-dose data are specified in the input file.

First we obtained effective dose on the surface of the 15 cm thick BPE, which was placed 75 cm from the isotropic source. To speed up the simulation and decrease amount of computer time needed to calculate the dose at certain distance, we analytically calculated resulting effective dose 1.25 m from the source, using inverse square law:

$$\frac{I_1}{I_2} = \frac{D_2^2}{D_1^2}$$

Where:
I_1 = Intensity at D_1 = 19.58 $\mu\mathrm{Sv/h}$
I_2 = Intensity at D_2 = ?
D_1 = 75 cm
D_2 = 125 cm

$$I_2 = \frac{19.58\ \mu\mathrm{Sv/h} \cdot\ (75cm)^2}{(125cm)^2}$$

The BPE shield alone was able to reduce the dose to 7.05 $\mu Sv/h$, or for 90.6 %. Next, we added 5 cm of concrete on top of the BPE shield, and 0.22 cm of lead, and obtained effective dose on the surface of lead sphere. With this geometry, overall effective dose (considering neutrons and photons) was decreased to 3.76 $\mu Sv/h$, or for 95 %. In addition, concrete and lead decreased effective dose of photons originated from BPE shield for 46 %. We run several simulations, using the same configuration, incrementing thickness of concrete for the 5 cm, until we gained dose that is beyond dose limit of 0.96 µSv/h.

Figure 3 shows effective dose calculated for different thicknesses of concrete. With 10 cm of concrete between BPE and Pb, overall dose was decreased to 2 $\mu\mathrm{Sv/h}$ or for 97.32 %. We also calculated that this extra 5 cm of concrete only, was able to reduce photon dose rate for 31 %. With 15 cm of concrete thick shield, we gain effective dose rate of 1.08 $\mu\mathrm{Sv/h}$, which is close to prescribed limit. Finally, 20 cm of concrete reduced the dose rate to 0.58 $\mu\mathrm{Sv/h}$. Figure 3 shows the effective dose rate (red dots), obtained with varying shielding thicknesses of concrete between BPE and Pb shields of fixed dimensions.

CONCLUSIONS

With limited laboratory space and budget for our project, we need to optimise shielding needed to reduce total effective dose to prescribed limits. With 15 cm of BPE and 0.22 cm of lead already available, we would need approximately 17cm of ordinary concrete, to accomplish this goal. This amount of concrete would occupy a lot of space in the lab, so we need to reconsider other alternatives, and do more simulations in the future. To reduce amount of concrete necessary, and save laboratory space, we can cover walls of the neutron lab with lead panels. Other possibility is to place polyethylene slabs around our neutron generator, within BPE shield. Different types of concrete, with some materials added, can enhance its shielding efficiency (Waly & Bourham, 2015). Unfortunately, we don’t have a supplier for any other type of concrete except ordinary. At the moment, we are studying variance reduction technique for MCNP that will enable us to model more realistic geometry for our shielding, and obtain more accurate results.

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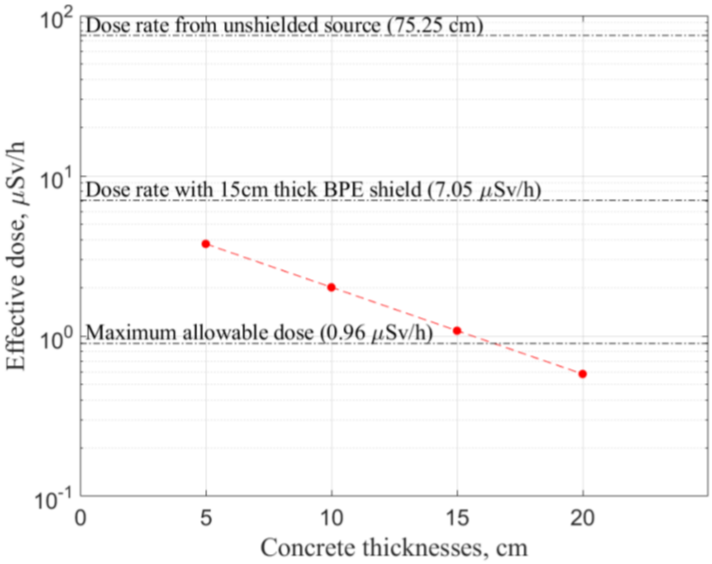

Figure 3. Effective dose rate obtained with shielding configuration with varying thicknesses of concrete.