Application of fluorine-based threshold activation detector for neutron flux calculation from D-T neutron generator

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***Abstract*—In this paper we propose a method of fast neutron flux estimation from a pulsed D-T neutron generator with application of single CaF_2 scintillation crystal. The analysis method relies on $^{19}F(n,\alpha)^{16}N$ threshold activation reaction having neutron energy threshold at 1.6 MeV. As a result, the ^{16}N undergo β^- decay with half-life of 7.1 s, emitting β particles with endpoint up to 10.4 MeV in the scintillator medium. Integration of the β distribution curve, preceded by calculation of (n,α) rate on F with Monte Carlo N-Particle Transport Code v6 (MCNP6) for fixed geometry, allows to estimate the neutron flux in 4π per second within few minutes.**

***Index Terms*—neutron detection, Threshold Activation Detection, scintillator, neutron generator**

I. Introduction

One of a technique useful for fast neutron detection is known as a Threshold Activation Detection (TAD) [1], [2]. It relies on registration of characteristic radiation emitted from specific nuclei after threshold reaction with neutrons. A possible approach is to use a scintillation material containing itself a material, which undergoes the activation reaction, such as fluorine. In this work we investigated a CaF_2 scintillator, which mass fraction of Ca and F is about 50 / 50% [3]. The F nuclei, present in the scintillator medium, undergo i.e. $^{19}F(n,\alpha)^{16}N$ reaction, resulting in emission of high energy beta particles (endpoints at 10.4 MeV and 4.3 MeV) and γ rays (6.1 MeV) with the half-life of 7.1 s. The (n,α) reaction cross section reaches about 100 mb at about 5 MeV (see Fig. 1), which is sufficient for such application. In this work, we propose a new method of neutron flux estimation from deuterium-tritium (D-T) neutron generators operated in pulse mode. An undoped CaF_2 with size of ⌀5”×3” was used, however, other scintillators containing fluorine such as EJ-313/BC-509, BaF_2 or fluorocarbon plastic [4] can be also considered for that application. The method usefulness for deuterium-deuterium (D-D) neutron generators flux monitoring will be also discussed. The CaF_2 scintillator was previously used for simultaneous detection of prompt photofission neutrons and delayed γ-rays from nuclear materials, potentially hidden in cargo container. These field trials were performed at Maasvlakte Rotterdam Seaport in the frame of C-BORD project.

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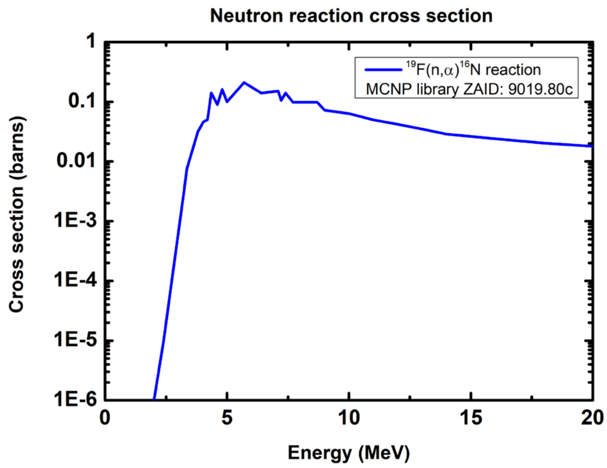

Fig. 1. The $^{19}F(n,\alpha)^{16}N$ neutron cross section used in the MCNP simulations, a ZAID 9019.80c library was chosen.

II. Methodology

A. Experimental setup

Fig. 2 presents the experimental setup at NCBJ measurement facility, equipped with a Sodern Genie 16D D-T neutron generator. Data were registered with a CAEN DT5730 8 channel 500 MS/s desktop digitizer with DPP-PSD firmware and DigiTes 4.5.10 control software. The 1 ms excitation and 10 ms repetition neutron generator pulse mode was set. The trigger width was shaped by a laboratory-grade digital logic pulse shaper based on Arduino,

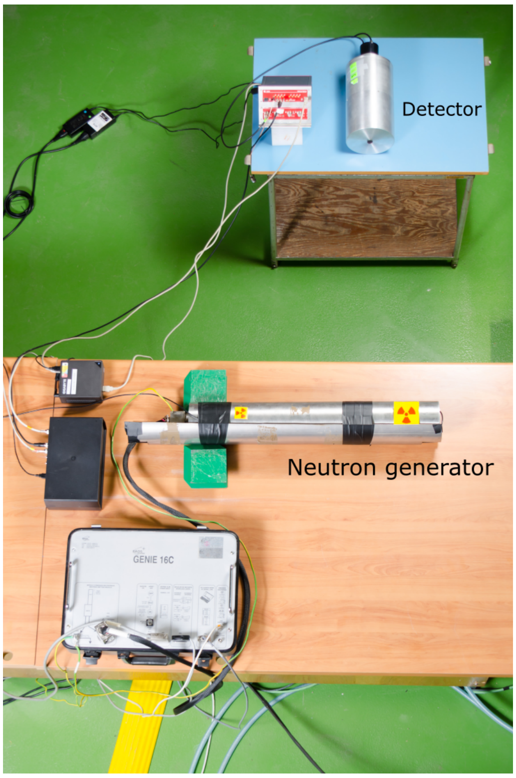

Fig. 2. The experimental setup for measurements of a neutron generator flux.

which blocked acquisition for 5 ms after each pulse in order to prevent detection of neutron capture γ-rays. The detector was placed 1 m from neutron generator tritium target and 83 cm above the concrete floor level. Each acquisition run last 10 minutes.

B. Monte Carlo simulations and data analysis

An MCNP v6.11 software package [5], [6] was used for investigation of the CaF_2 (n,α) reaction rate, expressed per one source neutron. We used average flux tally with FM4 card in order to calculate the reaction rate separately for Ca and F and run 1E8 histories. The obtained (n,α) rates for D-T and D-D neutron generators were presented in Tab. I. The uncertainties (expressed in percentage of the expected value) were taken from the MCNP calculation output, which are related only with the number of launched histories. The MCNP geometry model is shown in Fig. 3.

TABLE I
THE SIMULATED (n,α) REACTION RATES IN THE CaF_2 DETECTOR.

Neutron generator	(n,α) rate for ^{19}F
D-T ($E_n = 14\ MeV$)	$1.37 \times 10^{-5} \pm 0.0041\%$
D-D ($E_n = 2.5\ MeV$)	$6.46 \times 10^{-9} \pm 0.0037\%$

The (n,α) reaction rate for the simulated setup for D-T neutron generator is over four orders of magnitude greater than that for D-D one. This is in line with expectations, as the total (n,α) cross section for the 2.5 MeV neutrons is significantly

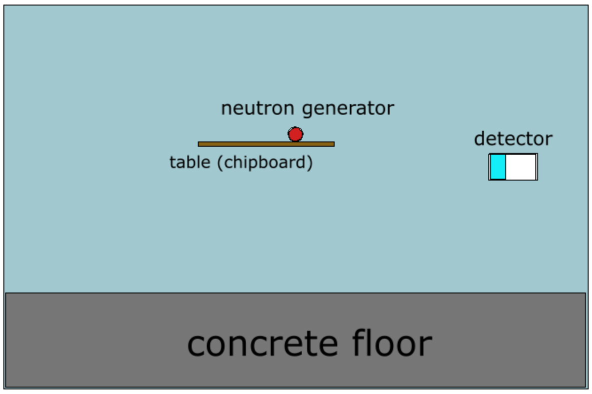

Fig. 3. Visualization of the MCNP geometry model.

lower than that for the D-T neutron generator. A D-D neutron generator would need to provide neutron flux of 2.5×10^{11} n/s to obtain the same statistics as the Sodern D-T generator at 90kV/40μA.

After the (n,α) reaction rate calculation, one need to calculate the number of generated β particle with the endpoint at 10.4 MeV. The intensity of this transition is 28%. In contrast, the intensity of the 4.3 MeV transition is 66%. The exemplary energy spectrum for neutron generator HV of 90 kV and current of 40 μA is presented in Fig. 4.

The β continuum with endpoints at 4.3 MeV and 10.4 MeV as well as 6.1 MeV γ-rays are clearly visible. The β continuum energy distribution was fitted with use of the following approximation (1):

$$\mathrm{N(E)} \sim \mathrm{C/c^5} \times \sqrt{(\mathrm{E^2 + 2Em_ec^2})} \times (\mathrm{Q - E})^2 \times (\mathrm{E + m_ec^2}), \tag{1}$$

where Q - beta endpoint, E - electron energy, m_e - electron mass, c - speed of light, C - constant.

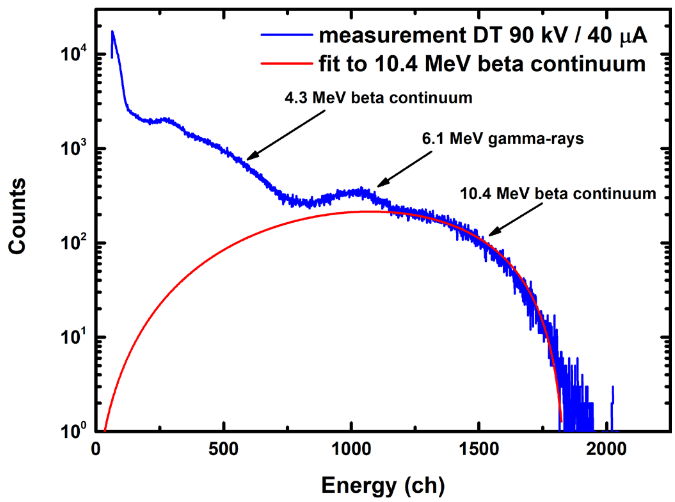

Fig. 4. The CaF_2 energy spectra after 600 s of exposition to 14 MeV neutrons from D-T neutron generator.

C. Results

The calculated neutron flux from the D-T neutron generator is presented in Fig. 5. The neutron flux per second was calculated with the following equation (2):

$$\phi_n = \frac{\int_0^Q N(E)dE}{R(n,\alpha) \times B}, \tag{2}$$

where $\int_0^Q N(E)dE$ is the integral over the beta energy spectrum distribution from 0 to the beta endpoint Q, $R(n,\alpha)$ is the (n,α) reaction rate on ^{19}F, obtained from the MCNP simulations, and B is the branching ratio of (n,α) for 10.4 MeV beta endpoint, equal to 0.28. Results were normalized to obtain the neutron flux per one second. The obtained results are in very good agreement with the neutron flux data provided by the neutron generator manufacturer.

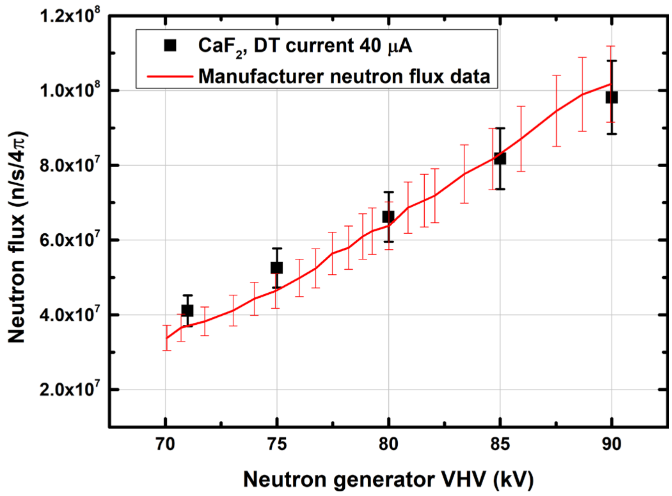

Fig. 5. Calculated neutron flux from Sodern Genie 16D neutron generator with use of CaF_2 detector and comparison with the manufacturer data.

III. Conclusions

We presented the approach for fast neutron flux estimation from D-T neutron generator with use of a single CaF_2 as the threshold activation detector. Main advantage of the method is the fact that neither activation foils nor High Purity Germanium (HPGe) detector is required to perform the flux calculation. After preceded MCNP simulations, required for (n,α) rate estimation at fixed detector and neutron generator positions, the neutron flux can be calculated in real time even for intense and short neutron pulses. This solution can be exceptionally useful in neutron generator test-bench facility, where at fixed positions one need to calculate the absolute neutron flux for any type of D-T neutron generator.

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