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Evaluation of radiation dose rates due to calibration and interrogation neutron sources while using the AWCC in horizontal configuration

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Abstract: *The Active Well Neutron Coincidence Counter (AWCC) is a device that can be used for nuclear safeguards purposes to measure different types of Nuclear Material (NM) in different modes. The detector uses neutron sources either for calibration (252Cf) or for interrogation of NM (AmLi). The radiation dose due to such sources should be estimated for safety reasons. In this work radiation dose due to neutron sources is estimated using the MCNP code. The results could be taken into consideration while measuring NM with the AWCC in open horizontal configuration.*
Keywords: *AWCC, Monte Carlo, Radiation Dose, Safeguards.*

I. Introduction

The Active Well Coincidence Counter (AWCC) is a device that could be widely used for NM verification. It was designed to non-destructively assay ^{235}U-bearing materials [1]. The counter is composed of two rings of neutron detectors embedded into a polyethylene moderator surrounding an assaying cavity. Lid and bottom end plugs serve as neutron shields for interrogation sources used to induce fissions in the assayed samples and also as reflectors to increase the counting efficiency by reducing neutron end losses. Fig. 1 shows a schematic diagram for the counter [2]. The operation of the AWCC is based on the detection of coincidence neutrons produced from induced fission of ^{235}U isotope caused by the interrogation neutron sources. The ability to measure samples in 4π-configuration, to change the volume of sample cavity and to operating the counter in different modes [3] allow using it to verify different types, sizes and categories of NM [4-7].

Normally, the counter is calibrated using a set of Nuclear Material Standards (NMS). Otherwise, cross calibration [8-10], absolute and semi-absolute methods could be considered [2, 11-14]. For semi-absolute cases the efficiency of the counter may need to be measured. A ^{252}Cf is usually used to measure the coincidence efficiency. In some cases, when sample dimensions exceed these of the cavity of the counter, the upper and lower lids are removed and replaced by another part; MTR insert (Fig. 1b), and the measurements are performed in horizontal configuration of the counter. In this configuration the counter is opened from both sides, which results in a higher radiation doses due to calibration and/or interrogation sources. Also, for relatively low masses of NM measuring times may be extended (up to three or four hours) to achieve accepted counting statistics which directly affect the radiation dose. Consequently, and for radiation safety purposes, the dose rate should be estimated.

In this work radiation dose rates due to calibration and interrogation neutron sources are estimated at different settings using the MCNP5 Monte Carlo Code.

II. Calculations

The general Monte Carlo Code, MCNP5, has the feature of calculating the radiation flux (gamma and neutrons) at a specified point using the point detector “F5” tally. The energy distribution of the neutron or gamma rays emitted from the source are identified using a distribution number “Dn” in the energy variable of the source definition card “SDEF”. The distribution itself is defined in the source information “SI” and source probability distribution “SP” cards. Finally conversion of the calculated flux at the specified point into dose is accomplished using the “flux-to-dose conversion factors listed in the DE/DF cards. The multiplier card “FMa” was used to modify the results such that the results are directly given in units of μSv/hr. The Table of neutron flux-to-dose rate conversion factor given in the MCNP manual was used [15,16].

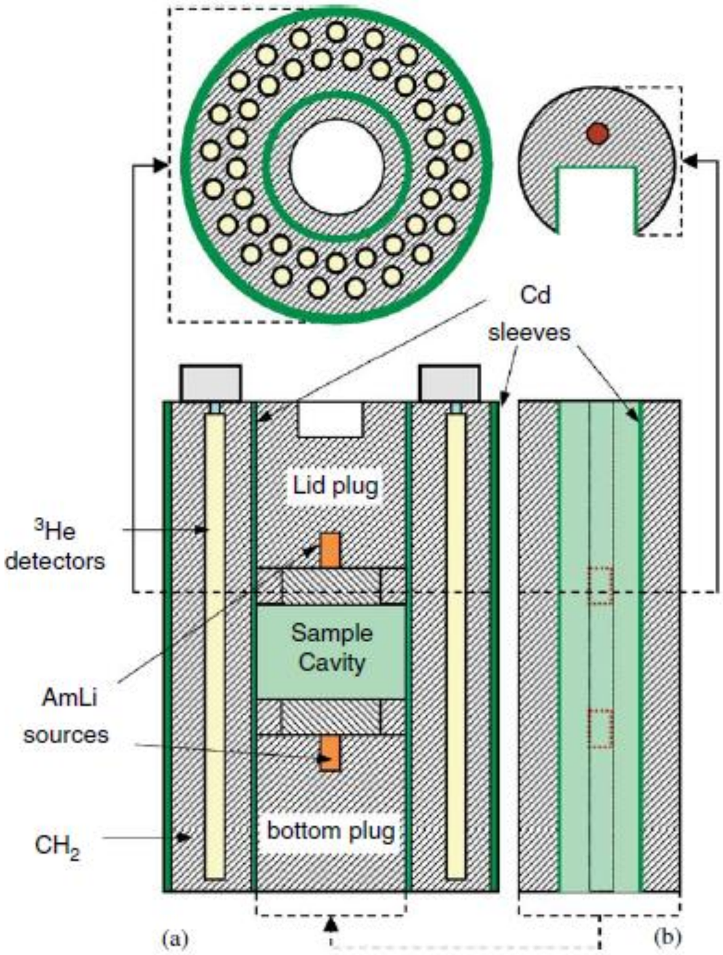


Fig. 1. Schematic diagram of the AWCC main components (a) and the MTR insert (b). [Ref. 2]

The detailed information used to construct the MNCP input files, including neutron sources specifications, were described in previous publications [2,14,17] and obtained mainly from the manufacturer [18,19,20].

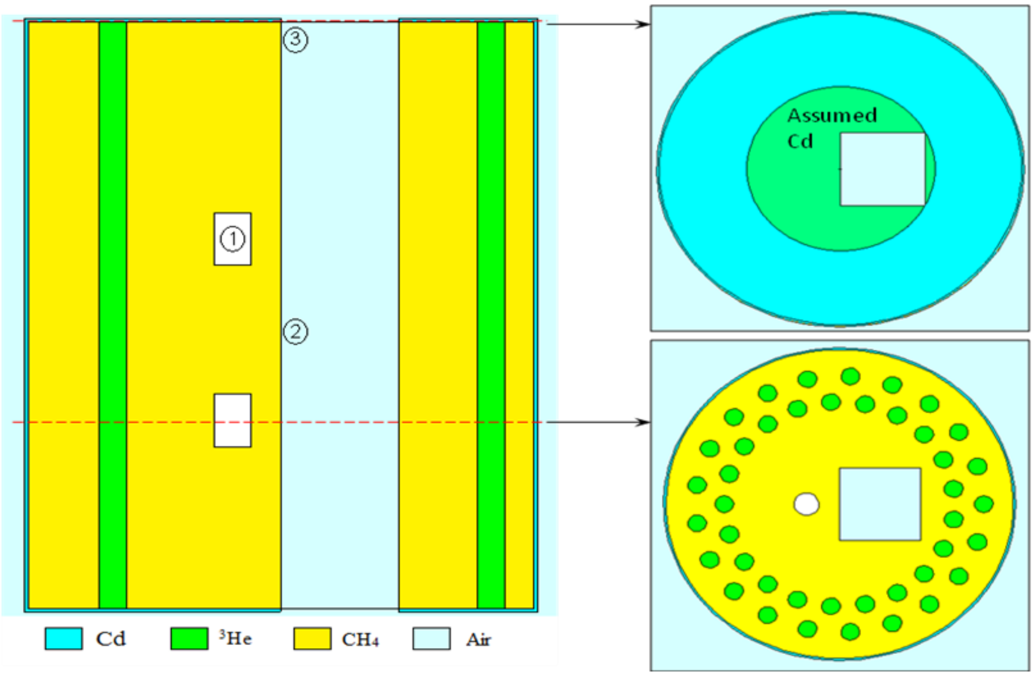
The system uses two 241AmO2–Li neutron sources (5×104 n/s emission rate each) to activate thermal fission in assayed samples. Each source is kept in a stainless-steel container. A tungsten shield is placed around each source to reduce γ-ray emission. The 241AmLi neutron source energy spectrum, in a numerical format used in the simulation process was obtained from reference [21]. Fig. 2 shows the MCNP longitudinal- and cross-sectional calculational model geometry for radiation dose evaluation. The location at which the AmLi is placed to interrogate the assayed sample is indicated on the figure as position①.

The ^{252}Cf isotope decays by alpha emission (96.91%) and spontaneous fission (3.09%) with an overall half-life of 2.645 years [3]. The neutron energy spectrum of ^{252}Cf used in this work was obtained from reference [22]. The calculations were performed at its initial activity (emits 5×10^4 n/s). The response of the counter to fission neutrons varies according to the location of the source inside it [1,2]. Therefore, the user must measure the neutron coincidence count rates due ^{252}Cf source along the axial and radial dimensions of the counter. The source is measured at equidistant locations along the MTR insert. The minimum evaluated radiation dose rate outside the counter is expected when the source is located at its center (position ② on Fig.2). While the maximum dose rate value is expected when the source is placed at the open end of the counter (position ③). The two positions corresponding to the maximum and minimum dose rates are the only locations considered in this work. Cadmium sheet were assumed to cover the end of the MTR insert as illustrated on the upper part of Fig. 2b.

III. Results and Discussion

An assessment of radiation dose due to gamma radiation from AmLi via experimental measurements showed that the dose rates in air with and without tungsten shield at 90 cm from the source were 0.15 and 21 μSv/h, respectively. Taking into consideration the location of AmLi sources inside the counter and the 0.3 mm Cd liner surrounding its body, the dose rates due to gamma radiation from AmLi sources could be ignored.

Radiation dose rate due to gamma rays from ^{252}Cf represent about 6% of that due to neutrons [3]. This small fraction was given at one meter form a californium source. Although this faction may vary for the current case, it is expected to remain small compared to that due to neutron and consequently is not considered in this work.



(a) (b)

Fig. 2. MCNP longitudinal (a) and cross sectional (b) Calculational model geometry for radiation dose evaluation.

The calculated dose rates outside the detector due to neutron from AmLi source is almost below the background. The maximum calculated dose rate (0.35 μSv/h) was found at the open end of the counter when the source is located at positions ① as indicated on Fig 2. The low dose values are due to the location of the source inside the polyethylene body of the counter for which most of the emitted neutrons are thermalized. The thermalized neutrons are then absorbed in the Cd liner surrounding the counter. Moreover, this type of interrogation source was mainly selected for many reasons including its relatively low average neutron energy (0.3 MeV) [1,3].

For the ^{252}Cf source, dose rates due to neutrons were calculated at 357 point detectors distributed in an XY matrix at the open end of the counter. The results are mapped in a contour diagram. The calculations were performed with the source at the center of the counter (position ②) and at the counter opening (position ③), as illustrated in Fig. 2. Dose rates at the side of the counter were not calculated since the designer took into consideration that the dose outside the closed counter is relatively very low [18].

Figures 3 is a contour diagram for the calculated dose rates in μSv/hr due to neutrons from the ^{252}Cf source located at the center of the counter (position ②). The maximum dose rate at the opening of the detector reaches about 6 μSv/h. A sharp decrease is observed at the positive and negative X-directions since most thermalized neutrons and absorbed in the Cd layer ring. At about 35 cm in the Y-direction the dose rate decreases to about 1 μSv/h.

The contour diagram for the calculated dose rates due to neutrons from the ^{252}Cf source located at the open end of the counter (position ③) is drawn in Fig. 4(a,b). Dose rate at the open end exceeds 100 μSv/h. It decreases till reaching about 5 μSv/h at 0.32 meters from the opening. It has to be emphasized that the illustrated dose rate values do not include the contribution of gamma rays and scattering effects.

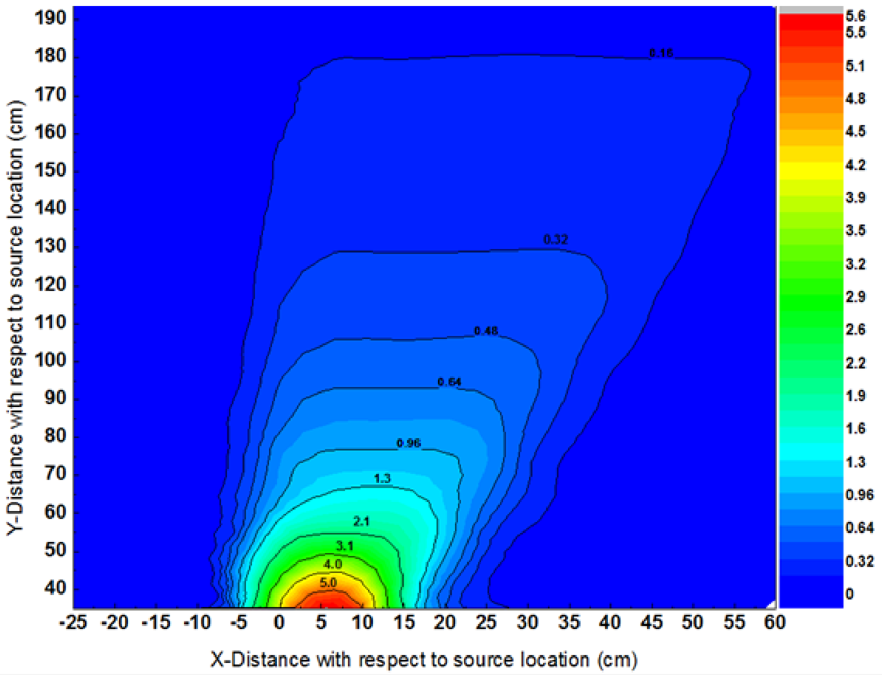
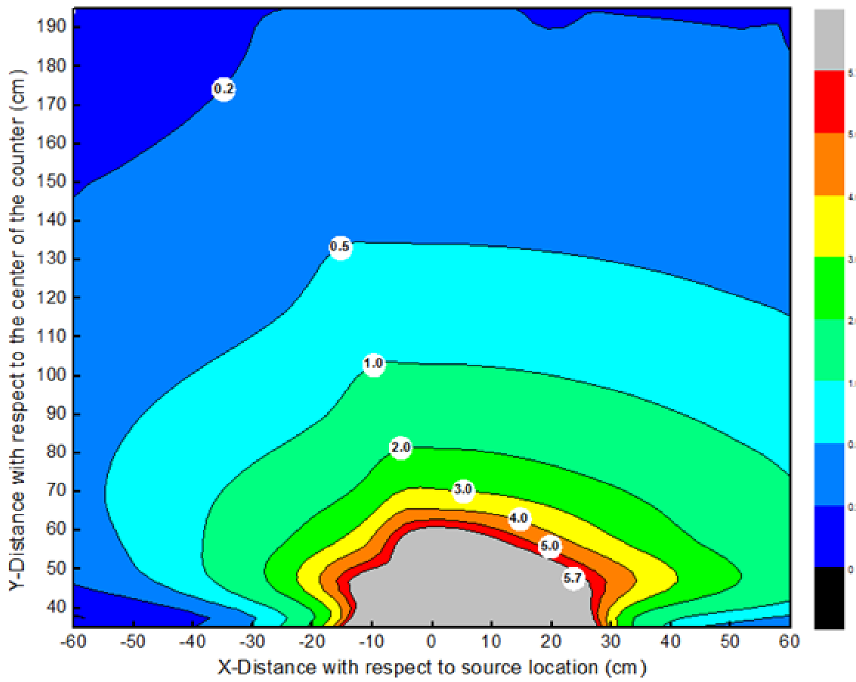


Fig. 3. Calculated dose rates in µSv/hr due to ^{252}Cf source located at the center of the counter (position ② on Fig.2).

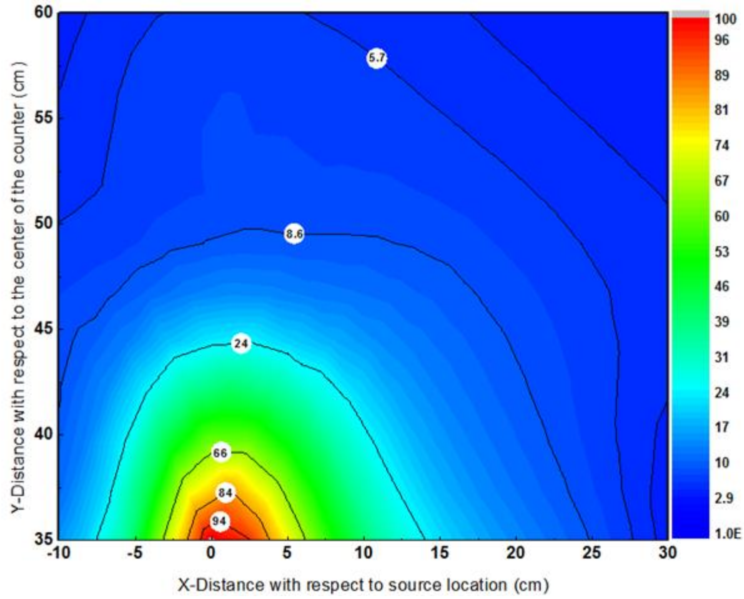
More accurate dose rate evaluations might be obtained if a more detailed ^{252}Cf γ-spectrum is considered [23]. Also, scattering effects from walls and ground should be taken into consideration whenever measurements are performed in narrow limited areas [24].

IV. Conclusion

Radiation dose rates due to calibration and interrogation neutron sources used with the AWCC were calculated for horizontal open configuration of the counter. The calculated values provide users with preliminary estimation of dose rates. More reliable values could be evaluated if radiation dose due to gamma radiation and scattering effects from walls and floor surrounding the counter are considered.



(a)



(b)

Fig. 4. Calculated dose rates in μSv/hr due to neutrons from ^{252}Cf source located at the open end of the counter (position ③) up to 1.6 meters from its open end (a) with enlarging the highest dose rate region (0.3 meter from the open end) (b).

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