Neutron coincidence counters for disarmament verification: experiments and modeling

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Abstract—A measurement campaign in support of nuclear disarmament verification was carried out in 2019 on the premises of the Belgian Nuclear Research Centre, SCK CEN. Well-characterized MOX sources with different types of shielding materials were used during the campaign. The relative content of Pu was up to 14% and the isotopic abundance of 239Pu was up to 93%. One of the technologies that were deployed was neutron coincidence counting with 3He based slab counters.

The obtained background and dead time corrected total and reals rates were used to determine the mass of the assayed sample by using the so-called point model equations. The obtained masses were systematically underestimated by typically few percent. However, outliers were observed.

In an effort to understand the obtained results, additional calculations were carried out to estimate the impact due to both the neutron emission by (α,n) reactions and spontaneous fission decay on the measured observables. The associated total and reals counts were obtained for various configurations and compared with the experimental data. The obtained results indicate the importance of (α,n) reactions when neutron multiplication is not negligible.

Keywords —Neutron coincidence counting; Monte Carlo modeling; Point model; Disarmament verification

I. INTRODUCTION

Within the framework of the activities of the International Partnership for Nuclear Disarmament Verification (IPNDV) [1], an international measurement campaign took place on the premises of the Belgian Nuclear Research Centre (SCK CEN) in the fall of 2019 [2]. The aim of the exercise was testing and comparing different non-destructive assay (NDA) techniques to be potentially used for the verification of nuclear material in the framework of the dismantlement of nuclear weapons. The measurements were carried out with well-characterized Mixed Oxide (MOX) fuel with different compositions, mass and shielding configurations. The gathered data would allow to draw conclusions on the capability of the considered NDA techniques to: verify the presence or absence of nuclear material originating from a nuclear weapon; distinguish weapon grade from civil grade nuclear material. One of the techniques of interest is neutron coincidence counting. Passive neutron counting is one of the techniques envisaged within the new START treaty for absence measurements [3].

In this work we report about the results obtained with 3He based slab counters for the considered MOX fuel assemblies. We first introduce neutron coincidence counting and how it can be used to determine the mass of material in the assayed material either by using the so-called point model approach or a Monte Carlo based approach. We will then present the measurement campaign by describing the used detectors and electronics and the measured MOX fuel assemblies. The analysis of the data with the proposed models follows, including a description of the used Monte Carlo model. A comparison of the obtained masses with the nominal ones is given as well as additional simulations that were carried out to interpret the results.

The uncertainties in this paper are quoted at a 68% confidence limit. We quote the uncertainties between parentheses as recommended in paragraph 7.2.2 of Ref. [4].

II. NEUTRON COINCIDENCE AND MULTIPLICITY COUNTING

The measurement of time correlated neutrons is a non-destructive technique that allows assaying material containing radionuclides undergoing spontaneous fission and has applications in the field of safeguards [5], waste and disposal [6] and disarmament verification [7]. Following a spontaneous or induced fission event, one or more neutrons are emitted simultaneously. Correlated neutron counting takes advantage of the simultaneous emission of fission neutrons to separate them from neutron emitted by other reactions such as (α,n) reactions where only one neutron is emitted.

Neutron coincidence counting allows estimating the mass of a sample containing material undergoing spontaneous fission...
provided that the radionuclide vector is known [8]. The technology measures the time correlation between events measured with neutron detectors. In addition to determine the total counts \( T \), the so-called \( R+A \) (reals plus accidentals) and \( A \) (accidentals) are determined from the number of detected neutrons in two time windows or gates that are opened every time a neutron is detected. In this work, the gates for the calculation of the \( R+A \) and \( A \) events have the same duration of 64 \( \mu \)s. The first gate opens 4.5 \( \mu \)s after the trigger and there is a time interval (delay) of 1000 \( \mu \)s between the gates. The concept is illustrated in Fig. 1., where the so-called Rossi-Alpha distribution (RAD), which represents the time distribution of neutron detections after an arbitrary neutron detection, is shown for a \(^{252}\text{Cf}\) source.

In addition to measuring the total counts in the two gates (neutron coincidence counting), the multiplicity distributions in the two gates can also be measured with dedicated systems; in this case one talks of neutron multiplicity counting. From the multiplicity distributions, one can derive the moments of the distribution. The moment of order zero represents the total counts, while the \( R+A \) and \( A \) terms corresponds to the moments of order one of the multiplicity distributions in the two gates, respectively.

The rate of reals or double \( D=(R+A)−A \) is a measure of the amount of material emitting neutrons by spontaneous fission. In this work, to determine the mass of the assayed sample we apply the so-called point model and a calibration approach, where the calibration factor is determined by Monte Carlo simulations.

\begin{equation}
T_{PM} = t\epsilon F_2 \nu_s (1 + \alpha)
\end{equation}

\begin{equation}
R_{PM} = t\epsilon^2 F_2 \nu_s \alpha
\end{equation}

where \( \epsilon \) is the detection efficiency, \( \alpha \) is the ratio between the neutron emission due to \((\alpha,n)\) reaction and spontaneous fission, \( F_2 \) the number of fission per seconds, \( \nu_s \) and \( \nu_c \) are the 1st and 2nd factorial moments of the distribution for the neutron emission through spontaneous fission, \( t \) the measurement time.

The corresponding experimental observables from the recorded pulse train are the total count rate \( T' \) and the measured double rate \( D' \) corrected by the \( f \) factor that accounts for the finite duration of the trigger gate and the presence of a predelay:

\begin{equation}
\frac{(R + A) - A}{tf} = \frac{D'}{f}
\end{equation}

Where

\begin{equation}
f = \int_{t_p}^{t_p+t_g} e^{-t/\tau} dt = e^{-t_p/\tau} \left(1 - e^{-t_g/\tau} \right)
\end{equation}

and \( t \) is the measurement time. The given expression for \( f \) holds only for an exponential RAD.

\section*{B. Monte Carlo based calibration approach}

The mass determination of the assayed sample can be also done by using a calibration approach, where the double rates are related to the mass of assayed sample (typically expressed as \(^{240}\text{Pu eq}\)) by a calibration coefficient [13]. This coefficient can be determined by using reference materials. If the geometry or chemical composition of the assayed samples differ from the ones of the reference materials correction factors need to be introduced, for example to account for the multiplication or hydrogen content [13]. The calibration coefficient can also be obtained from Monte Carlo simulations as explained in [8, 14]. One of the main advantages of this approach is that the impact of the geometry of the sample, the gate occupation factor \( f \) and, if the mass is known, the multiplication factor are accounted for in the simulations. Some knowledge about the composition of the source is however required in order to develop the model.

We have implemented a calibration approach where the number of doubles per fission event \( \nu_d \) is determined by Monte Carlo simulations [8, 12]. This term implicitly accounts for the occupation factor associated to the experimental conditions as well as the multiplicity distribution associated to the sample, the time response function of the moderator detector assembly is a pure exponential function with decay constant, and there are no dead time losses of signals. Other conditions also apply but we refer to the references for the full description.

\begin{figure}
\centering
\includegraphics[width=\textwidth]{ Rossi-alpha_distribution.png}
\caption{Rossi-alpha distribution from the measurement of a \(^{252}\text{Cf}\) source.}
\end{figure}

\section*{A. Point Model approach}

Starting from basic principles, theories were developed [9,10,11] that express the observables of neutron coincidence or multiplicity measurement as function of the mass of neutron emitting radionuclides and other parameters. This model, called the point model, can be applied in case the assayed sample has point geometry, no neutrons return from the detector to the
spontaneous fission in the source. The composition of the radionuclide source must be known to determine $\varepsilon_D$.

Therefore one can write:

$$D' = \varepsilon_D m \sum f_{m,j} F_{S,j}'$$

(5)

Where $F_{S,j}'$ is the spontaneous fission rate per gram and $f_{m,j}$ is the mass fraction of the radionuclide $j$, and $m$ mass of the sample.

Since the source composition may not be known at the moment when the Monte Carlo calculation is done, one could use a reference radionuclide (e.g. $^{240}\text{Pu}$) for which the $\varepsilon_{D,REF}$ is calculated.

One can write then

$$D' = \varepsilon_{D,REF} m_{REF,eq}' F_{S,REF}'$$

(6)

Where $m_{REF,eq}'$ represents the mass of the reference radionuclides yielding the same doubles rate.

From Equations 5 and 6 of $m_{REF,eq}'$ can be expressed as:

$$m_{REF,eq}' = m \frac{\varepsilon_D}{\varepsilon_{D,REF}} \left( \sum f_{m,j} \frac{F_{S,j}'}{F_{S,REF}'} \right)$$

(7)

The quantities $\varepsilon_{D,REF}$ and $\varepsilon_D$ depend on the multiplicity and neutron energy distributions associated to the source and can be very different e.g. for a $^{252}\text{Cf}$ or a $^{240}\text{Pu}$ source and not accounting for such a difference can be a source of systematic effects.

If we neglect the impact of the neutron energy distributions, in good approximation the number of doubles per fission event is proportional to the average number of pairs per fission generated by the sources and we can therefore approximate the efficiency ratio as the ratio of the average number of pairs per fission:

$$\frac{\varepsilon_D}{\varepsilon_{D,REF}} \approx \frac{\sum Q_j f_{m,j}}{Q_{REF}}$$

(8)

$$= \frac{\sum Q_j f_{m,j}}{\sum Q_j F_{S,j}' f_{m,j}}$$

$$= \frac{Q_{REF}}{F_{S,REF}'} \left( \sum f_{m,j} \frac{F_{S,j}'}{F_{S,REF}'} \right)$$

which provides a way to account for the actual radionuclide composition of the source and determine $m$ from Eq. 5 and using $\varepsilon_{D,REF}$ which may be easier to calculate.

III. EXPERIMENTS

A. Materials

The assayed material was unirradiated mixed oxide (MOX) fuel. The plutonium (Pu) is mixed with uranium (U), chemically in the form of oxide. The total plutonium content is up to 14%wt with a relative $^{239}\text{Pu}$ amount up to 96%wt. The MOX fuel pins were arranged in a compact hexagonal configuration as indicated in Fig. 2 to maximize the average density of the item to be assayed.

The measurement campaign allowed investigating the influence on the detector response due to the amount of nuclear material, the type of nuclear material, and the type of shielding material. For example, by performing measurements on samples with various amounts of plutonium mass and a fixed isotopic composition, one could study the sensitivity of the methods to the amount of nuclear material.

![Figure 2. Top view of a 19-pin configuration (Source: SCK CEN. Used by permission).](Image 358x441 to 504x625)
TABLE I

Neutron emission rates for the considered assemblies (values expressed in 10^3 neutron/s).

<table>
<thead>
<tr>
<th>Assembly ID</th>
<th>Spontaneous Fission</th>
<th>(α,n) reaction</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>U</td>
<td>Pu</td>
</tr>
<tr>
<td>79-19</td>
<td>51</td>
<td>0.1</td>
</tr>
<tr>
<td>96-19</td>
<td>21</td>
<td>0.1</td>
</tr>
<tr>
<td>62-1</td>
<td>23</td>
<td>&lt;0.1</td>
</tr>
<tr>
<td>62-19</td>
<td>439</td>
<td>0.1</td>
</tr>
<tr>
<td>62-61</td>
<td>1,410</td>
<td>0.3</td>
</tr>
</tbody>
</table>

B. Detector and setup

Two Canberra WM3400 slab counters [15] were deployed. A picture of the detectors during the measurement of MOX assembly is shown in Figure 3.

Each WM3400 counter measures 104 cm (height) × 41 cm (width) × 16 cm (depth) and has an approximate weight of 50 kg. The WM3400 counter contains six 3He tubes (active length of 91.44 cm and diameter of 2.54 cm) which are positioned in-line and embedded in a polyethylene slab. The polyethylene slab is then surrounded by 1-mm Cd sheet that covers five sides of the slab. During a measurement the WM3400 should be placed such that the side without the Cd cover is facing the sample being measured.

A dedicated electronic system [16] was used that includes a dedicated preamp for each 3He tube, the possibility to adjust for gain differences between the 3He tubes, and an optimised shaping time for 3He detectors. The logical OR of the signals coming from both slab counters was processed with a MCA-527 multi-channel analyzer equipped with time stamp processing firmware [17]. By considering the two detectors as a whole one limits the impact of possible asymmetrical detector arrangements. Such a system allowed saving the time stamps of the neutron detections for off-line data with a dedicated software to determine T, R+A, and A events, the multiplicity distribution, as well as to remove system instabilities due to high voltage and fluctuations or interferences.

C. Measured samples

All the five assemblies (96–19, 79–19, 62–1, 62–19, 62–61) were measured. All measurements were carried out with 1.1 mm Cd around the fuel assembly. For fuel types 79–19 and 96–19 the measurements were also carried out without Cd. The fuel type 96–19 was also measured with other two configurations: the first one with 5 cm CH2 around the assembly and 1.1 mm thick Cd sheet on the detector (shown in Figure 3) and the second one with 5 mm Pb around the assembly and a 1.1 mm thick Cd sheet on the detector. Measurements were done at two distances 26 cm and 96 cm from the outer surface of the assembly. The measurement time depends on the neutron emission of the sample and was either 1800 s or 3600 s. In addition to measurements with MOX fuel assemblies, also background measurements were carried out.

IV. DATA ANALYSIS

The background subtracted and dead time corrected totals and reals rate values for the considered assemblies are shown in Table II. The uncertainty due to counting statistics is also given.

The time stamps recorded with the MCA527 were analyzed by means of Python [18] scripts, in the Spyder environment version 4 [19]. The scripts first carried out stability checks of the data by monitoring the total count rate over 10 seconds interval. A dead-time correction was carried out by implementing the dead time correction algorithm with paralyzable model as described in Dytlewski [20] and Vincent [21]. The dead-time was estimated to be 1.0 µs with a corresponding maximum dead time correction for the case 62-61 at 26 cm, where the totals correction was about 11 % and the reals correction was about 34 %, respectively.

Then the data were analyzed to determine the totals and the reals rate for each measurement. The background subtraction was then carried out and the net total and net real count rates, indicated with T′ and D′ respectively, were determined.
The data in Table II for the assemblies 79-19 and 96-19, indicate that both the totals and reals rate are almost insensitive to the Cd absorber indicating that the counted neutrons are the mainly due fast neutrons emerging from the assembly. For the 96-19 assembly, both totals and reals were almost insensitive to Cd+Pb (5% reduction on the totals), with PE+Cd the totals were attenuated by 30% and the reals by 43%.

In all the cases the uncorrelated background was relatively high with the ratio A/(R+A) ranging from 88.4% (96-19 with PE+Cd shielding) to 99.8% (62-61 with Cd, at 1 m distance). In the worst case the reals rate could be determined with a relative uncertainty of 9%. This is in line with the data of Table I, from which one would expect that for 96-19 and 79-19 assemblies a lesser impact due to (α,n) reaction when compared to the 62 type assemblies.

The obtained data were used to determine the mass of the assayed sample by applying two methodologies, one based on the point model approximation and one based on Monte Carlo modeling. The nuclear data used in the data analysis such as decay constants, (α,n) emission probabilities, spontaneous fission half-lives, factorial moments of the neutron emission distribution by spontaneous fission used in the data analysis were taken from a compilation of different sources e.g. [22-31], with atomic masses taken from [32].

### Table II

<table>
<thead>
<tr>
<th>Assembly Shielding</th>
<th>d (cm)</th>
<th>T' (s⁻¹)</th>
<th>T_{DTCF} (%)</th>
<th>D' (s⁻¹)</th>
<th>D_{DTCF} (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>62-61 Cd</td>
<td>96</td>
<td>30435 (5)</td>
<td>3.1</td>
<td>104.6</td>
<td>93 (93)</td>
</tr>
<tr>
<td>62-1 Cd</td>
<td>96</td>
<td>431.8 (2)</td>
<td>0.1</td>
<td>1.1</td>
<td>1 (1)</td>
</tr>
<tr>
<td>62-61 Cd</td>
<td>26</td>
<td>100089 (13)</td>
<td>10.6</td>
<td>1115 (46)</td>
<td>34.3</td>
</tr>
<tr>
<td>62-19 Cd</td>
<td>26</td>
<td>30415 (5)</td>
<td>3.1</td>
<td>303.9</td>
<td>93 (93)</td>
</tr>
<tr>
<td>62-19 Cd</td>
<td>96</td>
<td>8848.8 (16)</td>
<td>0.9</td>
<td>23.9</td>
<td>17 (17)</td>
</tr>
<tr>
<td>62-1 Cd</td>
<td>26</td>
<td>1505.5 (10)</td>
<td>0.2</td>
<td>14.5</td>
<td>5 (5)</td>
</tr>
<tr>
<td>96-19 Pb,Cd*</td>
<td>26</td>
<td>1480.3 (0.2)</td>
<td>0.2</td>
<td>21.7</td>
<td>5 (0.6)</td>
</tr>
<tr>
<td>96-19 PE,Cd*</td>
<td>26</td>
<td>1044.5 (6)</td>
<td>0.1</td>
<td>12.4</td>
<td>2 (0.4)</td>
</tr>
<tr>
<td>96-19 Cd</td>
<td>26</td>
<td>1558.0 (10)</td>
<td>0.2</td>
<td>21.8</td>
<td>5 (0.5)</td>
</tr>
<tr>
<td>96-19</td>
<td>26</td>
<td>1558.4 (7)</td>
<td>0.2</td>
<td>21.7</td>
<td>3 (0.6)</td>
</tr>
<tr>
<td>79-19</td>
<td>26</td>
<td>3772.8 (15)</td>
<td>0.4</td>
<td>55.7</td>
<td>11 (1.2)</td>
</tr>
<tr>
<td>79-19 Cd</td>
<td>26</td>
<td>3731.5 (12)</td>
<td>0.4</td>
<td>55.7</td>
<td>8 (1.2)</td>
</tr>
</tbody>
</table>

The obtained ratio between the calculated mass of heavy metal and the nominal one from the fuel specification for different cases and as a function of the mass of heavy metal is shown in Fig. 4.

The error bars represents the 68% confidence limit due to counting statistics and include a 1% uncertainty of the gate occupation factor f. The results show an average underestimation of the mass by about 15%. The fact that the multiplication factor is not accounted for can explain the observed bias as well as the fact that the conditions to apply the point model were not fully met. The main outlier is represented by the case with polyethylene shielding for which a larger correction for that multiplication in the assembly is needed when solving the point model equations.

### Figure 4

Ratio between the mass obtained from the point model approach and nominal masses as a function of the mass for the different measurement configurations.

B. Monte Carlo based calibration approach

The Monte Carlo code MCNP6 [33] was used to carry out a full scale modelling of the measurement setup. Following the results of previous modeling efforts [8], the model included the detectors and the fuel assembly, shielding, as well as the walls, ceiling and floor of room in which the measurements took place. Since the specifications of detectors and assemblies were very well known, the geometry and material composition could be modelled with a high degree of fidelity. A cross section of an example of a modeled geometry is shown in Fig. 5.
MCNP6 was run in the so-called analog mode i.e. tracking event-by-event the interactions of neutrons rather than their average behaviors. A tally of the type F8 with the CAP special treatment [34] was used. With such a tally the neutron multiplicity is scored in two time intervals both of duration of 64 \mu s corresponding to the experimental ones: the first one called \( R+A \) gate and starting after 4.5 \mu s from the spontaneous fission event; the second one called \( A \) gate and starting 1,000 \mu s after the end of \( R+A \) gate. The simulations implicitly account for the pre-delay and finite duration of the gate and therefore a correction for the \( f \) factor, such as the one used in the point model approach, is not necessary. The simulations only account for neutrons emitted by spontaneous fission and therefore do not account for reals induced by \((\alpha,n)\) reactions which are present only in the presence of a multiplying sample. A source term corresponding to the spontaneous fission of the source radionuclides was used. The nuclear data library JEFF 3.3 was used.

The obtained masses \( m_n \) with the previously outlined procedures were compared with the nominal masses \( m_n \) of the measured samples are shown in Figure 6. The obtained ratios \( m_n/m_m \) for the considered cases are given in Table III for the point model and Monte Carlo approaches. The quoted uncertainties result from counting statistics on the net reals rate and on the reals per spontaneous fission event from Monte Carlo simulations, which varied between 0.2% and 0.4%, and have been obtained by first-order Taylor expansion-based error propagation.

With the exception of the data for the 62-19 and 62-61 assemblies measured at 26 cm distance, are overestimated between 2% and 24%. As previously stressed, the Monte Carlo simulation do not account for the effect of \((\alpha,n)\) reactions. By not accounting for \((\alpha,n)\) reactions, the share of the reals due to \((\alpha,n)\) reactions are attributed to spontaneous events and this may result in a systematic effect on the estimated mass. We notice that for the two outliers for the 62-19 and 62-61 assembly at 26 cm distance fuel for which the dead time correction of the real was higher than for most of the other cases.

In an effort to understand the role of \((\alpha,n)\) reactions on the obtained results additional MC calculations were carried out.

### TABLE III

<table>
<thead>
<tr>
<th>Assembly</th>
<th>Shielding</th>
<th>( d )</th>
<th>( m_n/m_m ) (Point Model)</th>
<th>( m_n/m_m ) (MC)</th>
</tr>
</thead>
<tbody>
<tr>
<td>62-61</td>
<td>Cd</td>
<td>96</td>
<td>0.76(7)</td>
<td>1.24(11)</td>
</tr>
<tr>
<td>62-1</td>
<td>Cd</td>
<td>96</td>
<td>0.92(6)</td>
<td>1.21(7)</td>
</tr>
<tr>
<td>62-61</td>
<td>Cd</td>
<td>26</td>
<td>0.77(3)</td>
<td>0.87(4)</td>
</tr>
<tr>
<td>62-19</td>
<td>Cd</td>
<td>26</td>
<td>0.84(3)</td>
<td>0.93(3)</td>
</tr>
<tr>
<td>62-19</td>
<td>Pb,Cd*</td>
<td>96</td>
<td>0.89(7)</td>
<td>1.13(8)</td>
</tr>
<tr>
<td>62-19</td>
<td>Pb,Cd*</td>
<td>26</td>
<td>0.82(3)</td>
<td>1.04(3)</td>
</tr>
<tr>
<td>96-19</td>
<td>PE,Cd*</td>
<td>26</td>
<td>0.70(2)</td>
<td>1.13(2)</td>
</tr>
<tr>
<td>96-19</td>
<td>Cd</td>
<td>26</td>
<td>0.88(2)</td>
<td>1.05(2)</td>
</tr>
<tr>
<td>96-19</td>
<td>---</td>
<td>26</td>
<td>0.89(2)</td>
<td>1.05(2)</td>
</tr>
<tr>
<td>79-19</td>
<td>---</td>
<td>26</td>
<td>0.92(2)</td>
<td>1.07(2)</td>
</tr>
<tr>
<td>79-19</td>
<td>Cd</td>
<td>26</td>
<td>0.90(2)</td>
<td>1.06(2)</td>
</tr>
</tbody>
</table>

C. Impact of the \((\alpha,n)\) reactions

The Monte Carlo approach proposed in previous section assumes that only spontaneous fission affect the observed reals rate. As indicated in Table in addition to spontaneous fission a non negligible amount of neutron are emitted by \((\alpha,n)\) reactions. In absence of multiplication, due to the fact that the emission of such neutrons is not time correlated, they do not contribute to the net real rate \( D' \). Given the mass of fissile material, the assumption of absence of multiplication is questionable and therefore we carried out additional simulations to account also for the \((\alpha,n)\) component.
The energy dependency of the neutrons stemming from the \((\alpha,n)\) reaction was deduced from the code SOURCE-4C [35] based on the radionuclide composition of the source. The results of the MC simulations are total and reals rate per fission event or per neutron emitted by \((\alpha,n)\) reaction. By multiplying these data by the corresponding source strength we obtain the total and reals rate due to spontaneous fission and \((\alpha,n)\), respectively.

We then determine the ratio between the dead time and background corrected experimental data and the simulated reals rate in Table IV. This ratio should be compared with the mass ratio \(m_r/m_s\) from MC calibration in Table III. The calculated mass \(m_r\) was obtained from Eq. 5 by using the \(E_D\) term, that is the number of simulated reals per spontaneous fission event, and the experimental data. The change in ratio due to the presence of \((\alpha,n)\) is an indication of the impact of accounting for the \((\alpha,n)\) in the mass determination.

We observe that accounting for \((\alpha,n)\) systematically reduces the ratio; this is due to the fact that with the first approach part of the reals were ascribed to spontaneous fission source and this resulted in an overestimation of the mass. For most of the cases there is an agreement within the uncertainties between calculated and measured values. An overestimation is still present on all the measurement data at 96 cm distance. The data for the 62-61 and 62-19 at 26 cm distance are clearly underestimated possibly due to insufficient dead time correction. The data for the 62-61 assembly at 96 cm probably suffer from both effects and have a relatively large uncertainty.

The quoted uncertainty accounts for the statistic uncertainty only. For completeness we also include the ratio between the experimental and calculated totals rates, the share of reals due to spontaneous fission \((D_{SP}/D)\) and multiplication factor \(M\) obtained from MC simulations.

### TABLE IV

<table>
<thead>
<tr>
<th>Assembly/Shielding</th>
<th>(d) (\text{cm})</th>
<th>(D'_{\text{E/C}})</th>
<th>(T'_{\text{E/C}})</th>
<th>(D_{SP}/D)</th>
<th>(M)</th>
</tr>
</thead>
<tbody>
<tr>
<td>62-61 Cd</td>
<td>96</td>
<td>1.07(10)</td>
<td>1.02</td>
<td>0.86</td>
<td>1.09</td>
</tr>
<tr>
<td>62-1 Cd</td>
<td>96</td>
<td>1.17(7)</td>
<td>1.05</td>
<td>0.97</td>
<td>1.01</td>
</tr>
<tr>
<td>62-61 Cd</td>
<td>26</td>
<td>0.75(3)</td>
<td>0.94</td>
<td>0.86</td>
<td>1.09</td>
</tr>
<tr>
<td>62-19 Cd</td>
<td>26</td>
<td>0.85(3)</td>
<td>0.99</td>
<td>0.91</td>
<td>1.05</td>
</tr>
<tr>
<td>62-19 Cd</td>
<td>26</td>
<td>1.05(8)</td>
<td>1.04</td>
<td>0.92</td>
<td>1.05</td>
</tr>
<tr>
<td>62-1 Cd</td>
<td>26</td>
<td>1.01(3)</td>
<td>1.00</td>
<td>0.97</td>
<td>1.01</td>
</tr>
<tr>
<td>96-19 Pb,Cd*</td>
<td>26</td>
<td>0.98(2)</td>
<td>0.94</td>
<td>0.95</td>
<td>1.04</td>
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<td>0.99(2)</td>
<td>0.98</td>
<td>0.95</td>
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<tr>
<td>96-19 Cd</td>
<td>26</td>
<td>0.99(2)</td>
<td>0.98</td>
<td>0.95</td>
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<td>1.01(2)</td>
<td>1.02</td>
<td>0.95</td>
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<td>79-19 Cd</td>
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<td>1.01(2)</td>
<td>1.01</td>
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V. CONCLUSIONS

We carried neutron coincidence measurements with \(^{3}\)He based slab counters in the framework of a measurement campaign in support of nuclear disarmament verification with well-characterized MOX sources.

The obtained background and dead time corrected total and reals rates were used to determine the mass of the assayed sample by using the so-called point model equations in absence of multiplicity. The obtained masses were systematically underestimated by 15% in average.

A calibration approach based on Monte Carlo simulations where the mass is estimated from the calculated number of reals per spontaneous fission event was also used. The obtained masses were slightly overestimated and few outliers, up to 20%, were observed.

In an effort to better understand the obtained results, additional calculations were carried out to estimate the impact due to both the neutron emission by \((\alpha,n)\) reactions and spontaneous fission decay on the measured observables. The associated total and reals counts were obtained for various configurations and compared with the experimental data. The obtained results indicate a very good agreement between experimental and calculated data. The importance of \((\alpha,n)\) reactions when neutron multiplication is not negligible. The remaining discrepancy are most likely due to the dead time correction.

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