ID: 301

Absolute Method for Characterization of Disused Depleted Uranium Containers

H. I. Khedr, Z. Ahmed, A. H. Osman
Egyptian Nuclear and Radiological Regulatory Authority
Email: hany_khedr@yahoo.com

ABSTRACT

- Depleted Uranium (DU) is encountered in various applications as radiation shielding in industrial containers used to transport radioactive materials.
- Disused containers of depleted uranium used to transport ^{192}Ir sources have been characterized.
- The characterization was carried out by gamma spectrometry (NaI detector) in combination with the MCNP method. The calculated absolute efficiencies have been used with experimental results to estimate the masses of ^{235}U and ^{238}U.

BACKGROUND

- The Egyptian System of Accounting for and Control of NM (ESAC) provides technical support for Location Outside Facilities (LOFs).
- One of these technical supports is a characterization of disused DU containers using different detection systems.
- These technical supports helps the ESAC to submit complete and correct declarations.
- In principle, any of the gamma rays from DU can be used to determine the mass of the isotope that produces them.
- The MCNP particularly useful for complex problems that cannot be modeled by computer codes that use deterministic methods.

2. MEASUREMENT SETUP

Five containers used to transfer ^{192}Ir radioactive source made of DU have been measured by using NDA technique. The container has cylindrical shape with 3.9 cm diameter and 5.5 cm height. Four containers have been measured with its aluminum cladding which has thickness 1 mm and only one sample take off its aluminum cladding. Figure (1) shows the outer shape of sample with its and without cladding.

(a) (b)

Fig. 1: The outer shape of DU container sample (#1) (a) with and (b) without its cladding.

The used gamma spectrometer in this work is a portable scintillation inspector 1000 (model IPROS-3 serial number 13000115). The gamma spectrometer has a NaI crystal with dimensions (76.2 x76.2 mm) and an Aluminum (Al) housing of 1mm. All DU samples were measured in such a way that its axis of symmetry is perpendicular to the extended axis of symmetry of the detector. The distance separate between the sample surface and the Al cap of the detector was adjusted or chose in such a way that errors due to electronic losses were minimized (dead time did not exceed 2 %) and equal 18 cm. Also, the measuring life time was optimized to achieve good statistics (statistical errors are always kept below 1%).

Fig. 2: The calculated ^{235}U mass value in comparison with it declared value

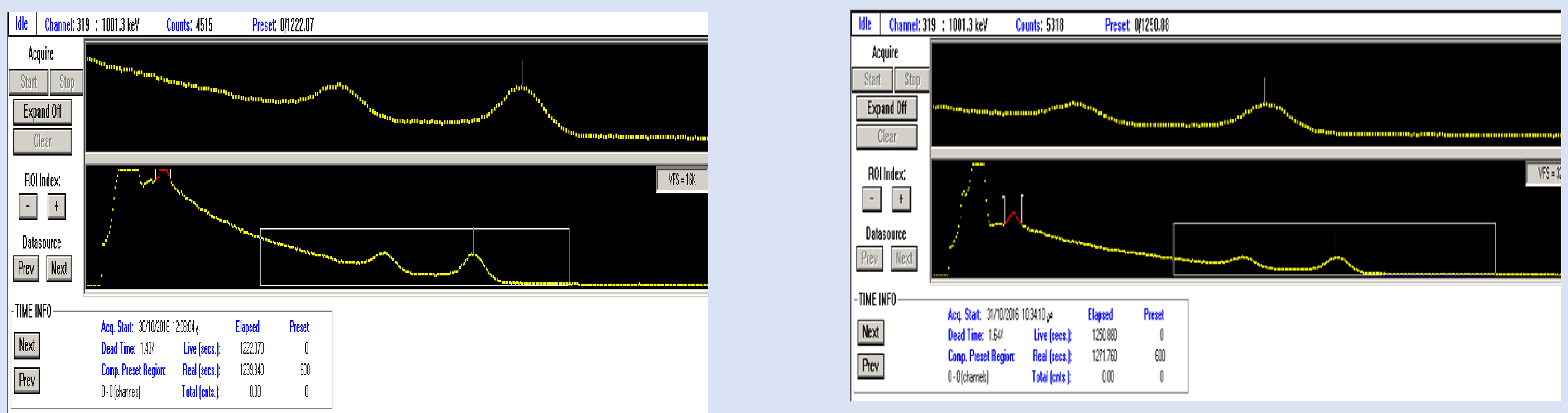

Fig. 3: The spectrum of the DU container (sample #1) without its aluminum cladding.

RESULTS AND DISCUSSION

- THE CALCULATED AND DECLARED VALUES OF 235U MASS CONTENT IN THE FIVE NM MEASURED SAMPLES.

Sample #	Calculated ^{235}U mass (g) $\pm\,\delta_M \times 10^{-2}$		Declared ^{235}U mass (g)	Accuracy%
1	3.25509	±6.713	3.3±0.001	1.36101
2	3.35172	±7.146		-1.56738
3	3.35574	±7.188		-1.68912
4	3.24182	±6.879		1.76313
5	3.26137	±6.564		1.17066

- THE CALCULATED AND DECLARED VALUES OF 238U MASS CONTENT IN THE FIVE NM MEASURED SAMPLES.

Sample #	Calculated ^{238}U mass (g) $\pm\,\delta_M$		Declared ^{238}U mass (g)	Accuracy%
1	1007.39	±1.536	1026.01±0.001	1.81469
2	1036.27	±1.596		-1.00026
3	1039.63	±1.600		-1.32709
4	1017.98	±1.552		0.78274
5	1006.03	±1.451		1.94724

- It's clear from the above tables that an agreement between calculated and declared results for ^{235}U and ^{238}U masses with accuracy ranged from 1.76% to -1.68% for ^{235}U and ranged from 1.94% to -1.33% for ^{238}U.

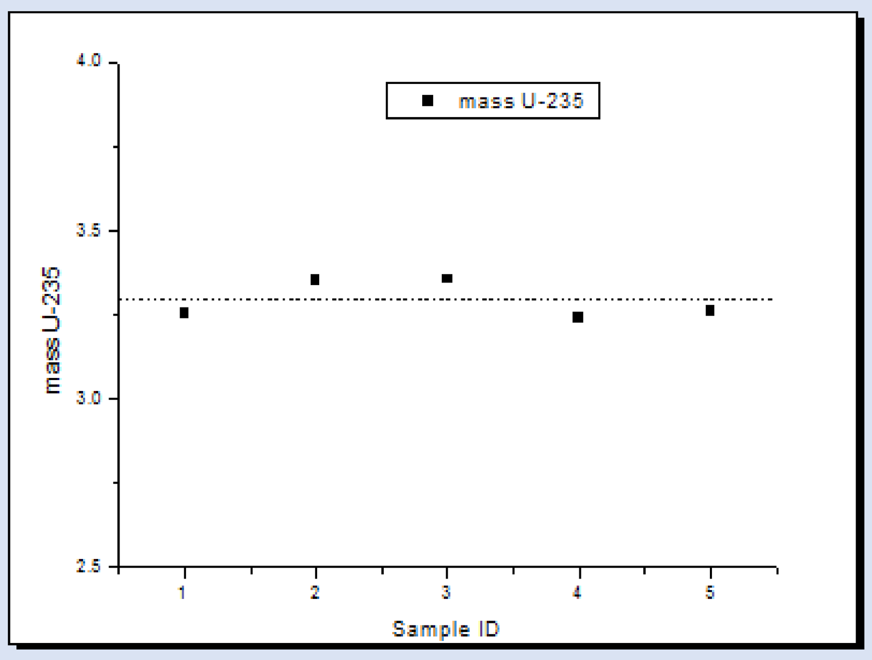

Fig4(a): The calculated ^{235}U mass value in comparison with it declared value

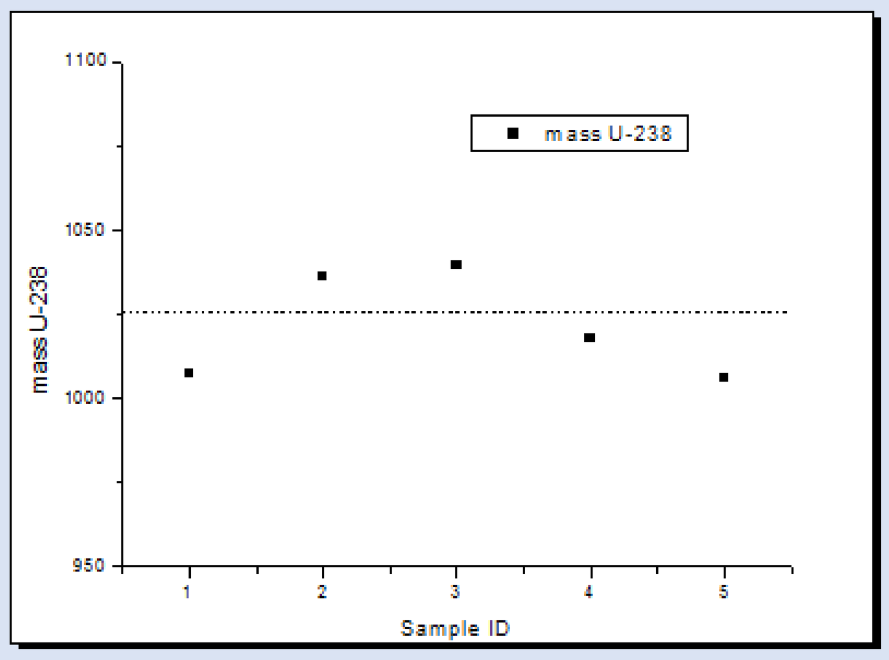

Fig4(b): The calculated ^{238}U mass value in comparison with it declared value

CONCLUSION

- Full characterization to disused DU transfer containers has been done for the nuclear material accounting purpose by using absolute method. The MCNP5 has been used for modeling the measurement setup in order to calculate the absolute efficiency of the detector. 235U and 238U masses have been calculated and compared with the declared values. The obtained results agreed with the declared data.
- The applied method could be saved the nuclear inspector time and saved him from radiation exposure during inspections activities. This work could be applied in the nuclear safeguards inspection on facility and location outside facility.

REFERENCES

- Inspector 1000 "digital hand- held MCA" user's manual
- X-5 Monte Carlo Team" MCNP — A General Monte Carlo N-Particle Transport Code, Version 5" April 24, 2003.