EXAMINATION OF IRRADIATED FUEL ELEMENTS USING GAMMA SCANNING TECHNIQUE

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ABSTRACT

The purpose of this paper is to validate the gamma scanning technique used to calculate the activity of gamma fission products from CANDU/TRIGA irradiated fuel elements. After a short presentation of the equipments used and their characteristics, the paper describes the calibration technique for the devices and how computed tomography reconstruction is done. Following the previously mentioned steps is possible to obtain the axial and radial profiles and the computed tomography reconstruction for calibration sources and for the irradiated fuel elements. The results are used to validate the gamma scanning techniques as a non-destructive examination method. The gamma scanning techniques will be used to: identify the fission products in the irradiated CANDU/TRIGA fuel elements, construct the axial and radial distributions of fission products, get the distribution in cross section through computed tomography reconstruction, and determine the nuclei number and the fission products activity of the irradiated CANDU/TRIGA fuel elements.

Key words: gamma scanning, computed tomography reconstruction

Introduction

In nuclear installations, the gamma spectrometric systems are used in a range of monitoring processes – analyzing - activity calculations for gamma emitting radionuclides. The gamma scanning technique it is a specific gamma spectrometry application which aims to: qualitatively and quantitatively analyze the fission products present in the irradiated CANDU/TRIGA fuel elements; achieving axial and radial profiles of the fission products; achieving distribution in cross section through computed tomography reconstruction; calculate the total number of nuclei and the activity of the fission products of interest from the irradiated fuel elements. The gamma scanning technique is a non-destructive post-irradiation examination method.

Experimental set-up

▪ Measuring Equipment Presentation

In this paper we used three different measuring systems (Fig. 1) in order to study how the technical characteristics of measuring systems influence the accuracy of measurement results. Determination of axial and radial profiles requires a very large number of measurements. In these circumstances, accuracies of measurement are essential for measuring time. The main technical characteristics of the three detectors are shown in Table 1.

Fig. 1. *PGT – IGC 12(1986)/ORTEC NOMAD PLUS (1998)/ORTEC GEM 30 P4-70 (2014) - HP Ge Detectors*

Table 1. *HP Ge detectors characteristics [1]*

Parameter	PGT – IGC 12 (1986)	ORTEC NOMAD PLUS (1998)	ORTEC GEM 30 P4-70 FACELIFT (2016)
FWHM 1.33 MeV 60Co	1.75 keV	1.85 keV	1.83 keV
Peak to Compton Ratio 60Co	37:1	55:1	60:1
EFF Relative 1.33 MeV 60Co	12 %	25 %	30 %
HV	+2500 V	+ 2500 V	+ 4800 V
Coolant	Dewar LN2 30 l	Dewar LN2 5 l	Electrical

Spectra acquisition and analysis was performed with the aid of ORTEC GAMMA VISION software, version 7.01. This software allows for the calibration, acquisition and spectra analysis obtained by each of the detector previously mentioned and can accommodate a large range of multichannel analyzers and ORTEC digital spectrometers systems. Also, it can utilize and store large data libraries for the radionuclide's of interest, it can achieve, store and use energy and efficiency calibrations, simple and the complex analysis can be performed (for example, several spectra simultaneously). [2]

Fig. 2. *Universal Examination Machine with command control desk (U.E.M.)*

Along with measuring equipment a mobile collimation system is operating with adjustable slit (there are 3 possibilities: width = 0.1, 0.25, 0.5 mm, length = 50 mm) which is mounted in the cell wall and may be positioned horizontally or vertically, and a U.E.M., located inside the hot cell (for example the photo 2 in Fig. 2) which, actuated by a control desk Acb - France (Fig. 2), is moving the sample under examination of the four axes (X, Y, Z, θ). For mounting the irradiated fuel element in the U.E.M. is necessary both to robotic arms pair and to crane which are handled/positioned related devices.

- **Measuring Equipment Calibration Procedure and Calibration Sources Used**

To perform the energy calibration were used three calibration point sources according to the calibration certificates (Fig. 3).

Fig. 3. *Calibration point sources*

The efficiency calibration of the spectrometric measuring systems was performed using a standard 137Cs source produced by CEA – France, having a similar geometry with the geometry of the irradiated fuel element of interest. First, an axial profile was acquired for the standard 137Cs source using a collimation system with adjustable slit in horizontally position, then using a dedicated computer code was determined the self-absorption coefficient of the radiation in the standard source, and finally, using the formula 1 was determined the measurement accuracy.

The absolute detection efficiency for a single energetic line corresponding to one gamma emitting radionuclide is given by the formula:

$$\varepsilon_a^{Rad} = \frac{\sum A}{\Lambda \cdot k_{auto} \cdot s \cdot t} \qquad (1)$$

ε_a^{Rad} = absolute detection efficiency corresponding to the gamma emitting radionuclide;

$\sum A$ = sum of the corrected area of the axial profile of a standard source;

Λ (Bq) = the activity of the standard source at the time measurement;

k_{auto} = self-absorption coefficient of the radiation emitted by the standard source;

s = yield of gamma transition for the energetic line of interest;

t = acquisition time per step for the U.E.M.

If the irradiated fuel element measurements were performed by placing an attenuator in front of the detector, the re-measuring of the standard source is necessary / repeated, using the same attenuator, in the same position.

The total number of nuclei in the irradiated fuel element is calculated for each gamma emitting radionuclide, using the formula:

$$N = \frac{\sum A}{\varepsilon \cdot k_{auto} \cdot s \cdot t \cdot \lambda} \tag{2}$$

N = total number of nuclei of gamma emitting radionuclide of interest, from the irradiated fuel element;

$\sum A$ = sum of the corrected area of the axial profile for the irradiated fuel element;

k_{auto} = self-absorption coefficient of the radiation in the irradiated fuel element;

s = yield of gamma transition for the energetic line of interest;

t = acquisition time per step for the U.E.M.

λ = decay constant for the gamma radionuclide of interest;

- **Computed Tomography Reconstruction Procedure**

Computed tomography reconstruction requires positioning and controlled displacement high precision devices which are often very expensive. P.I.E.L. has a special device that gamma scanning control is performed. Difference between the equipment used for the gamma scanning and the equipment absolute necessary for computed tomography reconstruction it consists in positioning adjustable slit. While for gamma scanning, the rectangular slit is perpendicular to the axis of irradiated fuel element for the construction of the radial profiles, the rectangular slit is parallel to the axis of the irradiated fuel element. The slit is moved so as to cover from one end to the other of the diameter of the entire irradiated fuel element, to give a radial profile. Other radial profiles are made by turning the fuel element about its own axis and repeating the procedure described. To perform computed tomography reconstruction must specify parameters such as the radius of the irradiated fuel element and attenuation coefficient of gamma radiation corresponding fission product of interest which inserted into the computing code M.E.M. (Maximum Entropy Method) perform computed tomography reconstruction for radionuclide chosen [3].

The positioning device in the U.E.M. for fragments of irradiated fuel element was design and realized at our institute, from stainless steel and it was built with many components in terms of its handling with the robotic arms in the hot cells. Its realization was necessary because they were required to examination both the normal irradiated fuel elements and fragments of irradiated fuel elements or even irradiated fuel elements with different problems (ex. cracks sheath and cracks in the plug end), (see Fig. 4).

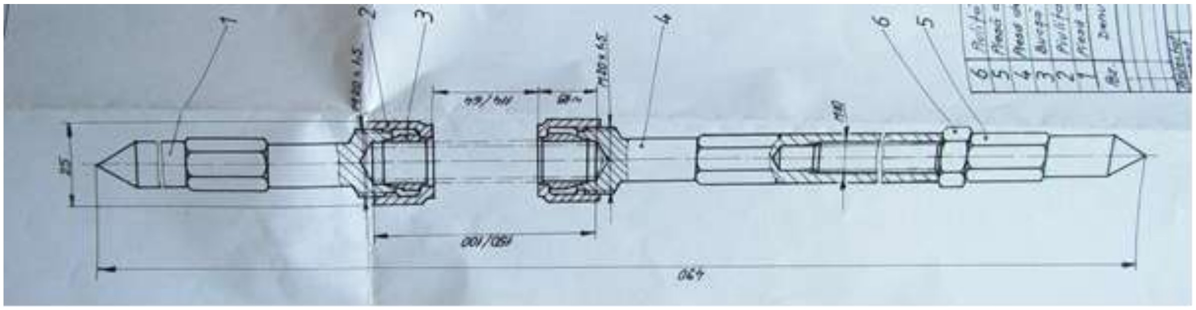

Fig. 4. Positioning device in the U.E.M. for fragments of irradiated fuel element

Results and discussion

- **Axial and radial distributions respectively computed tomography reconstructions for the standard ^{137}Cs source**

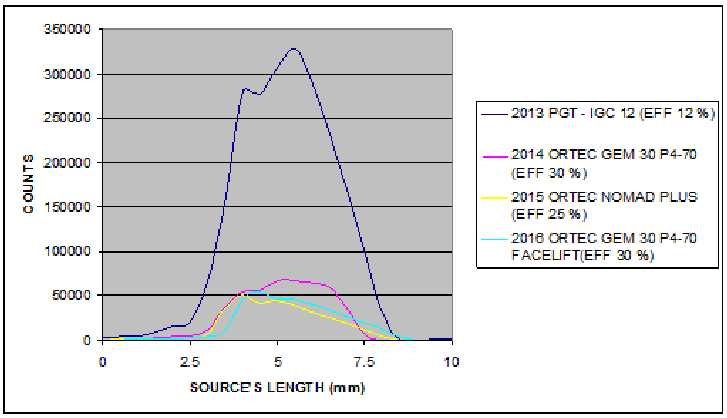

Fig. 5. *Axial profile of the standard ^{137}Cs source*

In Fig. 5 the axial profile obtained with the 2013 PGT – IGC 12 detector (EFF 12 %) differ compared to the other profiles due to the fact that the graphs is realized based on the number of counts acquired for a one measurement reported to the acquisition time/U.E.M. step.

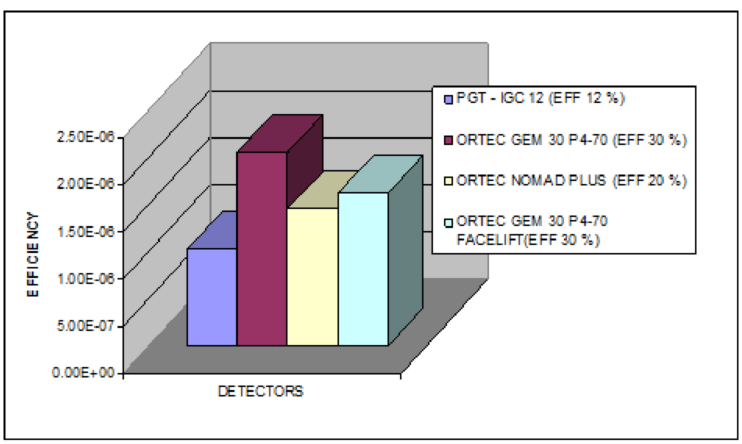

Fig. 6. *Measuring relative efficiency of the equipment used*

In Fig. 6 it can be observed that the measurement accuracy of the same equipment ORTEC GEM 30 P4-70 (EFF 30 %) / ORTEC GEM 30 P4-70 FACELIFT (EFF 30 %) are different due to the fact that for the ORTEC GEM 30 P4-70 FACELIFT (EFF 30 %) the position in relation to U.E.M. was slightly different, respectively the distance between detector and analyzed sample was different.

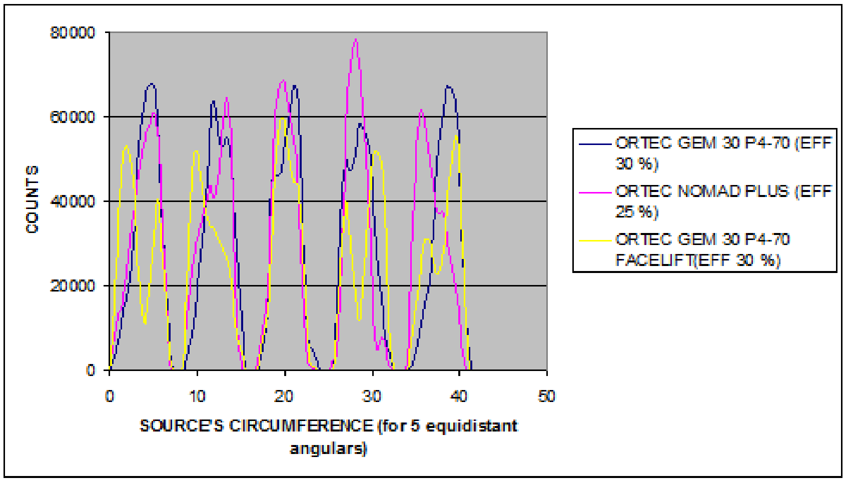

Fig. 7. *Radial profile of the standard 137Cs source*

Specters profile from Fig. 7 respectively computed tomography reconstructions from Fig. 8, both sets for the ^{137}Cs calibration source, shows that, perhaps, the standard ^{137}Cs source is in the form of encapsulated powder and whenever it is installed / removed from U.E.M. it takes a different form.

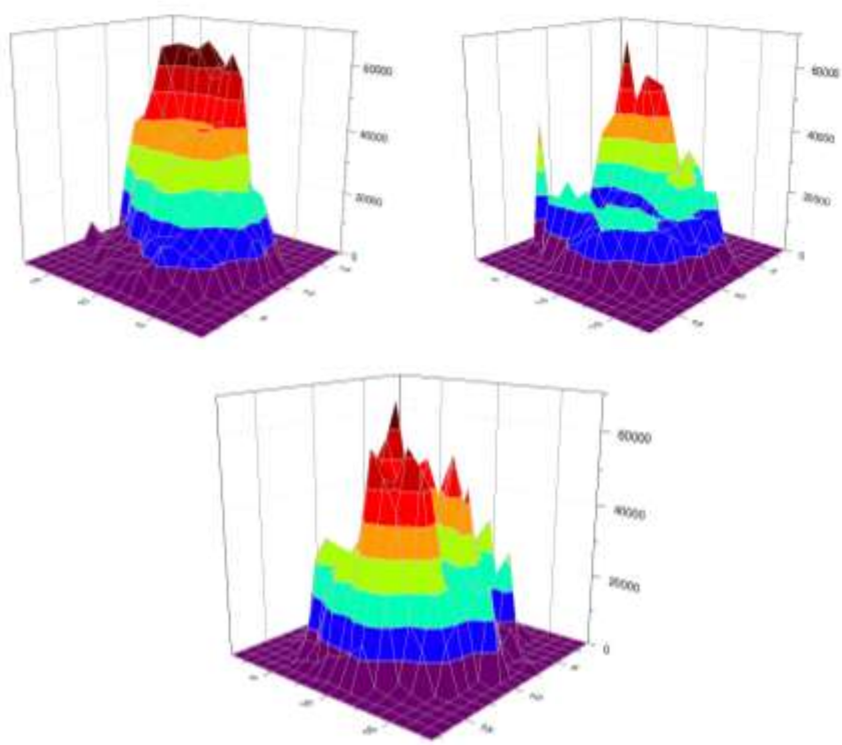

Fig. 8. *Computed tomography reconstructions for the standard ^{137}Cs source realized with ORTEC GEM 30 P4-70 (2014)/ORTEC NOMAD PLUS (1998)/ORTEC GEM 30 P4-70 FACELIFT (2016) – detectors HP Ge*

- **Axial and radial distributions, respectively computed tomography reconstructions of the fission product ^{137}Cs for irradiated fuel elements**

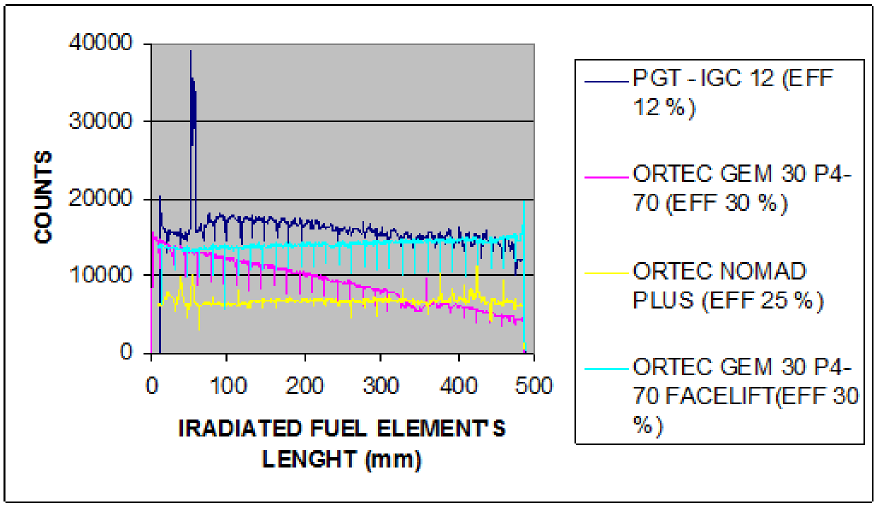

Fig. 9. *Axiale profiles of ^{137}Cs for different irradiated fuel elements*

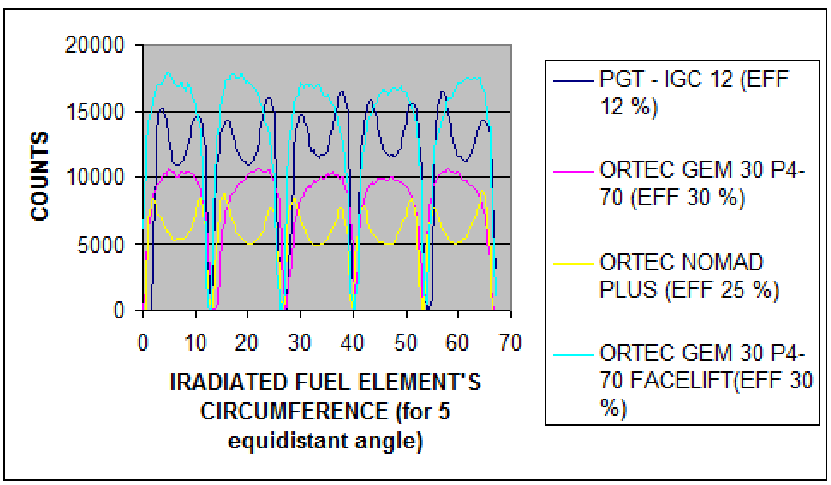

Fig. 10. *Radial profiles of ^{137}Cs for different irradiated fuel elements*

In Fig. 9 and Fig. 10 are shown the axial and radial profiles of different irradiated fuel elements having different particularities (ex. cracks sheath and cracks in the plug end, fission product migration in certain areas).

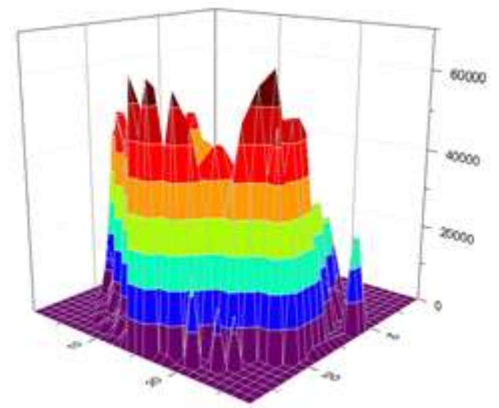

Fig. 11. *Computed tomography reconstructions of an irradiated fuel element cross section for ^{137}Cs realized with PGT – IGC 12(1986)*

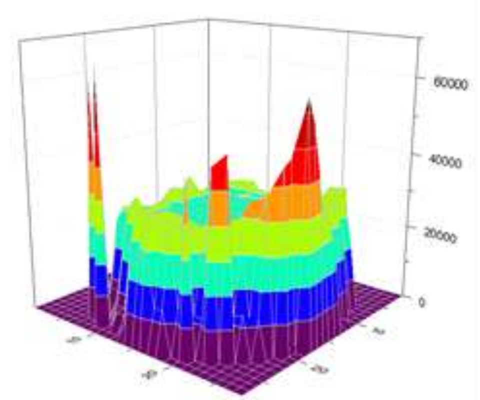

Fig. 12. *Computed tomography reconstructions of an irradiated fuel element cross section for ^{137}Cs realized with ORTEC GEM 30 P4-70 (2014)*

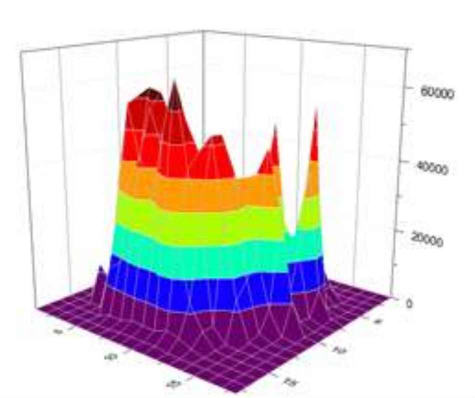

Fig. 13. *Computed tomography reconstructions of an irradiated fuel element cross section for ^{137}Cs realized with ORTEC NOMAD PLUS (1998)*

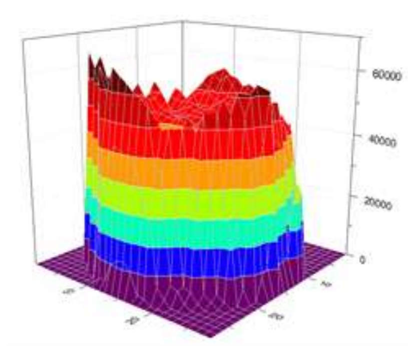

Fig 14. Computed tomography reconstructions of an irradiated fuel element cross section for ^{137}Cs realized with ORTEC GEM 30 P4-70 FACELIFT (2016)

In Fig. 11-14 are presented the computed tomography reconstructions of various irradiated fuel element cross section for ^{137}Cs fission product. Deep red areas represent how the activity of fission product is distributed in the irradiated fuel element, ^{137}Cs activity dropping from the outside to the inside of the irradiated fuel element.

Conclusions

The detection efficiency of the three detectors influences the length measurements, but with an optimal choice of the acquisition time at each step of U.E.M., measurement precision is not affected. Using a standard sources as close as possible to the geometry of which is measured in the irradiated fuel elements is very important for the accuracy of the measurements. It has to take into account the self-absorption both in the standard sources and in the irradiated fuel elements. The fission products distribution in the cross section offers information about thermal neutron flux distribution in the reactor and allows us an accurate evaluation of the self-absorption coefficients. Regardless the detection efficiency of the measuring equipment, gamma scanning technique remains one of the effective methods for the activity calculation of the gamma emitting fission products, existing in the irradiated fuel elements.

References

[1] ORTEC Solid – State Photon Detector – Operator's manual, p. 2.
[2] ORTEC Gamma Vision 7.01 HP Ge Gamma Ray Spectrum Analysis and MCA Emulation, 2014, p. 2.
[3] Dr. Teddy Craciunescu, MENER PROJECT nr. 4102, p. 4-5.