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Measurement of the Energy Spectrum of the Neutrons inside the Neutron Flux Trap Assembled in the Center of the Reactor Core IPEN/MB-01

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Abstract: This paper presents the neutron energy spectrum in the central position of a neutron flux trap assembled in the core center of the research nuclear reactor IPEN/MB-01, obtained by an unfolding method. To this end, we have used several different types of activation foils (Au, Sc, Ti, Ni, and plates) which have been irradiated in the central position of the reactor core (setting number 203) at a reactor power level (64.57 ± 2.91 watts). The activation foils were counted by solid-state detector HPGe (high pure germanium detector) (gamma spectrometry). The experimental data of nuclear reaction rates (saturated activity per target nucleus) and a neutron spectrum estimated by a reactor physics computer code are the main input data to get the most suitable neutron spectrum in the irradiation position obtained through SANDBP (spectrum analysis neutron detection code-version Budapest University) code: a neutron spectra unfolding code that uses an iterative adjustment method. The adjustment resulted in $(3.85 \pm 0.14) \times 10^9$ $n \cdot cm^{-2} \cdot s^{-1}$ for the integral neutron flux, $(2.41 \pm 0.01) \times 10^9$ $n \cdot cm^{-2} \cdot s^{-1}$ for the thermal neutron flux, $(1.09 \pm 0.02) \times 10^9$ $n \cdot cm^{-2} \cdot s^{-1}$ for intermediate neutron flux and $(3.41 \pm 0.02) \times 10^8$ $n \cdot cm^{-2} \cdot s^{-1}$ for the fast neutrons flux. These results can be used to verify and validate the nuclear reactor codes and its associated nuclear data libraries, besides, show how much effective it can be that the use of a neutron flux trap in the nuclear reactor core to increase the thermal neutron flux without increase the operation reactor power level. The thermal neutral flux increased 4.04 ± 0.21 times compared with the standard configuration of the reactor core.

Key words: Thermal neutron flux, flux trap, activation detectors, neutron spectrum, zero power reactor.

1. Introduction

Several nuclear reactor physics parameters are obtained through the spectrometry gamma of targets irradiated inside the research reactor core. This is the case of the nuclear reaction rates measured by activation foils, when parameters, such as irradiation time, the counting time, the wait time for the gamma spectrometry, the efficiency of the counting system and activation cross section are known. Thus, it is possible to determine the neutron flux in the very place where the foils are irradiated inside of the reactor core .The nuclear reaction rates induced at activation foils irradiated depend on several factors. The most important one is the cross section magnitude of the different materials (activation detectors) that cover different energy regions of the neutron spectrum. The experimental values obtained can be used to estimate important reactor physics parameters, such as neutron flux (thermal, intermediate and fast), extension of the asymptotic region, spectral indices (cadmium ratio), buckling, etc.. These experimental parameters when compared to the same parameters calculated by reactor physics codes enable to check the accuracy and precision of the different calculation methodologies and their nuclear data libraries associated.

The knowledge of the energetic and spatial neutron flux distribution in the reactor core is very important because it enables to estimate with good precision several of the reactors physics parameters, such as

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nuclear reaction rates, fuel burn up and safety parameters like temperature distribution, peak factors, and reactivity with good precision.

The neutron energy spectrum of research reactors can be obtained by nuclear reaction rates induced in activation detectors (metallic foils) irradiated inside of the reactor core always in the same position and the same reactor operational conditions, such as power level, control rods positions, water temperature (moderator), etc. [1, 2]. For this purpose may be used to unfold codes, such as SAND II, SANDBP (spectrum analysis neutron detection code-version Budapest University) [1], SPECTRA [3] and others. These computational codes work iteratively and the process start via a normalized initial spectrum usually calculated by a reactor physics code with an associated data libraries. In each iterative step, an initial normalized neutron spectrum is modified by the "best fit" between the measured foils reaction rates and the computed activities. The process stops when a set of conditions defined initially occur. This study used the SANDBP which is a computer code developed by the Technical University of Budapest, Hungary. This code is a modified version of the code SAND II [4] and the main difference is that uncertainties of saturation activities per target nucleus are considered. Thus, it is possible to obtain the neutron energy spectrum by groups and its uncertainties after running the code SANDBP. The input neutron energy spectrum of the irradiation site is estimated by reactor physics codes. The adjustment process is iterative and step-by-step, the flux values in the groups are obtained by the minimal deviation of the ratio between the nuclear reaction rates calculated in current iteration and the experimental values. By this process, the initial neutron spectrum is modified in the several energy ranges where the detectors are activated. In this work, the initial neutron spectrum was calculated using the MCNP-4C code [5] in 640 energy groups in the same position in which activation foils were irradiated (central position of the neutron flux trap) computed in 250 cycles, 1,000,000 stories per cycle and an average error of 1% in each flux group. The nuclear data library used in SANDBP is the ENDF/B-VII.0.

A fundamental activity of the research reactors is the production of radioisotopes. To maximize the thermal neutron flux responsible for most of the radioactivity induced in the irradiated material, use is made of so-called neutron flux traps that are essentially spaces within the core of the reactor filled with a moderator with a very high moderation ratio that in our case is light water, but could be other materials, such as beryllium or heavy water. This paper investigates the thermalization of the neutrons energy spectrum and the increase of the thermalneutron flux obtained from the use of neutron flux trap.

2. Experimental Methodology

The IPEN/MB-01 is a zero power nuclear reactor designed to measure a variety of nuclear reactor physics parameters with more than 3,000 operations realized where were done several experiments and some of them have international status of the NEA (Nuclear Energy Agency) (benchmarks) [6]. The standard reactor core consists of fuel assembly (26 × 28 rods rectangular configuration). This configuration has 680 fuel elements (rods), two safety bars and two control bars. Each control bars and safety bars contains 12 rods of materials with high neutron absorption cross section. The safety rods are made by Boron Carbide B4C and the control rods are made by a mixture of chemical elements Ag-In-Cd. The fuel is UO_2 enrichment of 4.3%. The fuel element is clad with 304 stainless steels. A complete description of IPEN/MB-01 can be viewed through a wide literature [6-9].

Research reactors can be used for the production of radioisotopes and radio-pharmaceuticals of broad social application in industry, agriculture and medicine. The most important radioactive elements produced are obtained from materials of very high capture cross section in the thermal neutron spectrum region. So, if the magnitude of the thermal neutron flux increases, this will result in the production of radioactive elements with higher activities. There are two ways to increase the thermal neutron flux in a research reactor: increasing the operating power or assembling neutron flux trap inside the core [9]. In this work, several core configurations containing inside a neutron flux trap were assembled, and the most effective was the configuration number 203, which is sketched in Fig. 1. To assemble this configuration was to withdraw a total of 32 fuel rods from the center reactor core. To minimize the neutron leakage from the system, it was assembled with a hexagonal cross section core (approximated cylindrical cross section) (see Fig. 1).

2.1 Irradiation Conditions

To obtain the neutron energy spectrum in the center of the IPEN/MB-01 reactor core, five different foils have been irradiated with their respective eight nuclear reactions. The characteristics of the activation foils irradiated are shown in Table 1.

The times of irradiations were estimated from knowledge of the half-lives of the different radionuclides and the magnitude of its cross section. Some foils were covered by cadmium boxes to prevent interference of thermal neutron flux. Each irradiation was made at the power of 64.7 watts in the same experimental conditions (control rods positions, positions foils). The foils were placed in the central position of the core by means of an articulated device consisting of an acrylic support. The foils are fixed in acrylic support, and then, the apparatus was inserted into the space between channels 14 to 15 shown in Fig. 1. The nuclear channel

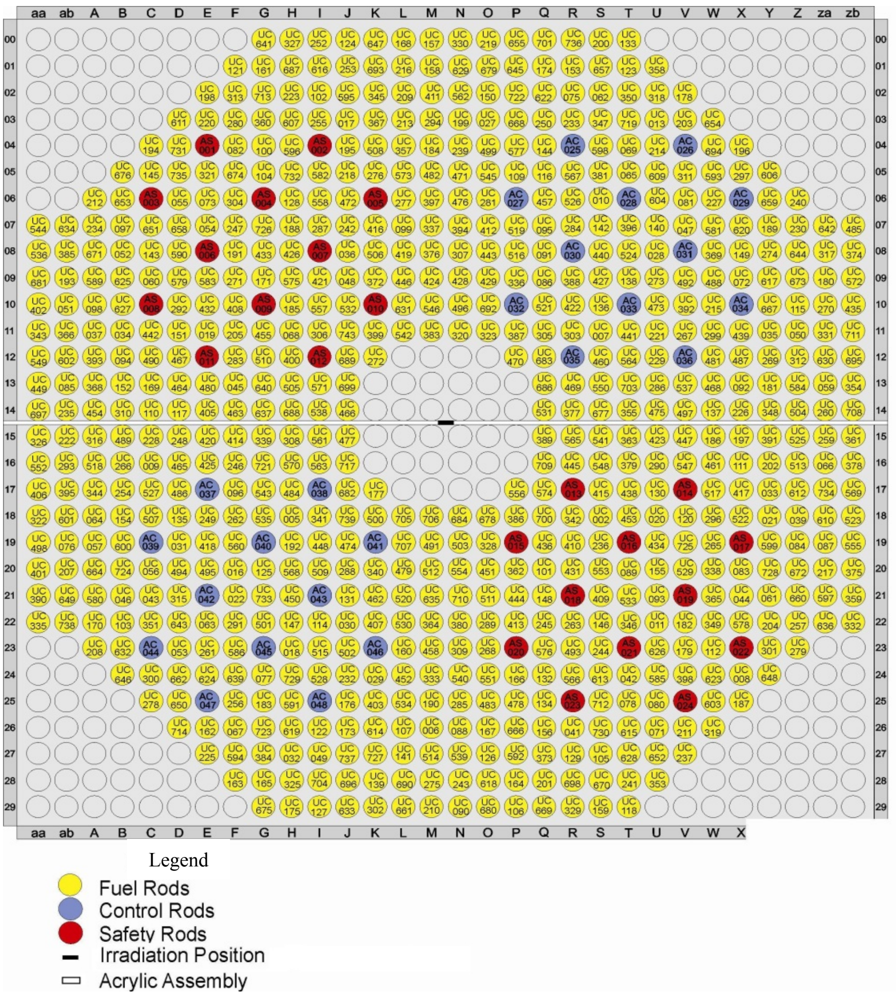

Fig. 1 Reactor core (configuration number 203) with neutron flux trap at the centre and position of activation foil at irradiation.

Table 1 Foils data: nuclear reactions, irradiation time, cross section, half-life of formatted radionuclide, mass and thickness.

Irradiated activation foil	Nuclear reaction	Irradiation time (h)	Cross section (barns)	Half-life (h)	Foil mass (g)	Nominal thickness (cm)
1% Au-Al	^{197}Au(n, γ)^{198}Au	1	99.57	64.56	0.02394	0.02
1% Au-Al*	^{197}Au(n, γ)^{198}Au	1	15,630**	64.56	0.02478	0.02
Sc	^{45}Sc(n, γ)^{46}Sc	2	27.21	2,011.92	0.01843	0.00127
Sc*	^{45}Sc(n, γ)^{46}Sc	2	11.18**	2,011.92	0.01848	0.00127
Ti	^{47}Ti(n, p)^{47}Sc	2	1.76×10^{-2}	80.16	0.14463	0.00254
Ni	^{58}Ni(n, p)^{58}Co	2	62.4×10^{-3}	1728	0.28500	0.00254
In	^{115}In(n, n')^{115m}In	1	183×10^{-3}	4.50	0.04046	0.00127
In*	^{115}In(n, n')^{115m}In	2	183×10^{-3}	4.50	0.03926	0.00127

*Cadmium covered (0.5 mm thickness); **resonance integral.

number 10 (B-10 detector) is the detector further away from the core and is on east side of the reactor core (approximately 40 cm). Nuclear channel 10 has been used to normalize the little difference of the operation power level between each irradiation. The control rods were always the same positions (90.2% withdrawn) during all irradiations to avoid neutron flux disturbance.

2.2 Gamma Spectrometry

The activation foils were sent to the laboratory for gamma spectrometry to determine radioactivity induced after irradiation. This parameter is proportional to the saturation activity of the target nucleus. An HPGe (high pure germanium detector) with efficiency of 45% was used to make the gamma spectrometry. Fig. 2 shows the radioactive decay curve of an infinity dilute foil compound with 1% of Au-198 and 99% of Al.

The net counting is obtained by gamma ray spectrometry of activation foil irradiated in the center of the core. The counting of photo peaks gamma emitted by foils made possible to obtain the nuclear reaction rate which is numerically equal to the saturation activity. This requires prior knowledge of the counting efficiency for the energy of the gamma width of the energy of the gamma photopeak spectrometry is performed, as well as the irradiation

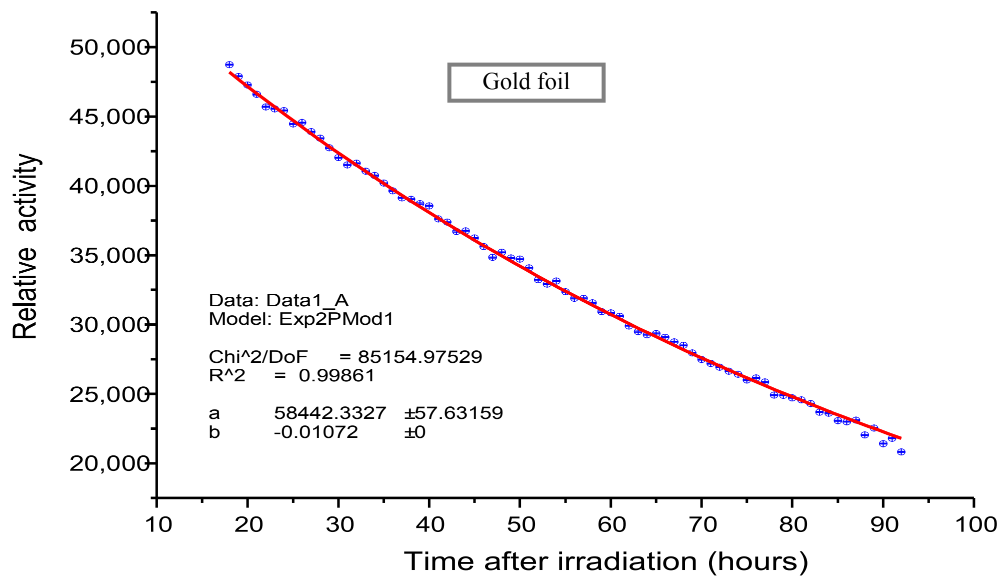

Fig. 2 Decay curve to 1% of Au (Au-198) and 99% of Al foil.

times: ti = irradiation time, te = wait time and tc = counting time. Eq. (1) calculates the nuclear reaction rates (saturation activity A^{∞}).

$$A^{\infty} = \frac{\lambda \cdot e^{\lambda \cdot t_e} \cdot (C - BG)}{\epsilon \cdot I \cdot (1 - e^{-\lambda \cdot t_i})(1 - e^{-\lambda \cdot t_c})} \cdot CF \qquad (1)$$

Thus, λ is the decay constant of the radionuclide formed an activation foil and CF is a total correction factor and corresponds to three different factors: normalization factor that corrects little fluctuation power level between the several irradiations, ramp factor that diminishes the value of detector activity obtained during ramp power elevation until the steady state radiation level and self-absorption factor. The saturated activity per target nucleus A_s is given by Eq. (2), where W is the atomic weight of the target nucleus, m is the mass foil, f_{iso} is the isotropic fraction of the target nucleus, and N_A is the Avogadro number.

$$A_s = \frac{A^{\infty} \cdot W}{m \cdot N_A \cdot f_{iso}} \qquad (2)$$

3. Analysis of the Results

The experimental saturation activities and the calculated activity obtained from SANDBP code after three iterations for each foil are given in Table 2. The total standard deviation is 5.44% between the measured activities and those adjusted by the most appropriated neutron spectrum obtained through SANDBP code. Table 3 presents the neutron fluxes values given by SANDBP code at different ranges of energy. Fig. 3 shows a comparison between the adjusted spectrum computed by SANDBP code and the initial spectrum calculated by MCNP-4C code.

The most appropriated energy neutron spectrum adjusted by SANDPB in 50 groups is showed in Fig. 4.

The neutron flux collapsed in three energy groups, thermal (<0.56 eV), intermediate (0.56 eV until 0.5 MeV) and fast (>0.5 MeV) at the central position of the configuration core number 203 (Fig. 1) is given below.

Table 2 Measured and calculated saturated activity per target nucleus after three iterations by SANDBP code. The MCNP-4C code has been used how input spectrum.

Foil	Daughter nuclide	Gamma photo-peak (keV)**	Saturated measured activity*** (DPS/nucleus)	Calculated activity by SANDBP code (DPS/nucleus)
Au-Al	^{198}Au	411.8	$3.011 \times 10^{-13} \pm 2.38\%$	2.919×10^{-13}
Au-Al*	^{198}Au	411.8	$8.626 \times 10^{-14} \pm 2.38\%$	8.850×10^{-14}
Sc$^{\Delta}$	^{46}Sc	889.20	$4.978 \times 10^{-14} \pm 5.22\%$	5.611×10^{-14}
Sc$^{\Delta,*}$	^{46}Sc	889.2	$1.076 \times 10^{-15} \pm 2.34\%$	1.01210^{-15}
Ti	^{47}Sc	159.4	$8.294 \times 10^{-17} \pm 4.62\%$	8.12610^{-18}
Ni	^{58}Co	810.8	$4.917 \times 10^{-16} \pm 5.22\%$	4.787×10^{-16}
In	^{115m}In	336.2	$8.021 \times 10^{-17} \pm 5.54\%$	8.238×10^{-17}
In*	^{115m}In	336.2	$8.321 \times 10^{-17} \pm 5.53\%$	8.143×10^{-17}
Total standard deviation				5.44% (2 sigma)

*Cadmium covered; **net peak counts during the gamma spectrometry; $^{\Delta}$corrected to self-shielding effect.

Table 3 Neutron flux values obtained by SANDBP code [6].

Energy (MeV)	Integral neutron flux ($n \cdot cm^{-2} \cdot s^{-1}$)	Standard deviation
$> 1\times10^{-10}$	3.8472×10^{9}	0.35%
$< 0.20\times10^{-6}$	2.2753×10^{9}	0.51%
$< 0.56\times10^{-6}$	2.4139×10^{9}	0.47%
> 0.1	6.1479×10^{8}	0.51%
> 0.4	4.954×10^{8}	0.42%
> 0.5	4.0905×10^{8}	0.41%
> 1.0	3.410×10^{8}	0.39%

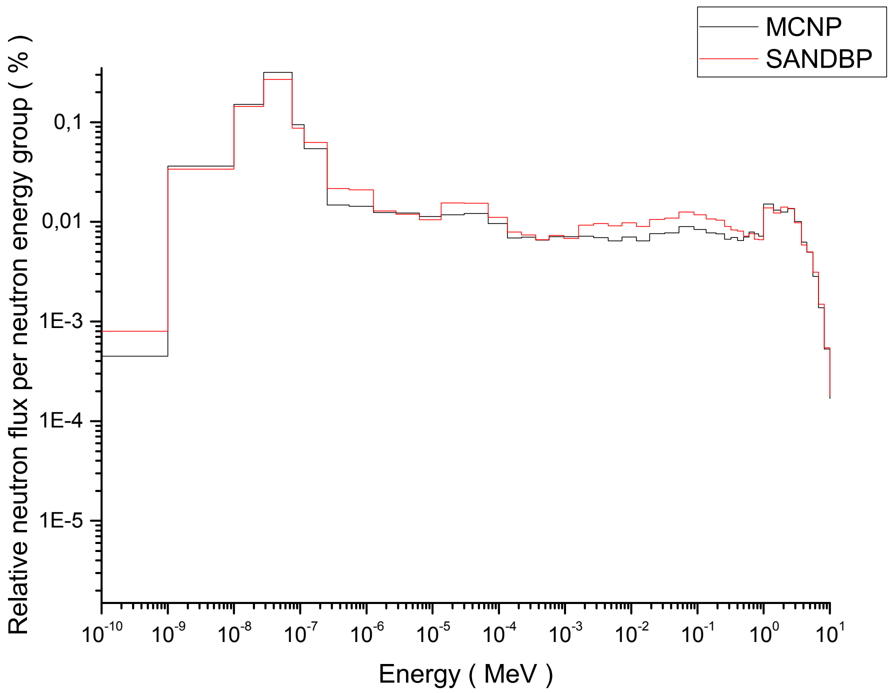

Fig. 3 Neutron spectrum at the central position of the IPEN/MB-01 obtained by SANDBP code (three iterations) to 50 energy groups [6] compared with calculate spectrum by MCNP-4C collapsed in 50 groups.

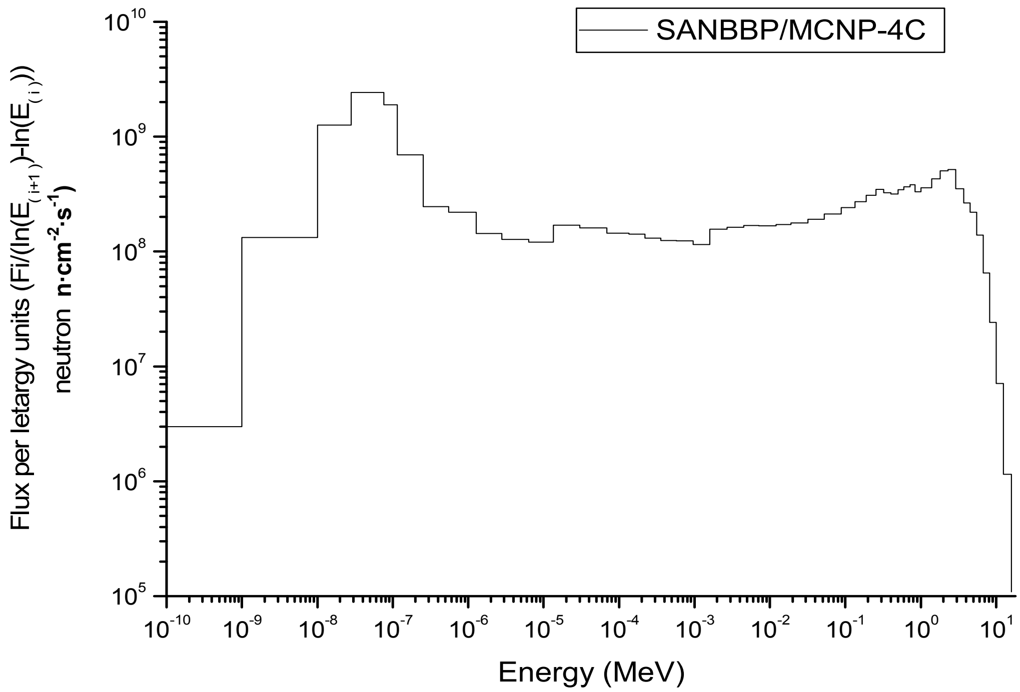

Fig. 4 Energy spectrum adjusted by SANDPP in 50 groups of energy obtained with initial spectrum calculated by MCNP-4C in 640 groups of energy.

Φ(thermal) = (2.42 ± 0.02) × 10^{9} $n\cdot cm^{-2}\cdot s^{-1}$

Φ(intermediate) = (1.09 ± 0.06) × 10^{9} $n\cdot cm^{-2}\cdot s^{-1}$

Φ(fast) = (3.41 ± 0.02) × 10^{8} $n\cdot cm^{-2}\cdot s^{-1}$

Table 4 shows a comparison between the thermal neutron flux in the central position of the flux trap and the rectangular standard core configuration in the same

Table 4 Thermal neutron flux obtained by SANDBP code at central position of the reactor core to two different core configurations of the IPEN/MB-01 reactor operating at the same power level.

Thermal neutron flux adjusted at experimental values by SANDBP code [1] ($n \cdot cm^{-2} \cdot s^{-1}$) at central position of the neutron flux trap—core configuration 203	Thermal neutron flux adjusted at experimental values by SANDBP code [1] ($n \cdot cm^{-2} \cdot s^{-1}$) at central position: of the rectangular core configuration 28 × 26 fuel rods	Increase of the thermal neutron flux in the same position and power level (64, 57 watts) by utilization of the neutrons flux trap
$(2.41 \pm 0.01)10^9$	$(5.97 \pm 0.30)10^8$	4.04 ± 0.21

position and same power level obtained by noise analysis technique [10].

The thermal neutron flux is increased by 4.04 ± 0.21 times. This result can be viewed with more details at Mura [9].

4. Conclusions

The total standard deviation between the measured activities and those adjusted by the most appropriated neutron spectrum obtained through SANDBP code is 5.44%. Only three iterations were used in the unfolding code SANDBP to obtain the neutron energy spectrum. The assembling of a neutron flux trap in the IPEN/MB-01 reactor core center (configuration number 203) generates an increase of about 304% in the thermal neutron flux when compared with that obtained using the standard rectangular configuration at the same power level. This work shows the importance to assemble a neutron flux trap in a region of the reactor core to increase the thermal neutron flux without increasing the standard reactor power level and consequent increasing the production of radioisotopes to medicinal and industrial purposes. The next work step is to investigate the increase of the thermal neutron flux caused by assembling a box with heavy water inside the neutron flux trap using the same core configuration 203.

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