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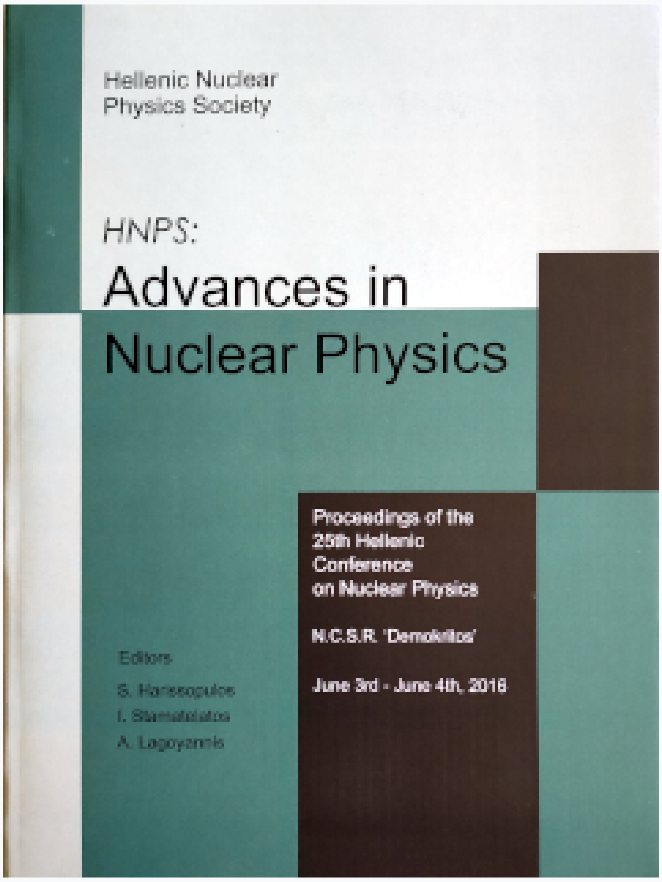

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Experimental study of the flux trap effect in a sub-critical assembly

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Abstract

The neutron flux trap effect was experimentally studied in the sub-critical assembly of the Atomic and Nuclear Physics Laboratory of the Aristotle University of Thessaloniki, using delayed gamma neutron activation analysis (DGNAA). Measurements were taken within the fuel grid, in vertical levels symmetrical to the Am-Be neutron source, before and after the removal of fuel elements, also permitting a basic study of the vertical flux profile.

Three identical flux traps of diamond shape and an area of 96 cm^2 were created by removing four fuel rods for each one. Two (n,γ) reactions and one (n,p) threshold reaction were selected for thermal, epithermal and fast flux study. For the thermal and epithermal flux, results obtained through the $^{197}Au(n,\gamma)^{198}Au$, and $^{186}W(n,\gamma)^{187}W$ reactions were used, with and without Cd covers, to differentiate between the two flux regions. For the fast flux, the $^{58}Ni(n,p)^{58}Co$ reaction was used.

All measurements were taken in a HPGe detector of 42% relative efficiency, with a resolution of 1.8 keV at 1332 keV and analyzed in the SPECTRW software package, developed at NCSR Demokritos.

An interpolation technique based on local procedures is used to fit the cross sections and the flux spectra.

End results show a thermal flux increase of 105% at the source level, and 90% across all levels, pointing to a high potential to increase the available thermal flux for future experiments. Furthermore, the vertical flux profile was found to be slightly asymmetric, with higher flux values at the top part of the assembly.

Keywords flux, trap, neutron, activation

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INTRODUCTION

For activation analysis and material irradiation purposes, the standard in thermal reactors is to utilize one or more neutron flux traps. These are empty channel positions inside the reactor's fuel grid, that feature an increased thermal flux, where samples can be placed to be irradiated. Since only coolant/moderator exists in a flux trap, it stands that any neutrons that end up in there, simply undergo moderation, without being absorbed by any fuel elements. Due to the moderator, thermal neutrons are also reflected towards the center of the channel and accumulate there, hence the name "flux trap".

The ideal position for a flux trap is the center of the fuel grid, where simple neutronics dictate that the flux is at its maximum. The Bessel function, approximates the flux profile of a symmetric fuel gird quite successfully.

The challenge presented in this experiment is the fact that, while critical reactors can easily accommodate the central flux trap design, the subcritical assembly (, open pool, natural metallic U fuel) of the Atomic and Nuclear Physics Laboratory relies on an ^{241}AmBe (α,n) source at its center. Thus, any trap design must take that into consideration. The use of a flux trap with the ultimate goal of providing an increased thermal flux for use in future experiments, has not been yet assessed.

A secondary aspect of the experiment explores the vertical flux profile of the subcritical assembly, both as a way of mapping the flux and as a non-invasive method to determine if any changes in the core geometry have taken place over the years of operation. Normally, the vertical flux profile can be described by a simple cosine fuction, its maximum at the center of the core. However, the presence of the AmBe source shapes the flux profile, so neither the approximation or the cosine approximation can be used in the case of the subcritical assembly and so there is merit in determing the vertical profile experimentally.

EXPERIMENTAL DETAILS

Two sets of irradiations were performed, one with the assembly in regular configuration and one within the flux traps. The irradiation positions were at a constant radial distance of 25 ± 0.5 cm from the centerline and they covered 7 vertical positions, axially symmetrical to the AmBe source. The geometries are given in Fig.1a and Fig.1b. The ^{197}Au(n,γ)^{198}Au and ^{186}W(n,γ)^{187}W reactions, with and without Cd covers, were used to study the thermal and epithermal flux and the ^{58}Ni(n,p)^{58}Co reaction for fast flux study.

Delayed Gamma Neutron Activation Analysis (DGNAA) was used to determine the various neutron fluxes. The basic principle expression is given in (1).

(1)

The expression might be simple, but its terms are not. The only term that can be calculated relatively easily is , if one knows the exact composition of the target material. The other terms require a significant amount of work that involves both experimental and modeling approaches.

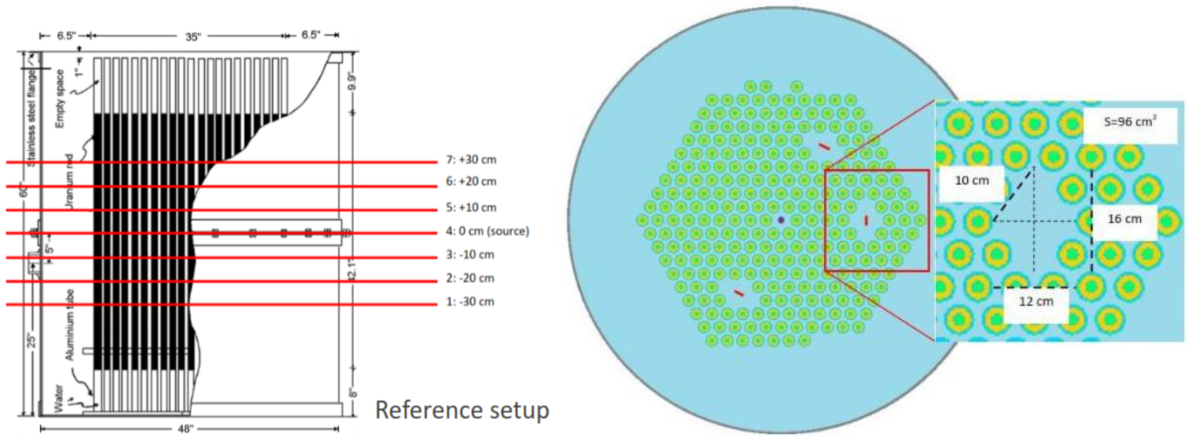

Fig.1. Irradiation positions setup

The saturated activity was derived by studying the radioactive decay of the desired isotope in the activated sample. Gamma ray measurements were taken in a HPGe detector of 42% relative efficiency and a resolution of 1.8 keV at 1332 keV. Depending on the half-life of the sample, measurements were taken for 1-4 half-lives. The spectra acquisition was done with the winTMCA32® software and spectral analysis wih the SPECTRW[1] software.

In the case of sufficient irradiation time and multiple measurements, the was calculated through fitting of the analysis results. Otherwise, the following expression was used:

(2)

where is the photopeak area, are the detector real, live and dead time, respectively, is the decay cosntant is the gamma ray intensity and finally is the detector efficiency for the particular sample geometry, composition, gamma ray energy, photon self-absorption and coincidence summing.

The next term to be calculated is the effective cross section for the reaction of interest. Since the neutrons that initiate the reaction cover a spectrum of energies and that spectrum is system dependent, the expression[2] used to calculate the effective cross section is:

(3)

The is the excitation function, the data taken from the ENDF-VII.1 and the quantity, called differential neutron flux, was taken from another experiment[3]. An interpolation technique based on local procedures was implemented to model the data. Polynomial curves were fitted between every pair of data points (X data points, X-1 curves and combined in a machine-stored piecewise function. The method provides a fast and easy solution, and was successfully validated using the ENDF-VII.1 library.

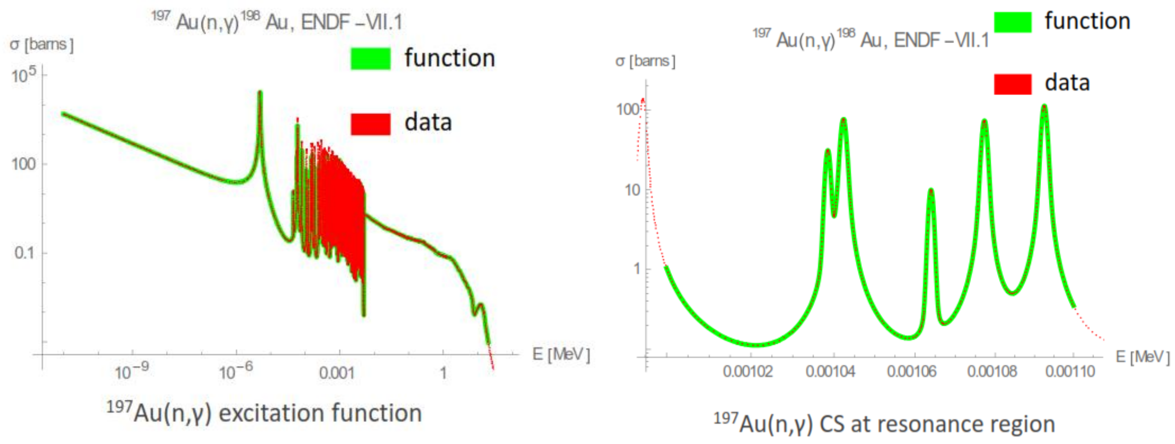

Fig.2: Modeling of the ^{197}Au(n,γ) cross section, over the entire spectrum and between 1 – 1.1 keV

RESULTS AND DISCUSSION

Flux results, after all correction factors are applied plus a paramenter for neutron self-absorption[4], are graphically presented below.

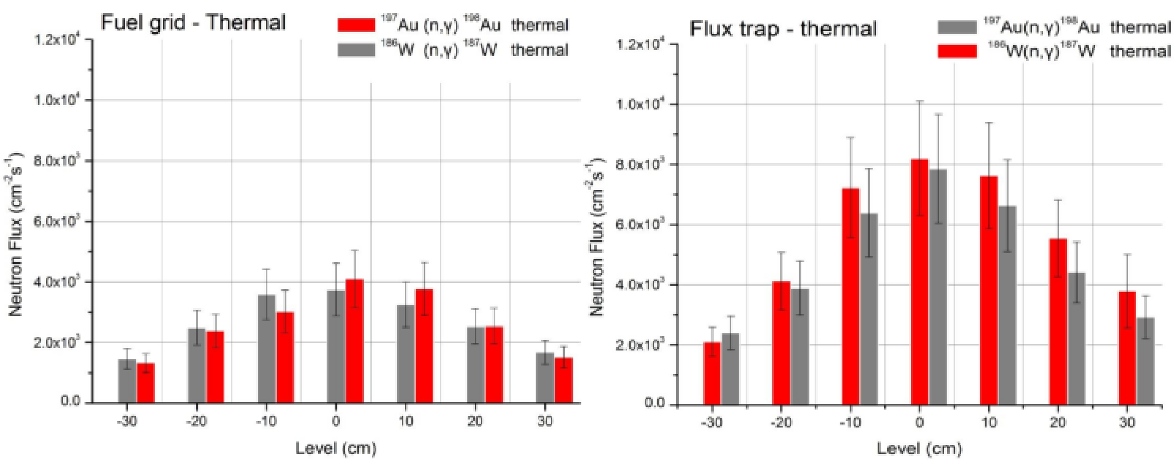

Fig.3. Thermal flux results in the fuel grid (left) and the traps (right)

Thermal flux results show an average increase of 90% across all positions. On average, the effect was similar across axial positions, with the exception of the two lower ones and that last bit led to some interesting conclusions. The central axial position yields the maximum flux, from a weighted average of the flux at initially, to , i.e. a 105% increase.

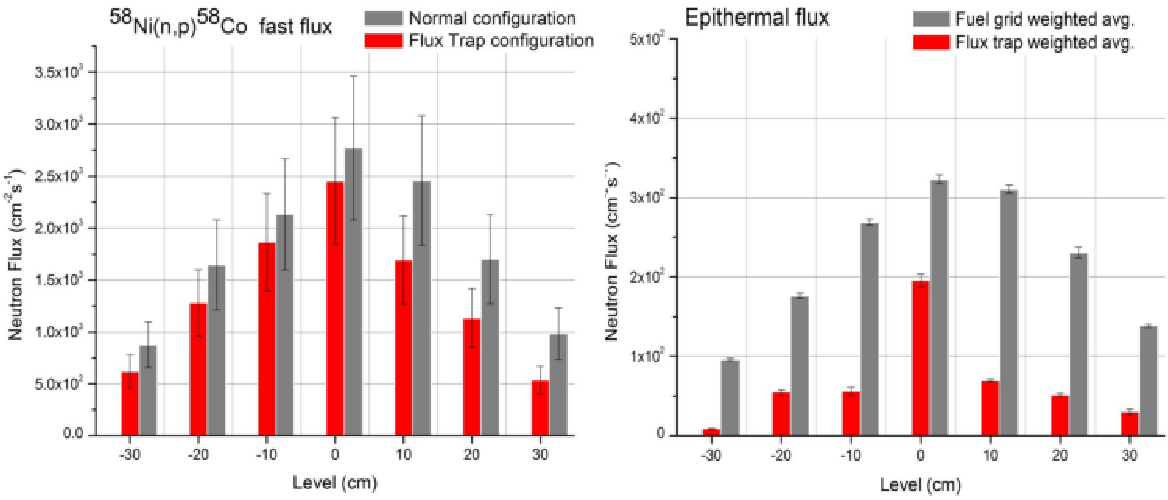

Fig.4. Fast (left) and epithermal weighted averages (right) flux results

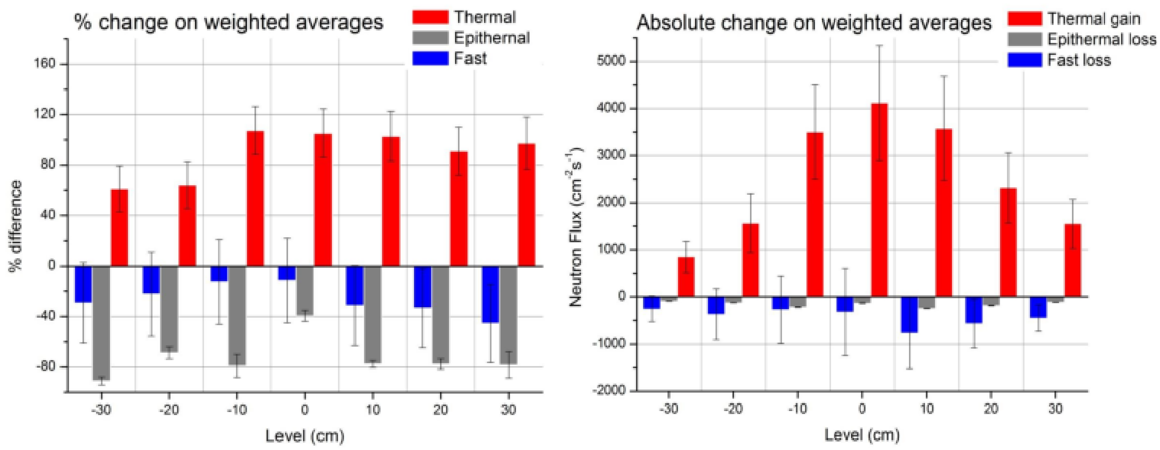

Fig.5. Percentage (left) and absolute (right) changes in neutron flux

What Fig.4 and Fig.5 show, is that the increase in thermal flux is not accompanied by a comparable decrease in epithermal or fast flux and that neutron reflection is the mainly responsible, contributing 73% on average and 89% at the central axial position.

Across all irradiation cycles, the two lower positions excibit lower flux compared to their axisymmetrical ones, ranging from 15% to 40%. This points to a downwards displacement of the fuel region. Considering that this is a subcritical system ( ), the fuel elements absorb more neutrons than they emit. The AmBe source makes up the primary flux profile and the lower irradiation positions are closer to the fuel region axial center than the upper ones and furthermore lack axial moderator reflection. Fig.6 schematically represents this, showing that the upper irradiation positions enjoy axial reflection, contrary to the lower ones.

Post-experiment dismantling of the reactor for retrofitting concluded that this was indeed the case, the fuel slugs having displaced down due to gravity, after years of operation.

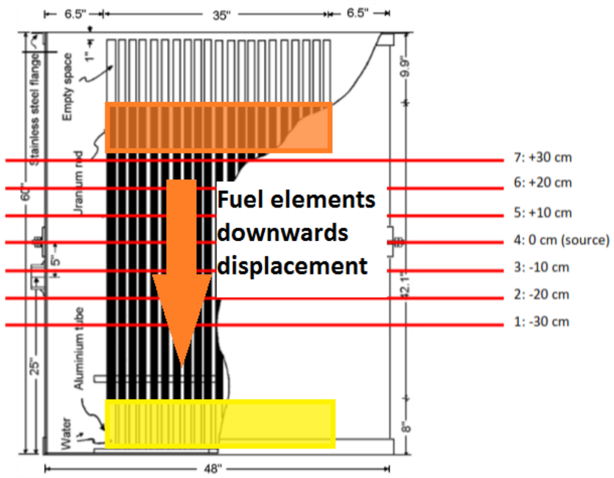

Fig.6. Displacement of the fuel region

CONCLUSIONS

The flux traps designed were found to increase the usable thermal neutron flux by a considerable amount (   ) in the source axial level, making the practice viable for future irradiation projects and showcasing the high contribution of the reflection effect in flux trap designs. At the same time, the experiment successfully determined that the fuel region had displaced towards the lower part of the tank, without requiring any physical examination of the assembly whatsoever.

It would be interesting to perform further experiments of this nature, after retrofitting of the assembly is complete, to determine the effect of the fuel displacement in the thermal flux increase. But more importantly, to properly utilize the flux trap effect and actually achieve a higher thermal flux than what is available anywhere in the standard assembly, a trap located right next to the AmBe source and surrounded by fuel elements would be the best bet and it definitely merits an investigation.

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