GRAPHITE MODERATED ^{252}Cf SOURCE

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Abstract

The Thorium molten salt reactor is an attractive and affordable nuclear power option for developing countries with insufficient infrastructure and limited technological capability. In the aim of personnel training and experience gathering at the Universidad Simon Bolivar there is in progress a project of developing a subcritical thorium liquid fuel reactor. The neutron source to run this subcritical reactor is a ^{252}Cf source and the reactor will use high-purity graphite as moderator. Using the MCNP5 code the neutron spectra of the ^{252}Cf in the center of the graphite moderator has been estimated along the channel where the liquid thorium salt will be inserted; also the ambient dose equivalent due to the source has been determined around the moderator.

Keywords: ^{252}Cf, neutrons, ambient dose equivalent, Monte Carlo.

1.- INTRODUCTION

A molten salt reactor (MSR) uses molten-salt fluid-fuel, which contains Th as fertile and ^{233}U, ^{235}U or Pu as fissile. Other advantages of these reactors are the power-size flexibility, no need of refueling, effective minor actinides incineration, and good safety and economy [Honma, Shimazu and Narabayashi 2008].

The MSR concept was developed in the early 1950s at Oak Ridge National Laboratory in the aim to have reactors whose fuel would be liquid that also acts as coolant [Mathieu, 2009; Suzuki and Shimazu, 2008].

The Thorium Molten Salt Reactor (ThMSR) is an attractive and affordable nuclear power option for developing countries. Its potential simplicity, enhanced safety and reliability, reduced waste generation, effective use of the energy content of the fuel and superior proliferation resistance makes it the ideal choice for countries with insufficient infrastructure and limited technological capability entering nuclear power usage. This type of reactor is particularly well suited for the thorium fuel cycle, and for the incorporation of reprocessing which is the most desirable fuel cycle feature. Reprocessing minimizes nuclear fuel waste and maximizes resource utilization. The Generation IV International Forum for the development of new future nuclear energy has established a set of desirable goals as research directions. These include the issues of inherent safety and reliability, efficient utilization of the fuel natural resources, nuclear weapons proliferation resistance and improved economic competitiveness. The ThMSR is certainly a strong candidate among the six chosen Gen-IV systems, being a strong competitor or excelling among the alternatives.

At the Nuclear Physics Laboratory of the Universidad Simón Bolívar (USB) in Caracas, Venezuela a liquid-fuel, zero-power, subcritical reactor will be designed and eventually it will be built. Due to its design features a core containment vessel is not required, neither a massive shielding. A subcritical reactor requires an extraneous neutron source and a neutron moderator; for the USB reactor the source is ^{252}Cf and the moderator is graphite [Greaves *et al*., 2005].

Like many trans-uranium elements, ^{252}Cf undergoing α-decay, also disintegrate by spontaneous fission releasing several neutrons. The ^{252}Cf has a half-life of 2.73 years, 3.2% of it decays by spontaneous fission releasing 3.7 neutrons/fission [Garga and Batra, 1986]. The specific source strength is 2.311x10^{12} s^{-1}-g^{-1}, with a neutron spectrum that can be represented by function shown in equation 1.

$$\Phi_E(E) = C\,\sqrt{E}\;\; \mathrm{Exp}[\,-E/T\,] \qquad (1)$$

Here, C is a constant, E is the neutron energy and T is the temperature. Using T = 1.42 MeV, the average neutron energy is 2.3 MeV [Vega-Carrillo *et al*., 2007]. The dosimetric features of a point-like ^{252}Cf source are 391, 407, 347 and 169 pSv-cm^2 for the fluence-to the ambient dose equivalent, personal, effective anteroposterior and effective isotropic, conversion factors respectively [Vega-Carrillo *et al*., 2005]. The source performance in a subcritical array must be characterized in the graphite bulk where the source neutrons are moderated and used to activate small samples.

The objective of this work was to estimate the neutron distribution along the irradiation channel, and the ambient dose equivalent in sensitive locations in the graphite used to moderate and to shield the neutrons emitted by the ^{252}Cf source.

2.- MATERIALS AND METHODS

The source is 0.35 Ø x 1 cm^2; originally it had 20 μg of ^{252}Cf producing 4.6 x10^7 s^{-1}. In order to reduce the gamma-rays the source is stored in a 15x15x15 cm^3 Pb-shield; in the figure 1 is shown the lead cube shielding with graphite around it.

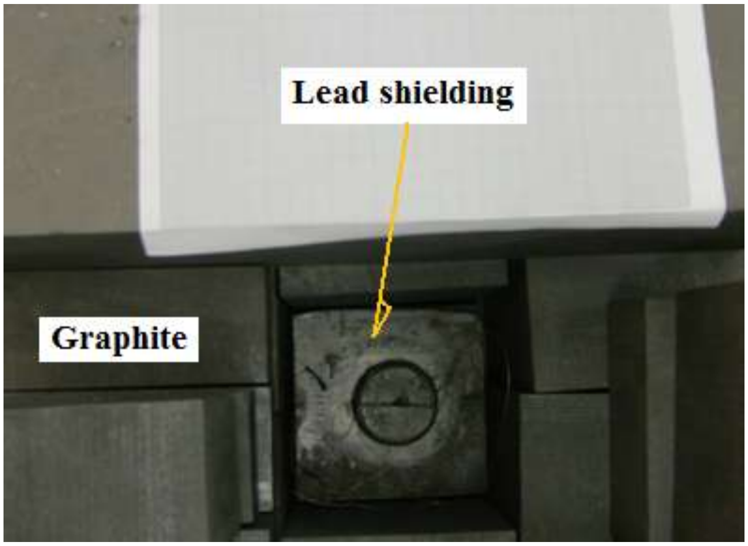

Figure 1.- Lead cube and graphite.

There are 16 graphite pieces of 20 x 20 x 60 cm^3 forming an approximate parallelepiped body of 80 x 80 x 60 cm^3 that is surrounded by borated polyethylene plates. This setup has an irradiation channel that can be seen in figure 2.

Figure 2.- Complete ^{252}Cf setup.

To 20 cm from the source's center there is a pipe used as irradiation channel [Greaves *et al*., 2005].

Using the Monte Carlo code MCNP5 [Forster *et al*., 2004] a detailed model of this array was designed. In figure 3 are shown two views of the model, where, the source, the lead cube, the graphite and the polyethylene are shown.

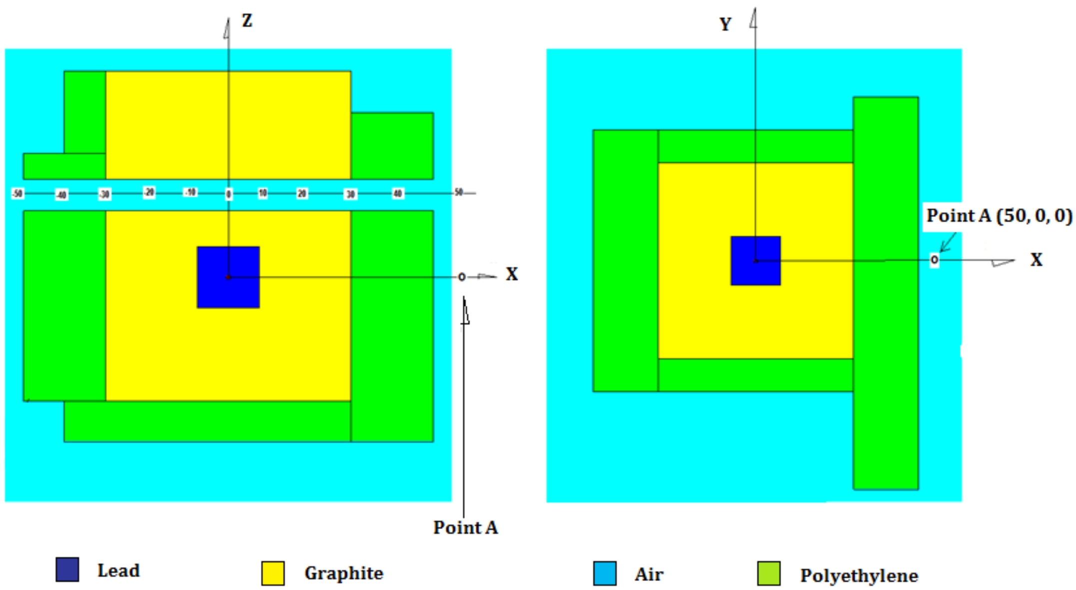

Figure 3.- Monte Carlo model of the ^{252}Cf complete setup.

Along the irradiation channel point-like detectors were simulated to estimate the neutron spectrum from the ^{252}Cf and the total fluence.

Two mesh tallies were used in the X-Y plane, one was in the middle point of the source axis and another was placed in along the irradiation channel. Neutron flux and Ambient dose equivalent per each neutron emitted by the source were calculated.

3.- RESULTS AND DISCUSSION

In figure 4 is shown the total neutron fluence, per neutron emitted by the source, along the axis of the irradiation channel.

The value in 0 cm is 20 cm above the center of the source. From the figure it can be noticed that there is symmetry, as expected. The total neutron fluence decreases as the distance, along the channel, increases with respect to the center.

The fluences in both extremes are located at the entrances of the irradiation channel.

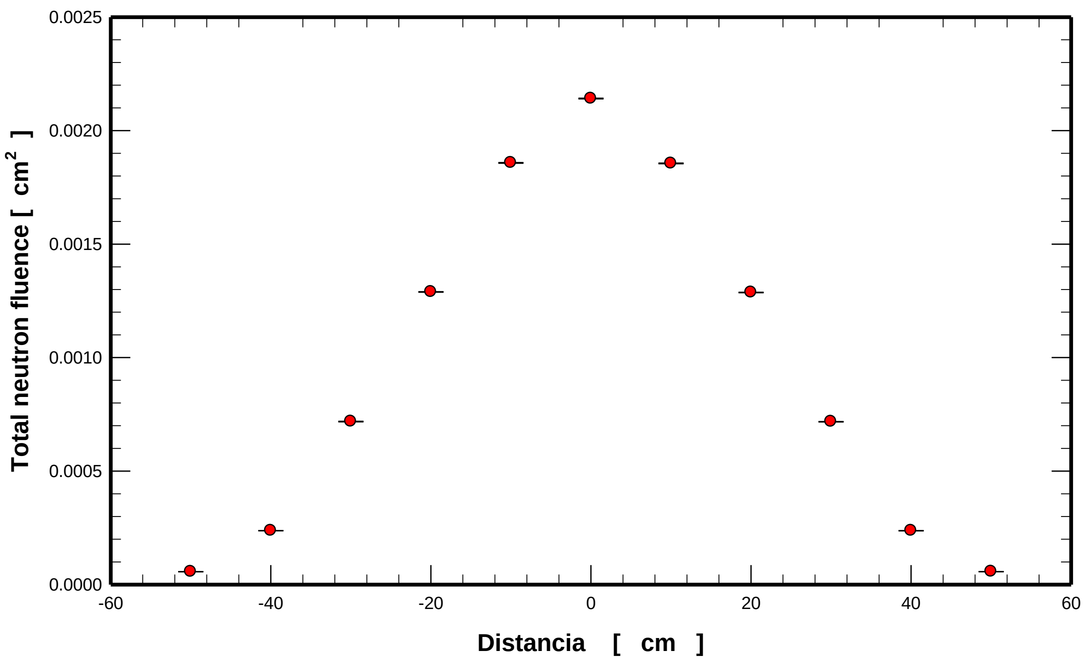

Figure 4.- Total neutron fluence along the irradiation channel.

In the figure 5 the neutron spectra at each point in the irradiation channel are shown.

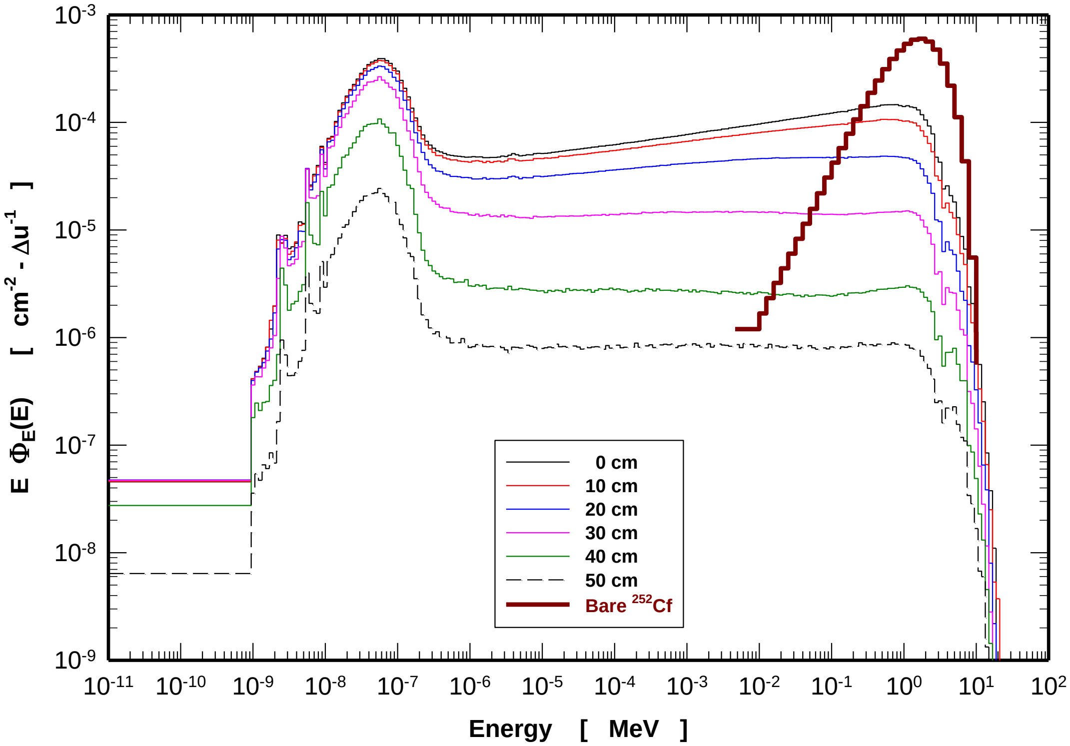

Figure 5.- Neutron spectra from 0 to 50 cm along the irradiation channel.

In figure 5 was also included the relative lethargy spectrum of a bare ^{252}Cf [IAEA 2001], having neutrons from 10^{-2} up to10 MeV. However the spectra along the irradiation channel has also epithermal and a large component of thermal neutrons due to the moderation in the graphite. As the distance about the center of the irradiation channel increases the spectra tend to decrease, and the peak of the bare ^{252}Cf tends to vanish.

In figure 6 is shown the contour distribution of the total neutron fluence in a 70 x 70 cm^2 X-Y plane perpendicular to the vertical axis of the source, in its centre.

The largest contribution of neutrons is presented around the source, as neutron are transported into the graphite the fluence is reduced, however outside the total setup still there are few neutrons leaking out.

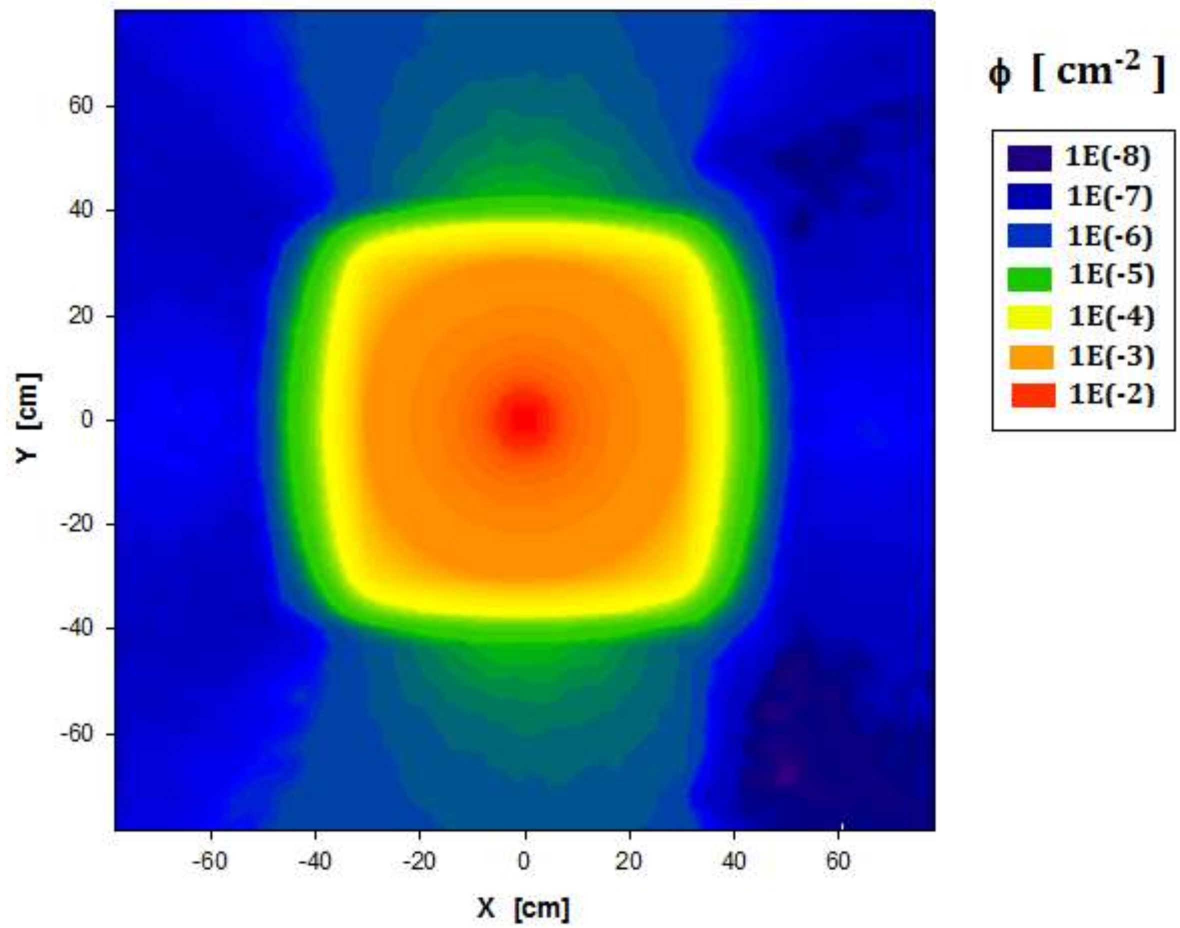

Figure 6.- Neutron fluence distribution in the horizontal plane located in the center of the source.

In the figure 7 is shown the ambient dose equivalent due to neutrons in the same plane as in the figure 6.

Here, it can be noticed that in two faces of the array, front and back, the H*(10) is larger than in the lateral surfaces, this finding will allow to improve the setup to reduce the dose in the benefit of the ^{252}Cf source users.

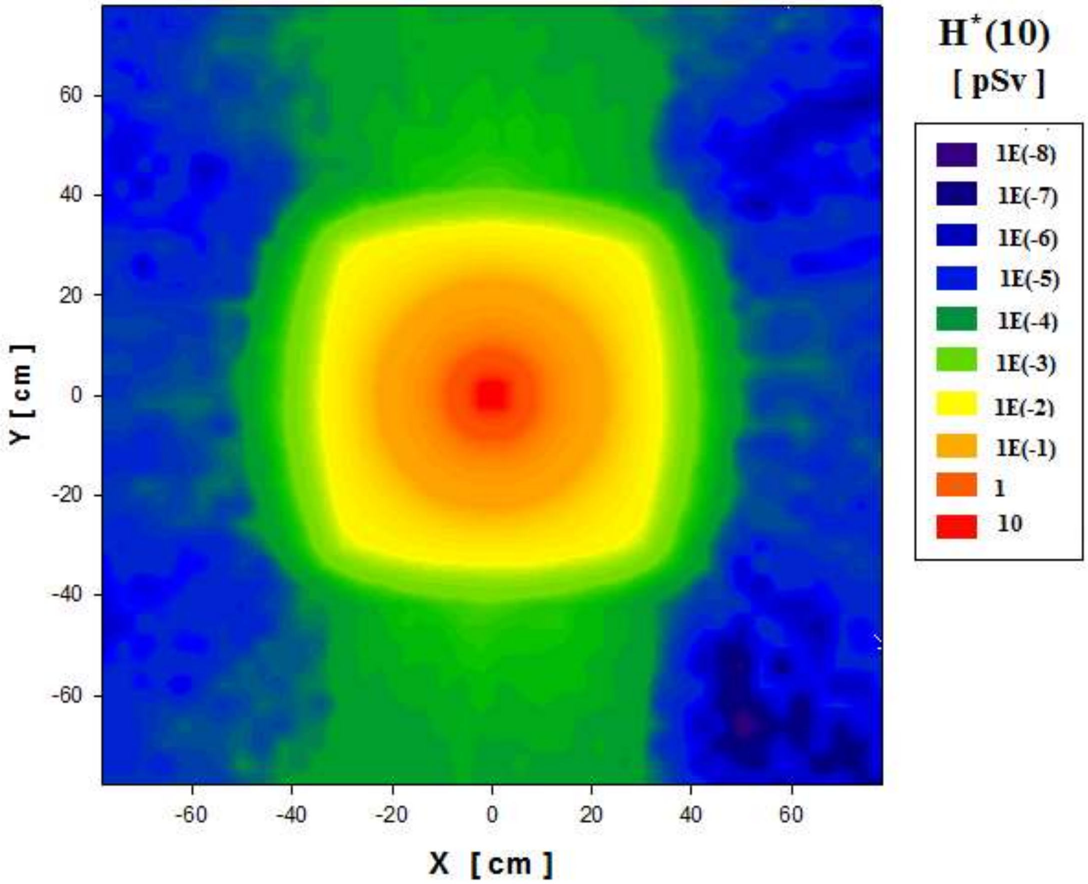

Figure 7.-H*(10) distribution in the plane located in the center of the source.

Another plane was located in the middle of the irradiation channel that is above the source. The axis of the irradiation channel is contained on this plane. In the figure 8 and 9 are shown the neutron fluence and the H*(10) distribution, per each neutron emitted by the source, respectively.

Figure 8 shows that neutrons are leaking out from the irradiation channels entrances; also, this figure confirms the neutron leakage from the front and rear surfaces of the array. The difference in the leakage is probable due that these polyethylene plates are thinner than the plates on the sides.

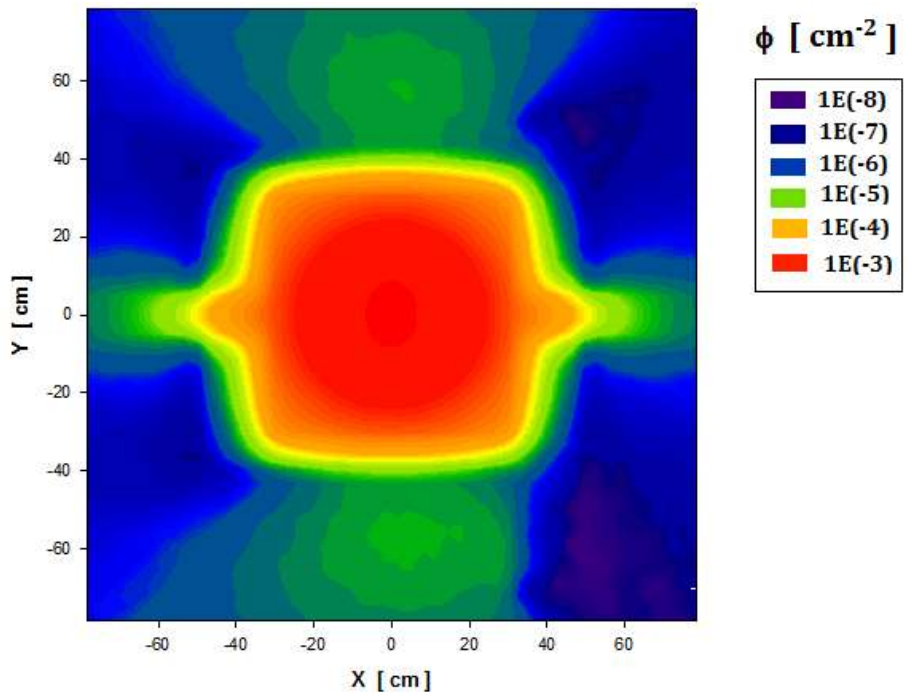

Figure 8.- Neutron fluence distribution in the X-Y plane located in the irradiation channel.

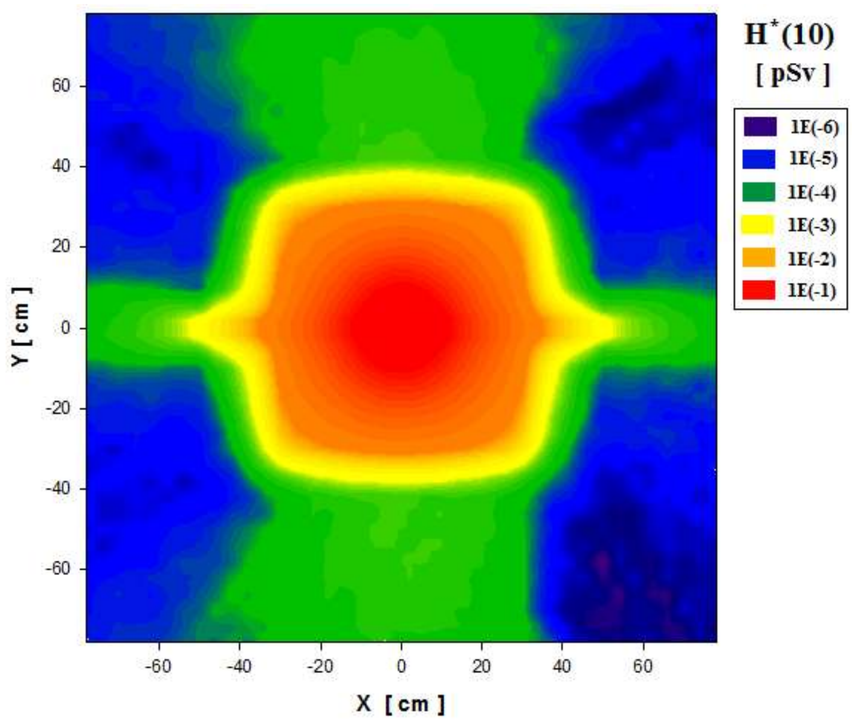

Figure 9.-H*(10) distribution in the X-Y plane located in the irradiation channel.

4.- CONCLUSIONS

Monte Carlo calculations were carried out with the aim to estimate the neutron fluence distribution in the graphite setup due to the ^{252}Cf neutron source. These will be used in the liquid fuel subcritical reactor that will be designed at the Nuclear Physics Laboratory of the Universidad Simón Bolivar in Venezuela. The setup is used for teaching and research and around it students and faculty are normally working. From the obtained results the main conclusions are the following:

- The neutron fluence distribution along the irradiation channel is symmetric from the center to the channel entrances.

- The neutron spectra in the channel have fast, epithermal and thermal neutrons. At the locations near the channel center the thermal neutrons are accompanied by epithermal and fast neutrons that can induce activation in the samples to be irradiated.

- In the front and the rear of the array there is a larger leakage of neutrons than in the lateral surfaces. Therefore in the front and the back of the array the ambient dose equivalent are larger.

- In both entrances of the irradiation channel there is a large neutron leakage and the H*(10) is also large.

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