CALCULATION OF THE NEUTRON FLUX DISTRIBUTION IN THE ACCELERATOR DRIVEN SUBCRITICAL REACTOR WITH (TH-^{233}U)O_2 AND (TH-^{235}U)O_2 MIXED FUEL

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Abstract:

This paper presents results of calculating the neutron flux distribution in an accelerator driven subcritical reactor (ADSR) with (Th-^{233}U)O_2 and (Th-^{235}U)O_2 mixed fuel. An ADSR consists of 90 fuel rods, and 10 graphite reflector rods. All objects are placed in liquid lead.

Thorium is replaced by mixture of (Th-^{233}U)O_2 and (Th-^{235}U)O_2; MCNP5 program has been used to calculate radial distribution of the neutron flux, axial distribution and energy distribution from (p,n) reaction.

The calculation results show that the axial distribution of the thermal and fast neutron flux reduce from the center of core but reduction rate is different. The thermal neutron flux decreases gradually from 0 to 2.5 cm; decreases rapidly from 25 cm to 5 cm. In comparison, the thermal neutron flux is smaller than fast neutron flux from 0 to 4 cm along the radius but the thermal neutron flux is larger than fast neutron flux at distances greater than 5 cm along the radius of the reactor.

Key words: *ADSR***,** *subcritical, neutron flux, thorium fuel*

1. INTRODUCTION

The idea of Accelerator Driven Subcritical Reactor (ADSR) was mentioned by K. Furukawa et al. [1], C. D. Bowman et al. [2] and C. Rubbia et al. [3]. There have been many studies of the structure of the ADSR. M. Hassanzadeh et al. have simulated TRIGA reactor by MCNPX program [4] as an ADSR. It contains 102 uranium fuel rods enriched 20% [4]. C.Rubbia et al. have calculated neutron flux in TRIGA [5]. The MYRRHA [6], an ADSR with thorium fuel and Bi-Pb mixture as target have been simulated by David Sangcheol Lee too [7].

ADSR is very interested in research because it has many advantages as compared to traditional nuclear reactors, such as: higher safety, the possibility of using various fuels, incinerating radioactive waste and producing energy. In this paper, we present the idea of directly using lead coolant in the reactor as a target, the proton beam will interact directly onto the lead [8,9]. There are some advantages in this way: the spallation target, instead, lead can be used not only as the coolant but also the spallation target which the proton beam from the accelerator interacts with. So, the target does not need to be changed, and the reactor is not shutdown during the operation. The entire lead which is located on the reactor will become the spallation target, the length of the target increases and thus the number of neutrons produced also increases.

The next research is the type of fuel. In this paper, thorium is replaced with (Th-^{233}U)O_2 and (Th-^{235}U)O_2 mixtures in ADSR.

2. CALCULATION AND RESULTS

In this paper, the MCNP5 program [10] is used to simulate the basic structure of an ADSR. ADS is simulated with 90 thorium fuel rods, and 10 reflectors made of graphite; all fuel rods and reflectors are placed in the liquid lead. Here, liquid lead is both a spallation target for (p,n) reaction and a coolant [8,9]. The outermost is the reflector layer with graphite. Details are described in Table 1.

Table 1: Structure parameters of the ADSR

Description	Value
Height of core	72.000cm
Diameter of core	56.000cm
Height of fuel rods	68.370cm
Diameter of fuel rods	1.818cm
Height of reflectors	68.370cm
Diameter of reflectors	1.818cm
Thickness of reflective layer	1.818cm

The MCNP5 program is used for simulation structure of ADSR, tallly Fmesh4 [10] is used in calculation, combines with equation from [4]:

$$\Phi = \frac{2\times10^{-3}C/s}{mA} \times \frac{1p}{1.6\times10^{-19}C} \times F4 \times Y_{n/p}$$

In which, Φ(ncm^{-2}s^{-1}) is neutron flux; $Y_{n/p}$ is the neutron yield.

The results of structural simulation are shown in figure 1.

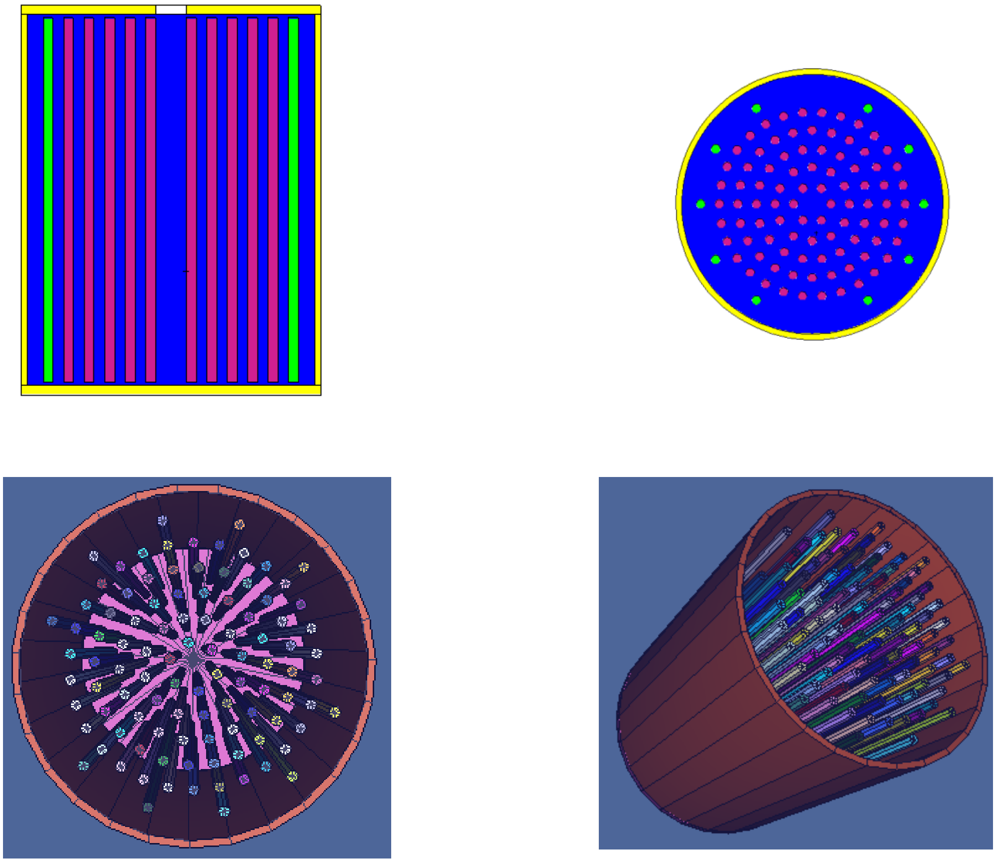


Figure 1: The position of fuel rods, reflectors inside ADSR is simulated by MCNP5

This structure is based on TRIGA reactor [4,5].

2.1. The radial distribution of the neutron flux

The calculation results are shown in figure 2.

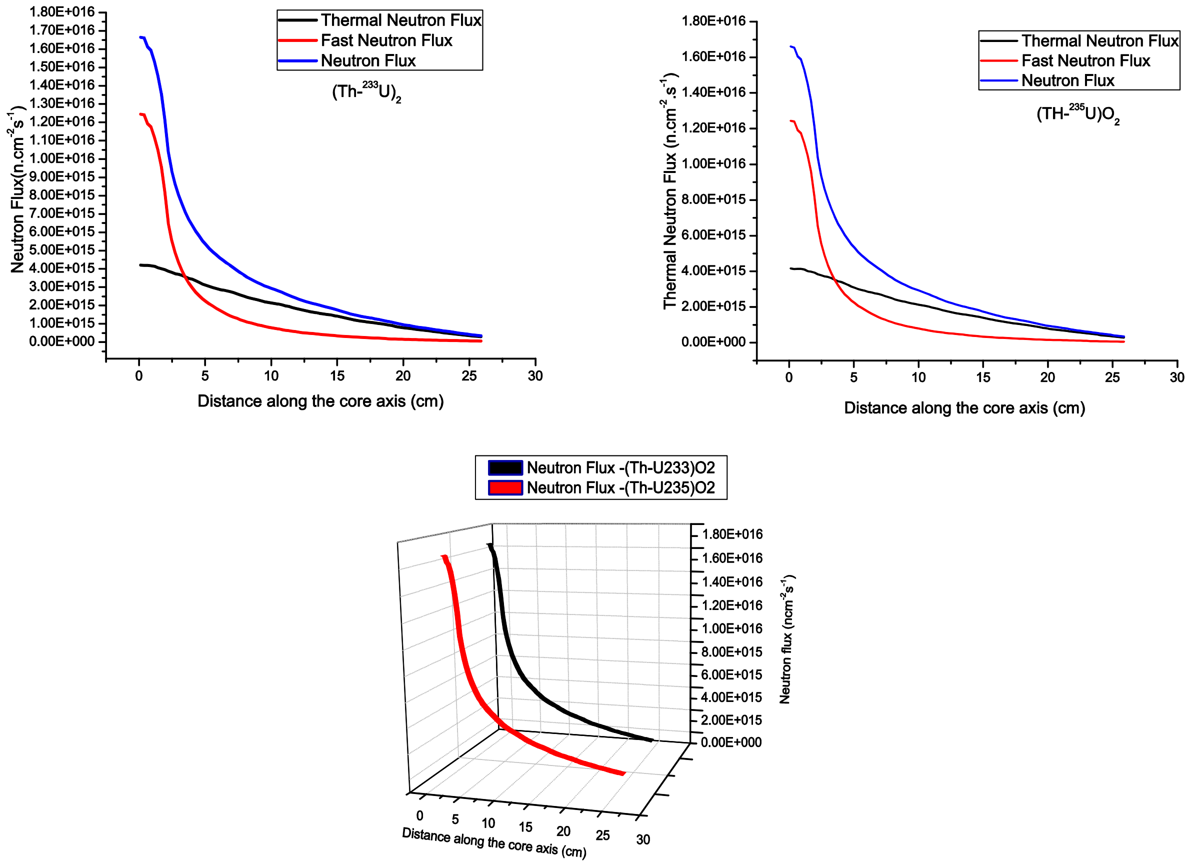


Figure 2: Radial distribution of the neutron flux

The results show that the thermal and fast neutron flux decrease from inside to outside but reduction rate is different. The thermal neutron flux decreases gradually from 0 to 2.5 cm; decreases rapidly from 2,5 cm to 5 cm. In comparison, the thermal neutron flux is smaller than fast neutron flux from 0 to 4 cm along the radius but the thermal neutron flux is larger than fast neutron flux at distances greater than 5 cm along the radius of the reactor. This can be explained because the rate of fast neutrons moderated to thermal neutron are higher as far away from the core of the reactor. In comparison, the neutron flux in the case of the (Th-^{233}U)O_2 is similar to that of the (Th-^{235}U)O_2. As comparison with results of Hassanzadeh [4] and C.Rubbia [5] in figure 3, it is similar about the shapes, but neutron flux calculated by this work is higher. This can be explained by the higher intensity of the proton beam used in this work (10 mA compared to 2 mA).

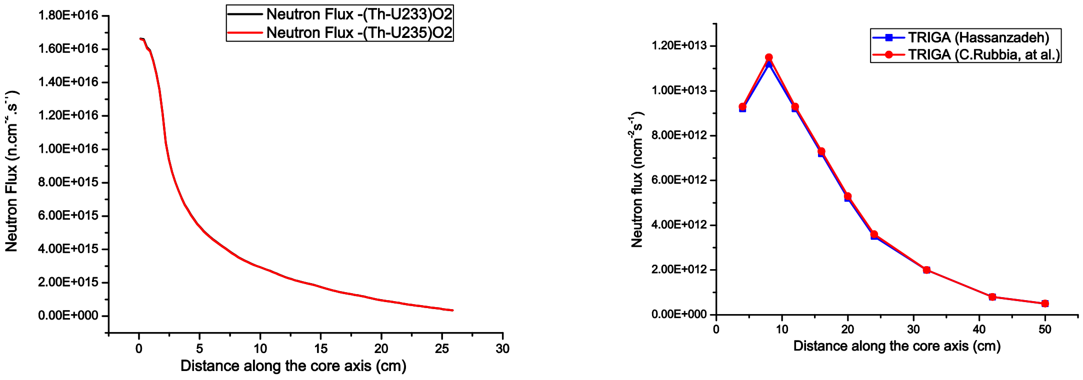


Figure 3. Radial distribution of the neutron flux from [4,5]

2.2. Axial distribution of the neutron flux

The calculation results are shown in figure 4,5.

The results show that the total neutron fluxes increase in the range of 0 to 35 cm, the maximum value is at 35 cm, then they decrease. The thermal neutron flux is higher than that of fast neutron. The calculation results of the (Th-^{233}U)O_2 and (Th-^{235}U)O_2 mixture are similar to each other, in comparison. This results are similar to previously published results [4,5] about shape but neutron flux in this work is higher. This may be also explained like the case of the radial distribution of the neutron flux.

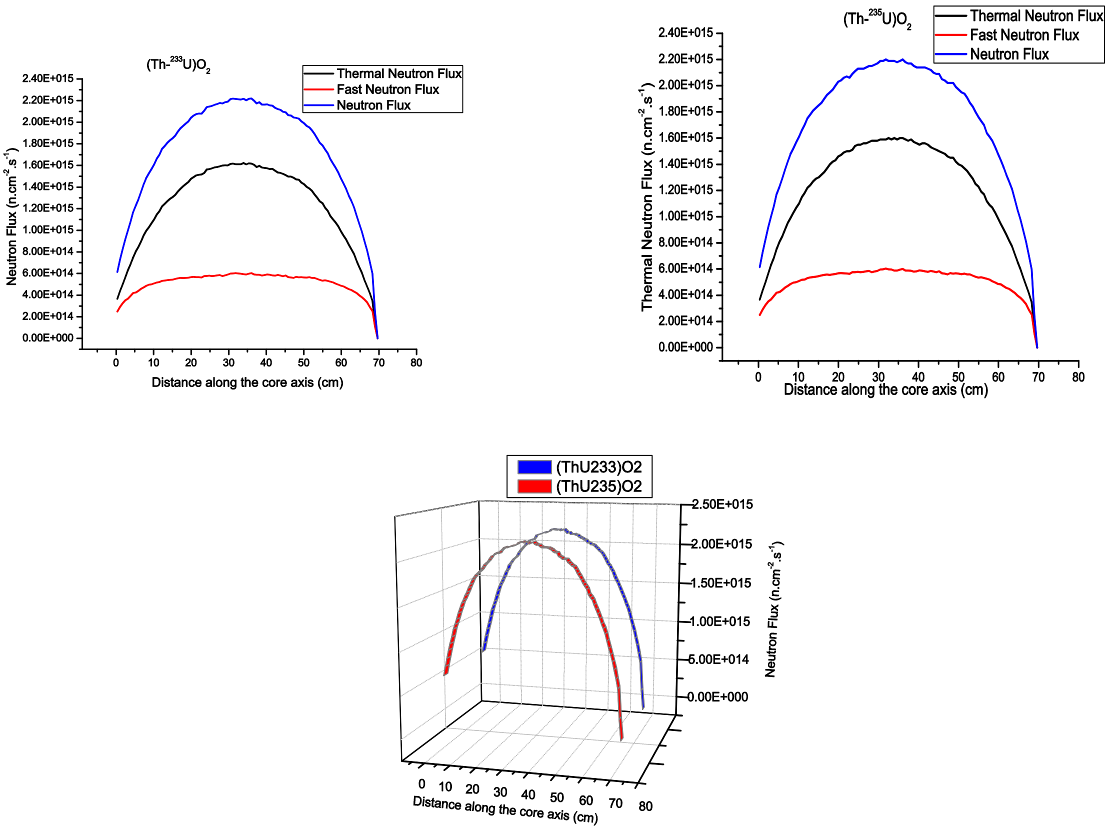
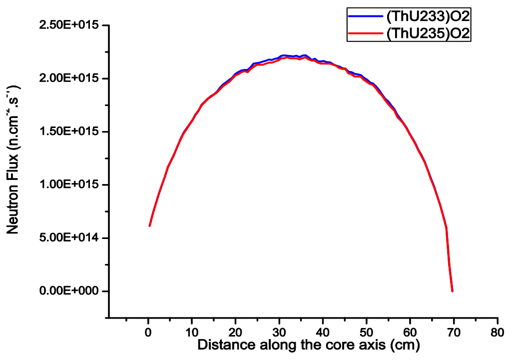
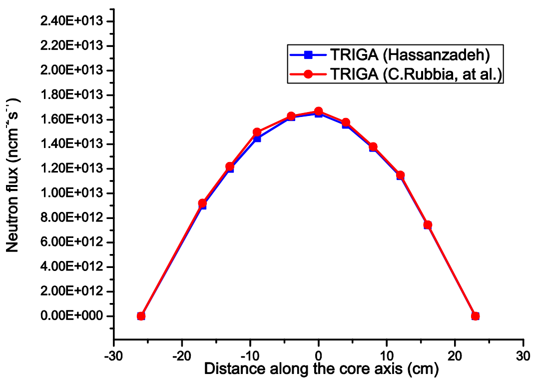


Figure 4: Axial distribution of the neutron flux



(a)



(b)

Figure 5: Axial distribution of the neutron flux in this work (a) from [4,5] (b)

2.3. Energy distribution of the neutron flux

The calculation results are shown in figure 6.

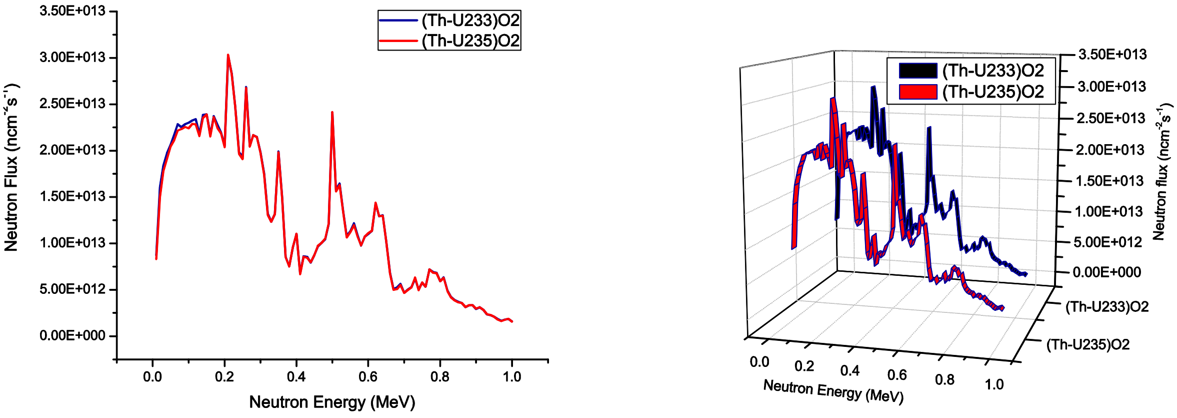


Figure 6: Energy distribution of the neutron flux

The results show that, the energy distribution of the neutron flux for two types of fuel mixtures (Th-^{233}U)O_2 and (Th-^{235}U)O_2 is quite the same. However, there are some distinct peaks in the spectrum shape. This needs further research to clarify.

3. CONCLUSIONS

By using the MCNP5 program, a basic structure of the Accelerator Driven Subcritical Reactor (ADSR) with two types of fuel mixtures (Th-^{233}U)O_2 and (Th-^{235}U)O_2 has been simulated. The results of radial distribution of the neutron flux, axial distribution of the neutron flux and energy distribution have been calculated.

The model and calculation results are reliable on the basis of comparison with published results.

These results are the basis for the further studies about the Accelerator Driven Subcritical Reactor (ADSR).

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PHÂN BỐ THÔNG LƯỢNG NƠTRON BÊN TRONG LÒ PHẢN ỨNG HẠT NHÂN DƯỚI TỚI HẠN ĐIỀU KHIỂN BẰNG MÁY GIA TỐC SỬ DỤNG NHIÊN LIỆU HỖN HỢP (TH-^{233}U)O_2 VÀ (TH-^{235}U)O_2

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Tóm tắt:

Bài viết này trình bày kết quả tính toán phân bố thông lượng nơ-tron bên trong một lò phản ứng hạt nhân dưới tới hạn điều khiển bằng máy gia tốc (Accelerator-driven subcritical reactor- ADSR) sử dụng hỗn hợp (Th-^{233}U)O_2 và (Th-^{235}U)O_2 làm nhiên liệu. Một ADSR được mô phỏng bao gồm 90 thanh nhiên liệu Thô-ri, và 10 thanh phản xạ làm bằng graphite, tất cả các thanh nhiên liệu và thanh phản xạ đặt trong môi trường chì lỏng.

Thô-ri được thay thế bằng hỗn hợp (Th-^{233}U)O_2 và (Th-^{235}U)O_2, Chương trình MCNP5 được dùng để mô phỏng cấu trúc cơ bản của một ADSR và tính toán phân bố thông lượng nơ-tron sinh ra theo bán kính lò, chiều dài của lò, năng lượng nơ-tron phát ra từ phản ứng (p,n).

Kết quả tính toán cho thấy thông lượng nơ-tron nhiệt và nơ-tron nhanh theo bán kính đều giảm dần từ trong ra ngoài nhưng tốc độ giảm khác nhau. Thông lượng nơ-tron nhiệt giảm dần từ 0 đến 2.5cm; giảm nhanh từ 2,5cm đến 5cm. So sánh giữa nơ-tron nhiệt và nơ-tron nhanh cho thấy thông lượng nơ-tron nhiệt lớn hơn nơ-tron nhanh ở bán kính từ 0 đến 4cm, từ 5cm trở đi thì thông lượng đối với nơ-tron nhanh lại lớn hơn.

Từ khóa: *ADSR , dưới tới hạn, thông lượng nơtron, nhiên liệu hỗn hợp tho-ri*