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Analyzing the Neutron Parameters in the Accelerator Driven Subcritical Reactor Using the Mixture of Molten Pb-Bi as Both Target and Coolant

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Abstract: In this paper, the Accelerator Driven Subcritical Reactor (ADSR) was simulated based on the structure of the TRIGA-Mark II reactor by the MCNPX program. The proton beam interacts on the Pb-Bi molten target with various energy levels from 0.5 GeV to 2.0 GeV. The important neutron parameters to evaluate the operability of ADSR were calculated as: the neutron yields according to various thicknesses of the target and according to the energy of the incident proton beam; the effective neutron multiplication factor for various fuel mixtures, along with its stability for some fuel mixtures; the axial and radial distributions of the neutron flux along with the height and radius of the core. The obtained results had shown a good agreement in using Pb-Bi molten as the interaction target and coolant for ADSR.

Keywords: ADSR; molten Pb-Bi; neutron yield; effective multiplication factor; neutron flux

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1. Introduction

The Accelerator-Driven Subcritical Reactor (ADSR) is a great research interest in the world since it was suggested back by many authors, such as C. Rubbia; K. Furukawa, C. D. Bowman et al. [1–3]. It can generate energy and transmute radioactive wastes cleanly and safely. The recent researches about ADSR concentrate on neutron source, designing target, and fuel.

Regarding interactive targets, the most popular studies are using solid targets to generate neutrons. In 1999, the group of authors X. Ledoux, F. Borne, A. Boudard, et al. calculated the energy spectrum of neutrons emitted at different angles when a proton beam interacts on the lead target with energies of 0.8 MeV, 1.2 MeV, and 1.6 MeV respectively [4]. Also in 1999, the group of authors S. Meigo et al. calculated the distribution of the neutron flux from a thick lead target with incident proton energies of 0.5 GeV and 1.5 GeV by the MCNP4A program [5].

Commonly used coolants are light water, molten salt, or molten Pb-Bi mixture. The group of authors Hasanzadeh et al. used the MCNP program to perform simulation of Moroccan TRIGA-Mark II Reactor and calculate Neutrons Kinetic and Parameters [6,7]. In these studies, the coolant was light water. Also from this model, Abdelaziz Darif et al. [8] calculated the dependence of the effective neutron multiplication factor (k_{eff}) on the number of reflectors in the core of the reactor. The results showed that keff can be increased without changing the size of the core. These authors also analyze neutron parameters using DRAGON.v5 and TRIVAC.v5 codes [9]. Ned Xoubi researched the neutron source for Jordan's subcritical reactor [10]. In this studying, a proton is coupled to a subcritical assembly with a lead target; the spallation neutron source is investigated; 3-D Monte Carlo simulating of radiation transport is performed for the system. These results had shown an increase of seven orders of magnitude compared to the current JSA Pu-Be source-driven core. Xue-Chao Zhao et al. conducted an analysis of Th-U breeding capability for an accelerator-driven subcritical molten salt reactor [11]. These results had shown the subcriticality of the core has a considerable effect on the Th-U breeding; a high subcriticality is favorable to improving the conversion ratio, increasing the net U-235 production, and reducing the doubling time.

There have been many proposals and calculations using liquid lead as an interactive target and coolant [12–17].

In general, there are not many studies on neutron parameters for ADSR using molten Pb-Bi as a target and coolant. In this paper, ADSR was simulated based on TRIGA-Mark II reactor structure [6–8] using the MCNPX program. From this structure, the neutron parameters were calculated, including:

- The neutron yields on molten Pb-Bi target;
- The effective neutron multiplication factors k_{eff};
- The distributions of the neutron fluxes along the height of ADSR with molten Pb-Bi target, U-Th mixed fuels;
- The distributions of the neutron fluxes along a radial of ADSR with molten Pb-Bi target, U-Th mixture fuels.

2. Model and Calculations

The TRIGA-Mark II subcritical reactor model was simulated as the Figure 1. Some characteristics of the core are shown in Table 1. The structure of the core consists of 108 fuel rods. The core is divided into 7 rings. All of these are assumed to be placed in the molten lead-bismuth medium.

Table 1. Structure of the core of the TRIGA- Mark II reactor.

Characteristics	Details
Fuels	UThO
Density of the fuel	10.32 g cm^{-3}
Coolant	Molten lead-bismuth
Reflectors	Graphite
Density of the reflectors	2.25 g cm^{-3}
Control rods	B_4C
Diameter of the fuels	3.73 cm
Height of the fuels	38.1 cm
Diameter of the core	56 cm
Height of the core	72 cm

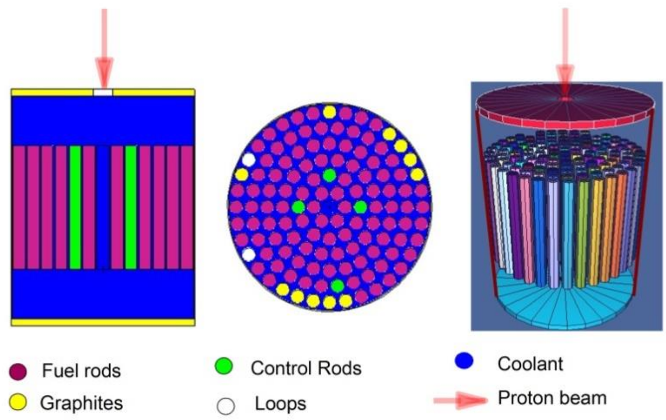

Figure 1. Structure of the TRIGA-Mark II core by MCNP code.

From this structure, the neutron parameters were calculated. The results were presented below.

2.1. The Neutron Yields ($Y_{n/p}$) from the Molten Pb-Bi Target

The proton beam from the accelerator interacts with the molten Pb-Bi target. From there, the neutron yields were calculated. The MCNPX program was used in this case. The Bertini model in MCNPX was chosen for the calculation because it is the most suitable model according to previous studies [18,19]. The interacting target was molten Pb-Bi consisting of isotopes Pb-204, Pb-206, Pb-207, Pb-208 with ratios of 1.4%, 24.1%, 22.1%, 52.4%, respectively. The molten Pb-Bi mixture consists of 44.5% lead and 55.5% bismuth.

- The neutron yields according to target thicknesses

In this section, the incident proton with 1 GeV energy interacts with the molten Pb-Bi target. From here, the neutron yields were calculated with different target thicknesses from 10 cm to 50 cm. These results were compared with previous works [20,21]. The authors Giao N. M. et al. [20] calculated the neutron yield by using data from the Japanese library JENDL-HE. Mariono et al. [21] performed calculations based on the Bertini model. These results were presented in Table 2 and Figure 2.

Table 2. The neutron yields from molten Pb-Bi target with different thicknesses from 10 cm to 50 cm.

Thickness (cm)	Neutron Yield (n/p)		
	Pb-Solid [14]	**Pb-Solid [15]**	**Pb-Bi-Liquid (This Work)**
10	10.968	9.080	9.899
20	18.852	16.087	17.616
30	22.215	20.859	22.839
40	23.926	23.480	25.862
50	24.856	24.954	27.561

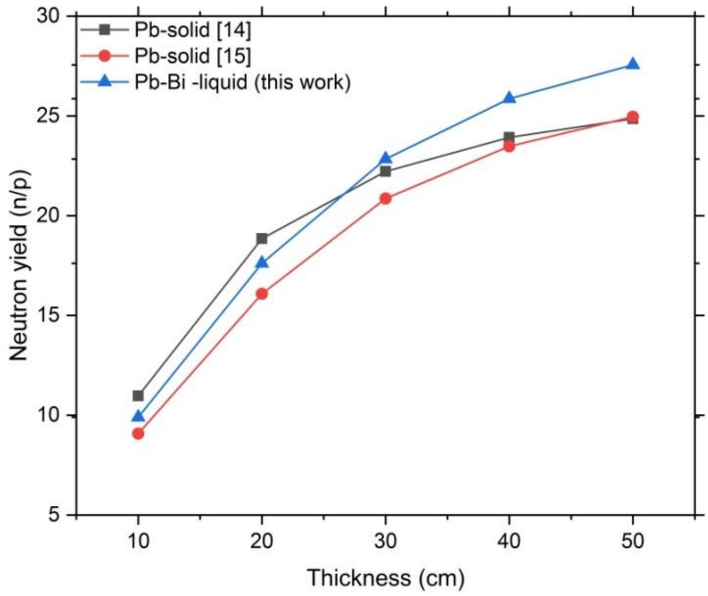

Figure 2. The neutron yields from molten Pb-Bi target with different thicknesses from 10 cm to 50 cm.

These results showed that the neutron yields on the molten Pb-Bi target by thickness were similar to that for molten lead. When the target thickness was 10 cm, the medium neutron yield was about 10 n/p. As the thickness increases, $Y_{n/p}$ also increases; at the 50 cm thickness, $Y_{n/p}$ from 25 n/p to 27 n/p. $Y_{n/p}$ for solid Pb-Bi was lower than that calculated by the authors N. M. Giao [20] et al. with thicknesses from 10 cm to 27 cm but higher from 27 cm to 50 cm. The difference was highest at 10 cm thickness with 10.8% and lowest at 30 cm thick with 2.7%. $Y_{n/p}$ for liquid Pb-Bi compared with the study of Maiorino et al. [19] on the solid lead target is always higher at all thicknesses. The highest difference was at the thickness of 50 cm with 9.5% and the lowest at the thickness of 10 cm with 8.3%. With these results, the use of Pb-Bi-liquid as a neutron-generating target in ADSR is completely appropriate. Based on these data, the designing of the ADSR core can be conducted to the requirements of each system.

- The neutron yields according to the levels of energy of the incident proton beam

In these calculations, the proton beam had energies from 0.5 GeV to 2.0 GeV. The length of the target was given to be equivalent to the height of the reactor core, 35 cm. These results were presented in Table 3 and Figure 3.

Table 3. The neutron yields from the proton beam with the levels of energy from 0.5 GeV to 2.0 GeV.

Proton Energy (GeV)	Neutron Yield (n/p)		
	Pb(Solid)	Pb (Liquid)	Pb-Bi (Liquid)
0.5	8.3540	8.5310	9.7400
0.8	16.896	17.035	19.193
1.0	21.952	21.968	24.526
1.5	32.804	32.758	36.569
2.0	42.595	42.192	46.909

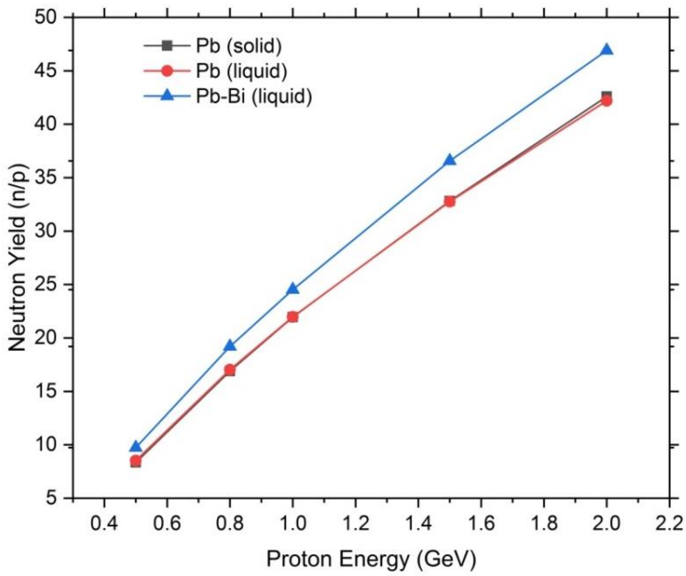

Figure 3. The neutron yields from the proton beam with the levels of energy from 0.5 GeV to 2.0 GeV.

The results have shown that the neutron yield from molten Pb-Bi target increases with the incident proton beam intensity, similar to the case of solid lead and liquid lead targets. However, $Y_{n/p}$ for the molten Pb-Bi was always higher than all with energies of the incident proton beam, from 9.2% to 14.2%. The highest difference is when compared with the case of proton beam intensity of 0.5 GeV, on the liquid lead target, with 14.2%. The lowest difference was when compared with the case of proton beam intensity of 2.0 GeV, on the liquid lead target, with 9.2%. These results further confirmed the possibility of using molten Pb-Bi for ADSR.

2.2. *The Effective Neutron Multiplication Factors k_{eff}*

For ADSR can operate and generate positive energy, the effective neutron multiplication factor k_{eff} must be greater than 0.68 [21]. In this section, the neutron was investigated for ADSR using the molten Pb-Bi and compared with the molten Pb case. The fuels used were a mixture of Th-UO with different component ratios. The stability of k_{eff} in each case was also investigated.

- Calculating k_{eff} factor

The investigated Th-UO fuel mixture consists of 8 cases, numbered from F1 to F8, according to the increasing proportion of uranium and decreasing of thorium. Details of fuel composition ratio and the results of medium keff calculation were presented in Table 4.

Table 4. Details of the ratio of components in the fuel mixture and the results of $k_{eff.}$

Fuel	$^{232}_{90}$Th	$^{233}_{92}$U	$^{16}_{8}$O	k_{eff} (Pb)	k_{eff} (Pb-Bi)
F1	0.800	0.100	0.100	0.45345	0.430634
F2	0.700	0.200	0.100	0.71546	0.708338
F3	0.610	0.290	0.100	0.90719	0.902307
F4	0.600	0.300	0.100	0.92764	0.919357
F5	0.580	0.320	0.100	0.96123	0.959649
F6	0.570	0.330	0.100	0.97984	0.974404
F7	0.565	0.335	0.100	**0.98751**	**0.979434**
F8	0.560	0.340	0.100	**1.00044**	**0.983260**

The results have shown that, as the uranium ratio increased, the thorium ratio decreased, the keff increased. With F1 fuel, the keff for both Pb and Pb-Bi was low, less than 0.5. From F2 fuel, keff reaches a value higher than 0.68. The k_{eff} for molten Pb and Pb-Bi differs on average by 0.01 with the fuel mixtures. However, with the F8 fuel mixture, the average k_{eff} was greater than 1.0 for Pb, but for Pb-Bi it is still less than 1.0. These should be considered carefully because, with Pb, the reactor has reached the critical state, while with Pb-Bi, it is still in the subcritical state.

- Stability of k_{eff} in some fuel mixtures

From the results above, some fuel mixtures were selected to evaluate the stability of the keff in some cycles. The selected mixtures were F2, F3, F5, and F7. With these mixtures, the ADSR generates positive energy and operates in a subcritical state. Critical calculations skipped the first 100 cycles, selects 16 keff values of cycles from 100 to 250 to evaluate. The results were shown in Figure 4.

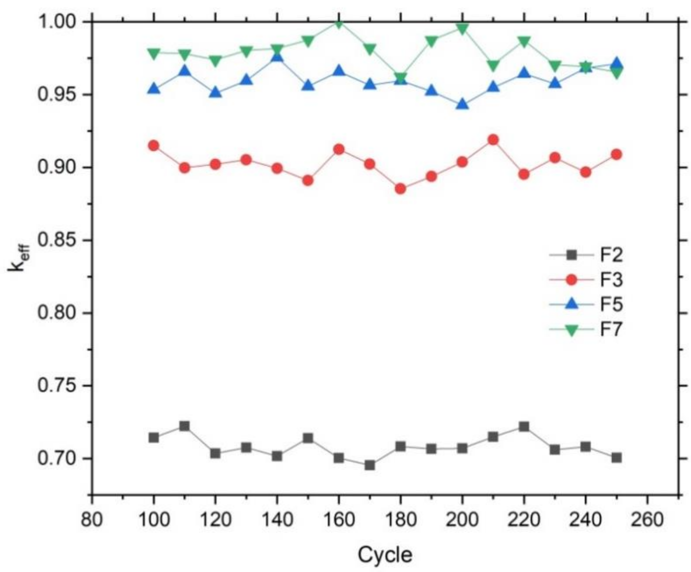

Figure 4. The k_{eff} from 100 to 250 cycles.

These results have shown that the values of k_{eff} were relatively stable throughout all cycles. For the case of the F2 fuel, the medium k_{eff} is 0.708338 then the range is from 0.69541 to 0.72226. With the F3 fuel, the medium k_{eff} is 0.902307, the values vary from 0.88528 to 0.91899. In the case of the F5 fuel, the value of k_{eff} is 0.902307, ranging from 0.94279 to 0.97565. With the F7 fuel, the mean k_{eff} is 0.979434 and ranges from 0.9621 to 1.00029. In particular, for the case of the F7 fuel, the maximum of k_{eff} is sometimes greater than 1.0. Therefore, careful studies and calculations are needed if using this fuel at this ratio.

2.3. The Radial and Axial Distributions of the Neutron Fluxes

In this section, the neutron fluxes were calculated using the F2, F3, F5, F7 fuel mixtures. The MCNP5 program and F4mesh tally were used. The neutron fluxes were calculated from the F4mesh tally in MCNPX based on Equation (1):

$$F4 = C\int \Phi(E)R(E)dE \tag{1}$$

with normalization for the current of proton beam I (mA) and neutron yield $Y_{n/p}$. Where C is a multiplication, ($ncm^{-2}s^{-1}$) is the neutron flux and R(E) is any combination of sums and products of energy-dependent quantities known to MCNPX [22]. The neutron fluxes have been determined according to Equation (2) [7]:

$$\Phi = \frac{2\times 10^{-3}C/s}{mA} \times \frac{1p}{1.6\times 10^{19}C} \times F4 \times Y_{n/p} \tag{2}$$

with $Y_{n/p}$ (n/p) is the neutron yield.

- The axial distributions of the neutron fluxes

The neutron fluxes were calculated at 20 locations along with the height of the fuel rod inside the core. The results were shown in Figure 5.

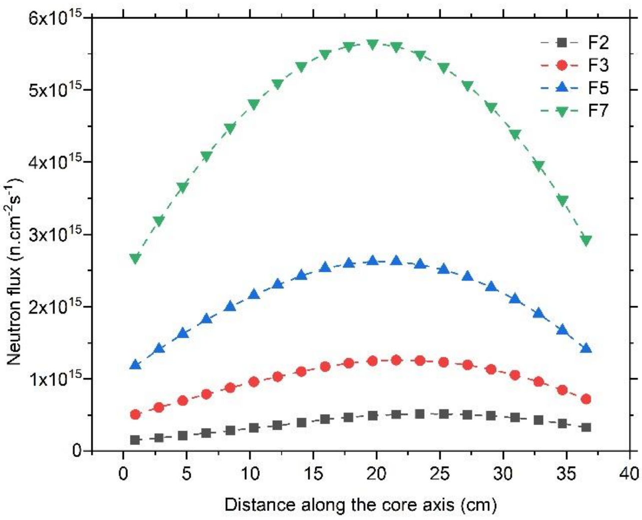

Figure 5. The axial distributions of the neutron fluxes with molten Pb-Bi target.

These results have shown that the distributions of the neutron fluxes with different fuel cases have the same form. The neutron flux was maximum value in the center of the core and decreases towards the sides. As the proportion of uranium in the mixture increased, as the proportion of thorium decreased, the maximum neutron flux increased. However, the exact maximum locations of the neutron flux were also varying. With the F2 fuel, the maximum flux was 5.15903×10^{14} ($ncm^{-2}s^{-1}$) at position h = 23.438 cm; with the F3 fuel, the maximum flux was 1.25981×10^{15} ($ncm^{-2}s^{-1}$) at position h = 21.562 cm. Particularly for the two cases F5 and F7, the maximum flux position was the same, at the same position 17.812 cm, with the maximum flux being 2.62486×10^{15} ($ncm^{-2}s^{-1}$) and 5.6457×10^{15} ($ncm^{-2}s^{-1}$), respectively. These results had shown that the greater the percentage of uranium, the closer the position of the maximum neutron flux was to the center of the core.

- The radial distributions of the neutron fluxes

The neutron fluxes were calculated at 20 locations along the radius of the core. These results were shown in Figure 6.

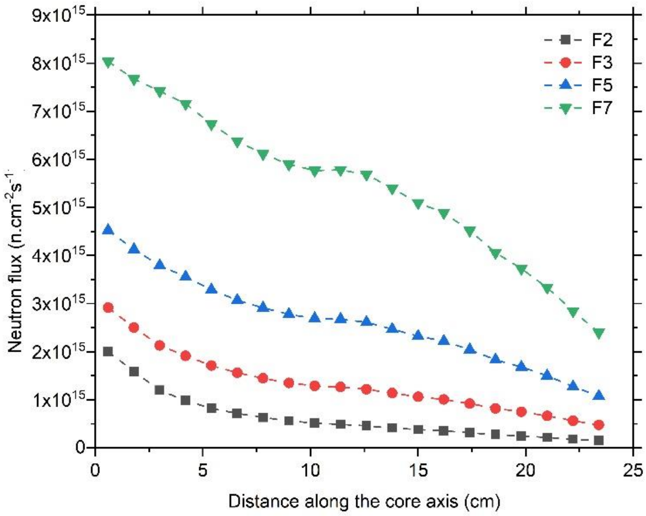

Figure 6. The radial distributions of the neutron flux with molten Pb-Bi target.

These results also showed that the maximum neutron flux at the center of the core and gradually decreases to the outside. As the proportion of uranium in the mixture increased, the maximum neutron flux increased. With the F7 fuel mixture, the maximum neutron flux reached more than 8×10^{15} (ncm^{-2}s^{-1}). The neutron fluxes decreased rapidly from the center of the core to a radius of 8 cm, then it began to decrease slowly, and from 8 cm to 12 cm, the neutron fluxes were almost unchanged. From 12 cm, the neutron fluxes continued to decrease but slower than at the beginning.

These results showed that the neutron fluxes for the molten Pb-Bi target using the mixed fuel cases F2, F3, F5, F7 were very high and suitable for the operation of ADSR. Depending on the actual conditions, requirements, and operational purposes of each ADSR, it is necessary to choose the appropriate fuel component ratio.

3. Conclusions

In this paper, the ADSR was simulated based on the structure of the TRIGA-Mark II reactor. From this structure, the neutron parameters have been calculated by MCNPX and MCNP5 programs. Here, the molten Pb-Bi mixtures were used as both a target and coolant. The results included the neutron yield on molten lead and molten Pb-Bi targets; the effective neutron multiplication factor (k_{eff}) with different ratios of fuel mixtures containing thorium and uranium; evaluate the stability of the k_{eff} in several fuel mixtures over several cycles; calculated the radial and axial distribution of the neutron fluxes. The results have shown that Molten Pb-Bi was very suitable to be used both as an interactive target and as a coolant in ADSR. There are only a few remarks about the proportions of uranium in the mixture that needs to be carefully calculated so that the ADSR generates positive energy and remains subcritical. These results contribute more data for the researching and designing of ADSR most effectively and economically.

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