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دراسة حاسوبية لتصميم حزمة نترونية حرارية في مفاعل البحث المنخفض الاستطاعة من النوع منسر للتصوير بالنترونات الحرارية

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□ ملخّص □

صممت حزمة نترونية حرارية في مفاعل البحث المنخفض الاستطاعة من النوع منسر باستطاعة حرارية 30 kWمن أجل التصوير بالنترونات الحرارية. وضعت الحزمة في بركة المفاعل وبالاتجاه العامودي والموازي للقنوات التشعيع الخارجية. أنجز تصميم مجمع النترونات باستخدام الكود MCNP4C. وأخذت قيم المقاطع العرضية في حالة طيف الطاقة المستمر لجميع المواد الانشطارية وغير الانشطارية من المكتبة ENDF/B-VI وتوابع تشتت النترونات الحرارية $s(\alpha,\beta)$ المستخدمة في الكود MCNP4C. قسم المجال الطاقي للنترونات إلى ثلاثة مجموعات حرارية، وفوق حرارية وسريعة كما يلي: أصغر من 0.4eV وما بين 0.4eV-10keV وأكبر من 10keV على الترتيب.

وللحصول على صورة عالية الجودة في نهاية الحزمة النترونية ومنع تأثير أشعة جاما على جودة الصورة، استخدم البزموت لامتصاص أشعة غاما. في هذا التصميم، كانت النسبة L/D لهذه المنظومة النترونية تساوي القيمة ١١٠. تم الحصول على تدفق للنترونات الحرارية في نهاية الحزمة النترونية من مرتبة 1.843×10^5 n/cm^2.s. يمكن استخدام هذه الحزمة النترونية في حال إنشائها في المفاعل منسر في تطبيقات علمية عديدة باستخدام التصوير بالنترونات الحرارية.

الكلمات المفتاحية:المفاعل منسر، التصوير بالنترونات، النترونات الحرارية، الكود MCNP4C.

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Computational study to design a thermal neutron beam in low-power research reactor of type MNSR for neutrons radiography

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□**ABSTRACT** □

Thermal neutron beam was designed at the low-power research reactor of type Miniature Neutron Source Reactor (MNSR) with 30 kW thermal power for thermal neutrons radiography. The neutron beam was located in the reactor pool with vertical direction and parallel the outer irradiation sites. The design of the neutrons collimator was performed using MCNP4C code. Continuous energy cross-section data of the all fissile and non-fissile materials from the ENDF/B-VI library and $s(\alpha,\beta)$ thermal neutron scattering functions distributed with the MCNP4C code. Thermal, epithermal and fast neutron energy ranges were selected as: <0.4eV, 0.4eV-10keV and greater than 10keV, respectively.

To obtain a high quality image at the beam exit and prevent the effect of the gamma rays on the image quality, the bismuth was used to absorption a gamma rays. In this design, the L/D ratio of this facility had the value of 111. The thermal neutron flux at the beam exit was about 1.853×10^5 n/cm^2.s. If such neutron beam was built into the MNSR many scientific applications would be available using the thermal neutrons radiography.

Key words:MNSR, Neutron radiography, Thermal neutrons, MCNP4C code.

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1. Introduction

Neutron Radiography (NR) is a non-destructive testing technique that is complementary to X and gamma radiography. It utilizes neutrons, which penetrate into matter deeper than the charged particles and photons. Neutrons in the specimen are attenuated by the material and detected by a two-dimensional imaging device. The image contains information about the inner structure of a given object. NR is an important tool for the study of radioactive materials.

The major advantage of NR is its ability to detect the light elements such as H, O, N, B…etc which contribute to corrosion [1], [2], [3], [4], [5].

Nuclear reactors supply high neutron fluxes, which can be used to build up the NR facility. The energy spectrum of the emitted neutrons from reactors is composed of the three neutronic energy groups, namely, thermal, epithermal and fast. The NR which uses thermal neutron is widely accepted as a good tool for inspection in industry, because thermal neutrons provide good contrasts with a large number of elements, and they are more easily available at largest neutron sources [1], [2].

The Miniature Neutron Source Reactor (MNSR) belongs to the class of tank-in-pool research reactors, with thermal power rated at 30kW. The MNSR research reactor has ten irradiation sites, five insides and five outside the annulus beryllium reflector. The maximum values of thermal neutron flux in the inner and outer irradiation sites with fresh fuel are 1.0×10^{12} n/cm^2.s, and 5.0×10^{11} n/cm^2.s, respectively. The MNSR is designed for neutron activation analysis, production of short-lived radioisotopes, and training of nuclear engineers, nuclear physicists and radiochemist's [6], [7].

2. The main problem of this research

The main problem of this research is:

- design the thermal neutron beam at the low-power research reactor of type MNSR for thermal neutrons radiography.
- calculate the thermal neutrons flux and gamma dose at the beam exit.

3. The importance of this research

The importance of this research is using the Monte Carlo method (MCNP4C code [8]) in the neutronic calculations in the low-power research reactors, this helps in:

1. Increasing the scientific experience and knowledge in neutronic and criticality calculations of low-power research reactors.
2. Determining the neutronic and criticality parameters of the reactor before making any modification in the structure of the reactor core (such as: replacing a depleted fuel rods with new fuel, replacing the control rods and design neutronic channels for various scientific applications…etc.).

2. Methodology

2. 1 Monte Carlo Model of the Syrian MNSR.

The 3-D Monte Carlo MCNP4C model of the MNSR was simulated to estimate the criticality parameters such as: excess core reactivity, shutdown margin, control rod worth and thermal neutron flux in the outer and inner irradiation sites in the reference [9].

The cross-section of the MNSR core which comes out from MCNP4C code is shown in Figure 1. The vertical cross-section of the Syrian MNSR with the control rod totally inserted in the core is illustrated in Figure 2.

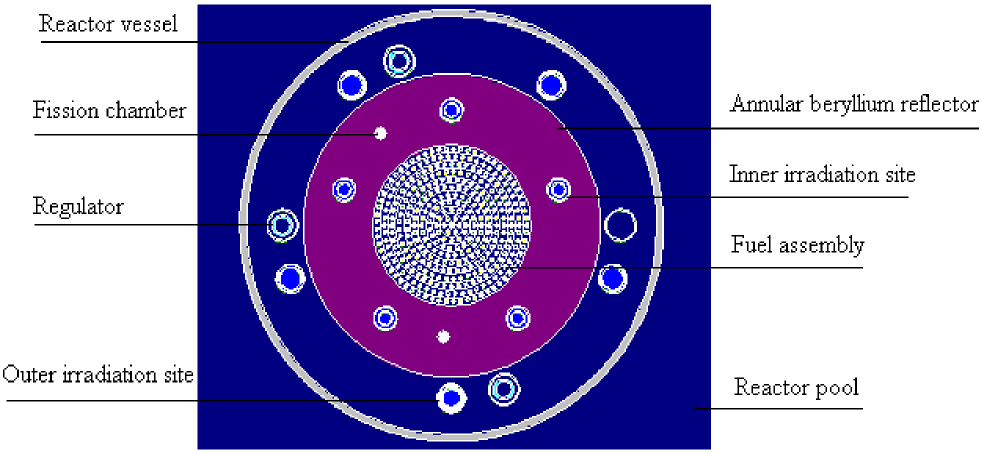

Figure 1. Horizontal cross-section of the MNSR core using MCNP4C code.

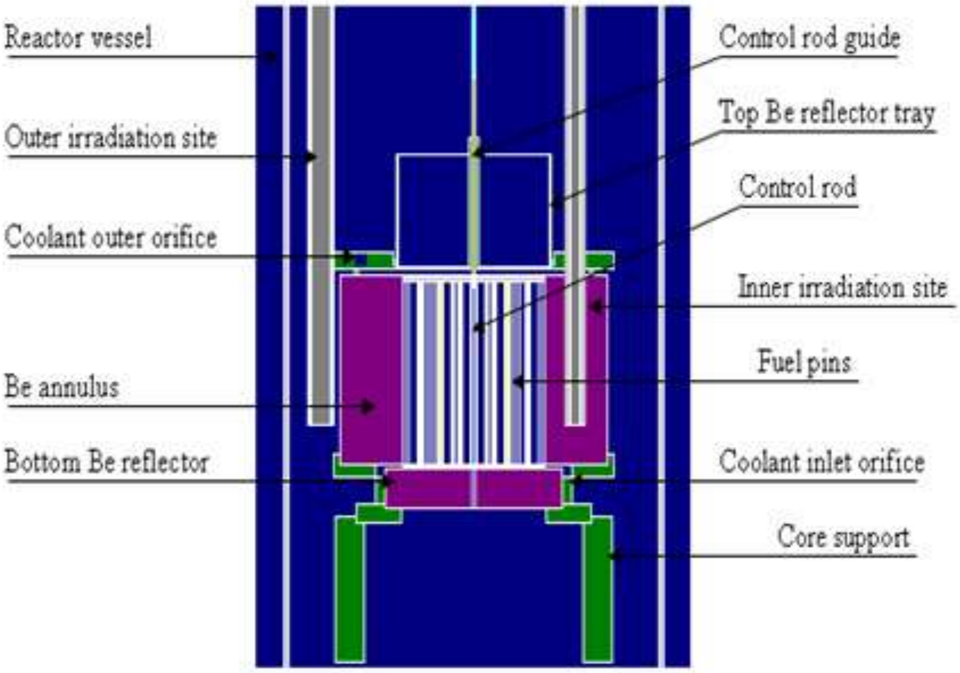

Figure 2. Vertical cross - section of the MNSR core using MCNP4C code.

3. Design of the thermal neutron beam for neutron radiography at the MNSR using MCNP4C code.

3. 1 Collimator model

The NR facility will locate in the reactor pool with vertical direction and parallel the outer irradiation sites. The collimator of neutrons will penetrate into the reactor pool as shown in Figure 3 and Figure 4. This collimator consists of Two Rectangular Tubes (TRT):

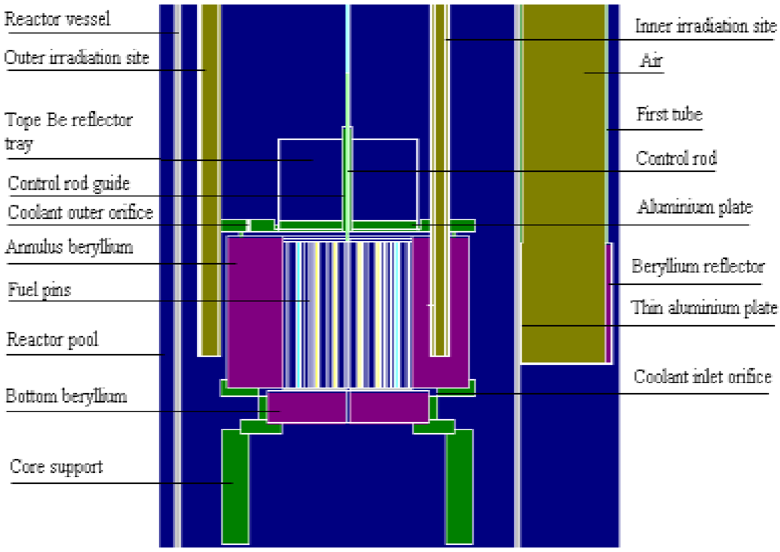

Figure 3: Vertical cross section of the MNSR core with the first tube of the neutron radiography facility in the plane Y-Z using MCNP4C code.

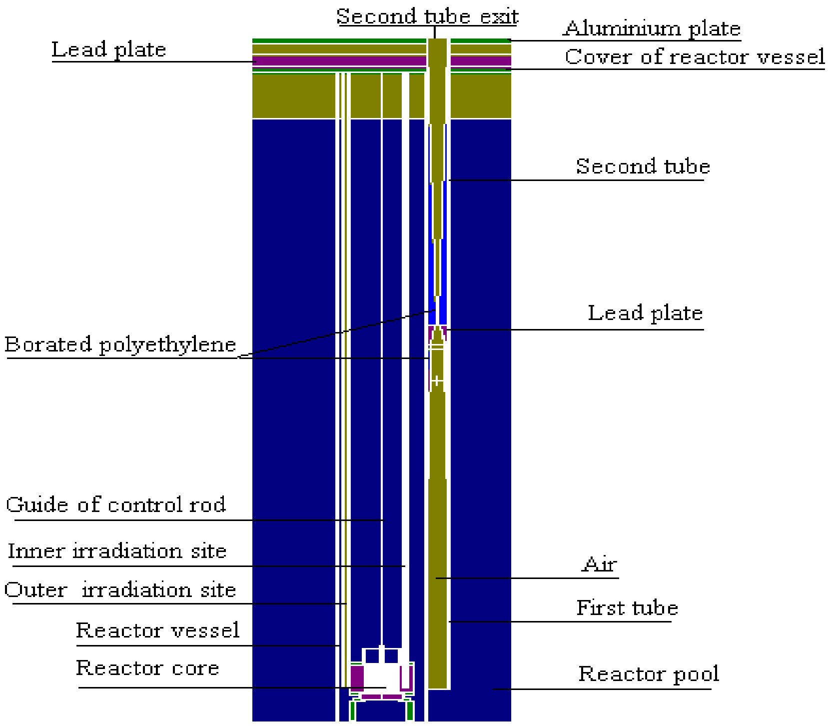

Figure 4: Sketch of the collimator of the neutron radiography facility in the plane Y-Z at the Syrian MNSR using MCNP4C code.

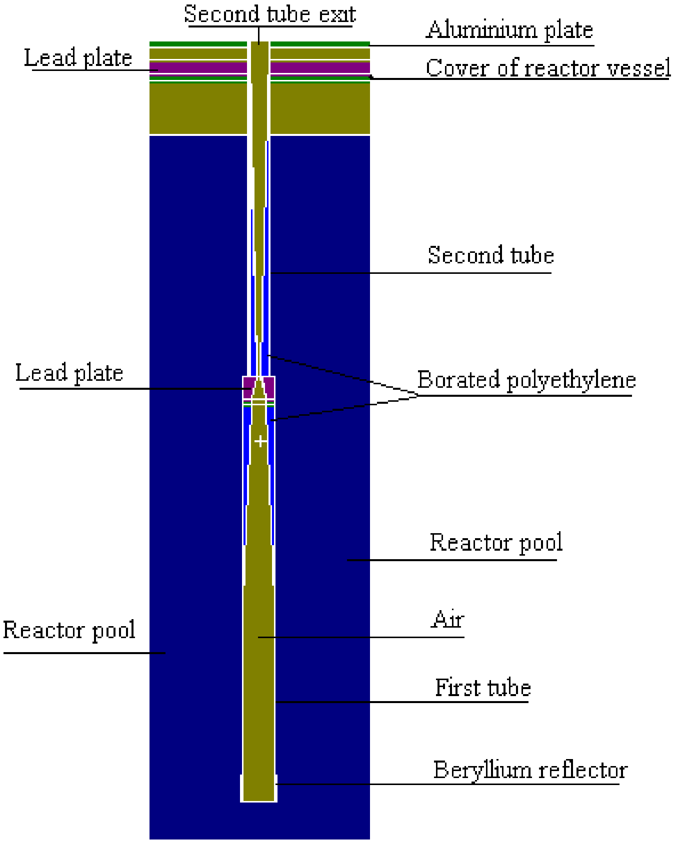

Figure 5: Sketch of the collimator of the neutron radiography facility in the plane X-Z at the MNSR using MCNP4C code.

The dimensions of the first and the second tubes are 150.0 (wide) mm x 230.0 (length) mm x 3105.0 (height) mm and 150.0 (wide) mm x 150.0 (length) mm x 2510.0 (height) mm, respectively as shown in Figure 3, Figure 4 and Figure 5. The thickness of walls of the first and the second tubes is 2.0 and 3.0 mm, respectively.

The wall of the first tube is very thin (with 0.3 mm in thickness and 190.0 mm in height) from the side of the reactor core as shown in Figure 3 and Figure 4. The distance between the reactor core center and the second tube exit is 5615.0 mm.

To improve the neutron flux at the second tube exit, the beryllium, heavy water and graphite materials were used as reflector at the bottom of the first tube as shown in Figure 3, Figure 4 and Figure 5 with thickness 10.0 mm and 230.0 (length) mm x 190.0 (height) mm. These dimensions are optimized using MCNP4C code and the obtained results are given in Table 1.

Table 1: Calculated values of the neutron flux at the second tube exit as function of the dimensions of the first tube.

Dimensions (height) mm × (length) mm	Neutron flux at the second tube exit n/cm^2.s
190×230 and beryllium reflector	$(5.749 \pm 0.032) \times 10^6$
190×230 and heavy water reflector	$(5.540 \pm 0.019) \times 10^6$
190×230 and graphite reflector	$(5.366 \pm 0.018) \times 10^6$

From Table 1 the increase in the value of the neutron flux at the second tube exit is 20.7, 16.3 and 12.6% for beryllium, heavy water and graphite, respectively. Therefore, the beryllium reflector is used at the bottom of first tube.

The Initial Converging Cone Section (ICCC) with height 1700.8 mm is mounted inside the first tube and used as elementary collimator of neutrons as shown in Figure 5. The converging section helps to properly define the aperture by removing the neutrons not converging to the aperture. In addition, the Diverging Cone Section (DCC) with a height 2510.0 mm is mounted inside the second tube and used as main collimator of neutrons as shown in Figure 5. The walls of the ICCC and DC are made of beryllium and Al-6061 alloy with a thickness 3.0 mm and 1.0 mm, respectively.

Table 2 shows the difference in the calculated values of the neutron flux at the second tube exit as function of the type of the walls material of the ICCC. From this table it results that beryllium increased the value of the neutron flux by 5.4% as compared with Al-6061 alloy.

Table 2: Calculated values of neutron flux at the second tube exit as function of the type of the wall material of the elementary collimator of neutrons.

Type	N Neutron flux at the second tube exit n/cm^2.s
Walls of beryllium	$(3.187 \pm 0.063) \times 10^5$
Walls of aluminum (6061-alloy)	$(3.020 \pm 0.062) \times 10^5$

The space between the walls of the ICCC and the DC and the walls of the TRT is filled with borated polyethylene (1.0 % boron) and lead at the upper surface of the reactor pool as shown in Figure 6 and Figure 7.

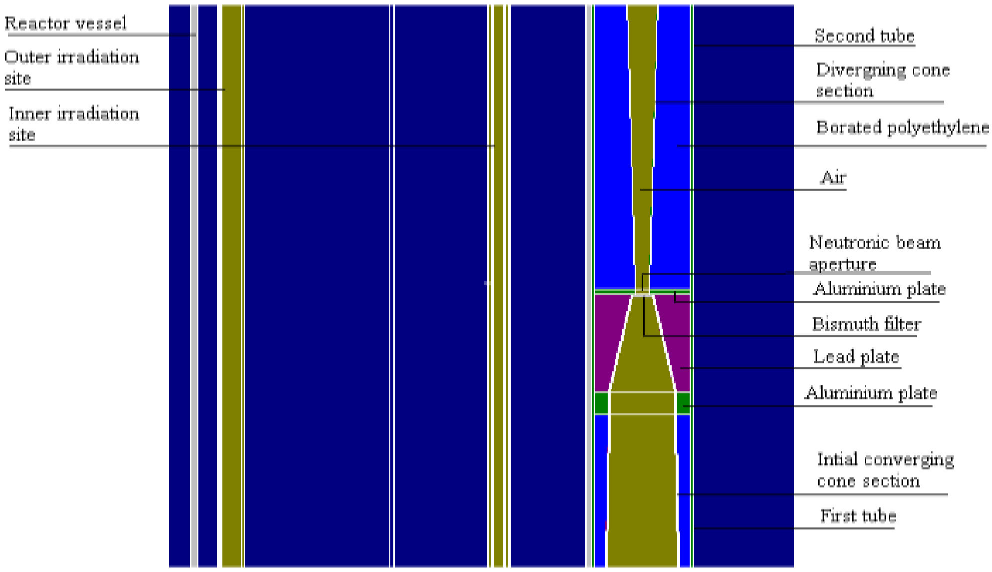

Figure 6: Vertical cross section of the ICCC and DCC of the neutron radiography facility in the plane Y-Z using MCNP code.

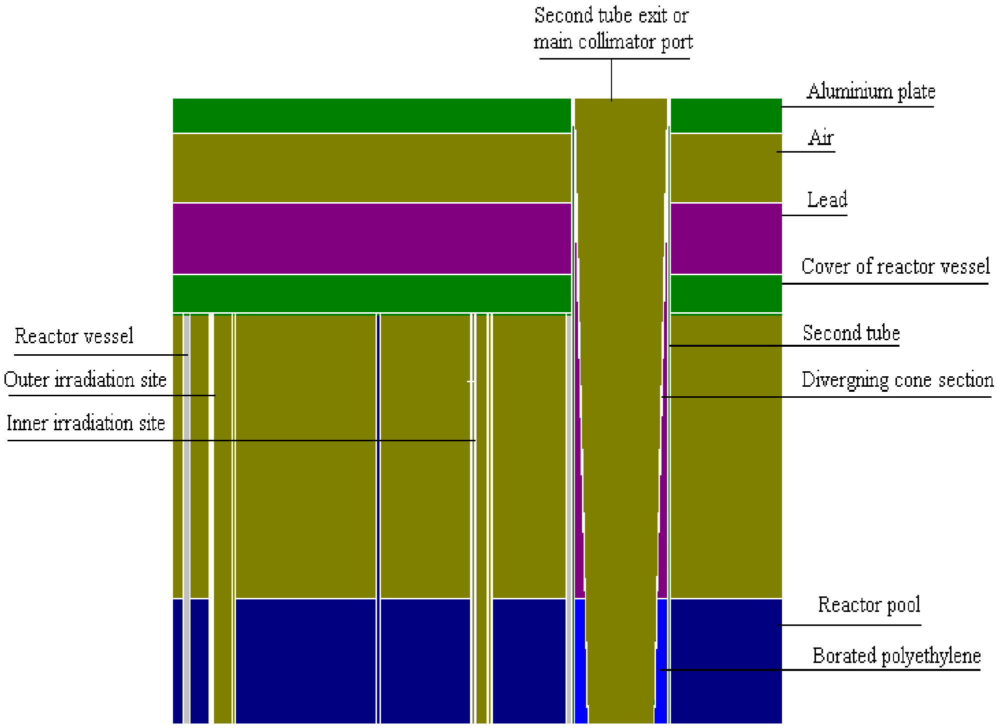

Figure 7: Vertical cross section of the end of the second tube with DCC of the neutron radiography facility in the plane Y-Z using MCNP4C code.

At the end of first tube and the beginning of the main collimator of neutrons, the lead plate with thickness 170.0 mm is used for absorbing the gamma rays at the neutron beam aperture D as shown in Figure 6.

3.2 Beam filter.

***Photon filters*: **The bismuth is a good material for shielding gamma rays since it has a neutron capture cross section smaller than that of lead and with nearly identical gamma-ray attenuation coefficient [1], [10], [12]. Therefore, the gamma-ray background emitted by the reactor is reduced by a bismuth filter which is mounted at the end of the elementary collimator of neutrons as shown in Figure 6.

3.3 Position of the aperture of the NR facility.

The location of the aperture in the beam tube was decided by the simple similar triangle analysis as shown in Figure 8 [1].

From Figure 8 it is clear that.

$$\frac{d1}{d2} = \frac{L1}{L2 - L1}$$

With d1 = 190.0 mm and the beam size at 5615.0 mm chosen to be 150.0 mm (d2 = 150.0 mm) to accommodate large size film, the aperture location can be obtained to be L1 = 3180.0 mm (from the bottom of the first tube). The length of the main collimator of neutrons is L = L2 – L1 = 5690.0 – 3180.0 = 2510.0 mm and defined as the distance from the neutron beam aperture to the main collimator port (second tube exit). Also, based on the above analysis the divergence of the collimator is chosen to be 1.2^{o}.

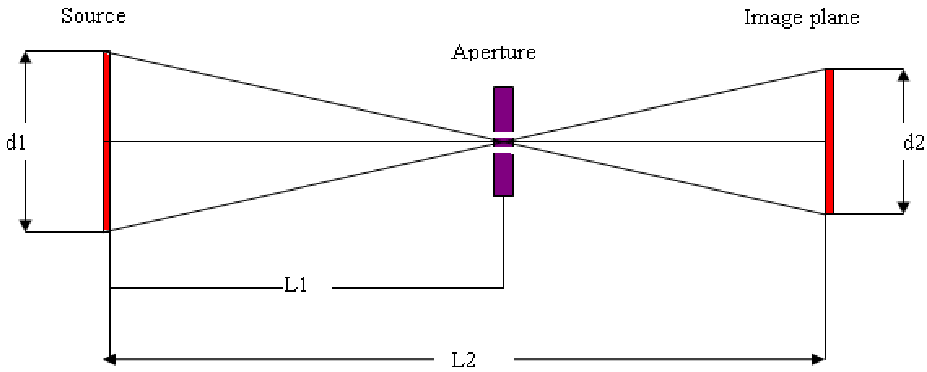

Figure 8: A sketch showing the position of the neutron beam aperture.

4. Parameters selection of the thermal neutronic beam and neutron flux.

The design of the thermal neutron beam aimed at obtaining a thermal neutron flux $\geq 1.0\times10^5$ n/cm^2.s at full reactor power (30 kW) with the maximum Thermal Neutron Content (TNC), the least photon content $Ø_{th}/D_\gamma \geq 1.0\times10^6$ n/cm^2.mR (where**:** $Ø_{th}$, D_γ and mR – is the value of the thermal neutron flux and gamma-ray dose at the main collimator port and milli Rem, respectively), and a value of L/D ratio equal to 111.

Where:

L - is the length of the main collimator of neutrons.

D - is the diameter of the neutron beam aperture.

To evaluate these parameters, dimensions of the photon filter and excess core reactivity ρ of the MNSR with NR facility, both MNSR and the collimator of neutrons were simulated using MCNP4C code, starting each time with fission neutrons from the homogenized core and traveling to the main collimator port of neutrons.

The distance between the reactor core center and the main collimator port (second tube exit) of neutrons is 5615.0 mm. The number of fission neutrons in the reactor core were normalized to 2.7×10^{15} n/s for the operating power of 30 kW. The obtained results of the thermal, epithermal, fast neutron flux, TNC ratio, $Ø_{th}/D_\gamma$ ratio (at the main collimator port), dimensions of the photon filter are given in Table 3 and Table 4.

The position, the diameter of the neutron beam aperture and the thickness of Bi were varied using MCNP4C code to have a high value the TNC at the main collimator port. The obtained results are given in Table 3.

Table 3: Geometrical parameters of the neutron radiography facility at the MNSR.

Type	Length of the main collimator L (mm)	Diameter of aperture D (mm)	L/D	Diameter of the main collimator port (mm)
	2510.0	22.60	111	150×150 (mm×mm)
Gamma-ray filter				
Bismuth	Thickness (mm)	Dimensions of filter (mm×mm)		
	3.0	30.0 × 30.0		

Table 4: Calculated values of the thermal, epithermal and fast neutron flux, TNC, $Ø_{th}/D_\gamma$ and angle of beam divergence at the main collimator port (at the second tube exit).

Type	Calculated value
Thermal neutron flux $\times10^5\ n/cm^2.s$	1. 853 ± 0.0774
Epithermal neutron flux $\times10^3\ n/cm^2.s$	4.672 ± 0.0378
Fast neutron flux $\times10^3\ n/cm^2.s$	3.055 ± 0.0189
TNC %	96.47 ± 0.072

$Ø_{th}/D_{\gamma}\times10^{6}$ (n/cm^2.mR) [(a)]	7.8401 ± 0.1502
Angle of beam divergence $2\theta^{o}$	1.2^{o}

Where:

[(a)] $Ø_{th}$ and D_{γ} is the thermal neutron flux and gamma dose at the main collimator port, respectively.

5. Results and discussions

The Calculated values of the thermal, epithermal and fast neutron flux, TNC, $Ø_{th}/D_{\gamma}$ and angle of beam divergence at the main collimator port of neutrons are summarized in Table 4. To calculate these values at the main collimator port, the input file of the MNSR reactor was run by the MCNP4C code using the E, the F_4, the FS, the SD tallies [11], [12].

In the MCNP4C code, tallies are normalized per source particle. The flux tally will then be in units of neutrons/(cm^2.source particle), and this will give the correct spectral shape of the neutron scalar flux but not the correct magnitude of the flux. The normalized flux can be calculated using the average number of neutrons produced per fission $\tilde{\nu}$, the reactor operating power (P in MW) and the MCNP4C flux tally normalized per source neutron as [11], [12]:

$$\varphi = \varphi_{MCNP4C} \times P(MW) \times \tilde{\nu}\left(\frac{Neutrons}{Seconds}\right) \times \left(\frac{1MeV}{1.6022 \times 10^{-13}Joules}\right) \times \left(\frac{Fission}{E = 200MeV}\right)$$

Where:

φ_{MCNP4C} - is the neutrons flux calculated by MCNP4C code.

$\tilde{\nu} = 2.43$ - is the average number of neutrons produced per one fission.

$P = 30kW$ - is the reactor power.

$E = 200MeV$ - is the average fission energy of isotopes formed in the nuclear fuel.

Form Table 3, the thickness of bismuth crystal was optimized using MCNP4C code to have a low intensity of the gamma-ray at the main collimator port. This condition ensures that gamma-ray contribution to the generation of the image will be small relative to that from neutrons.

The value of the TNC ratio is about 96.47 % at the main collimator port as shown in Table 4; this value is defined as the ratio of thermal neutron flux below 0.4eV to the total neutron flux in the range (0-20) MeV. In addition, this value ensures that the image of the sample being studied is arising essentially from thermal neutrons.

The angle 2θ of beam divergence is an important measure of the usefulness of the beam near its periphery. If the neutron beam diverges very rapidly to a large size, then the outer portion of the images produced will suffer significant distortion. Conversely, if the beam is long or if the image size is small then the outer portion of the image will be less distorted [12].

From Tables 4 the angle of the beam divergence and the image size at the main collimator port is small. In addition, the length of the main collimator of neutrons is long. Therefore, as a result, the outer portion of the image will be less distorted and this image will have good contrast [12].

Where the θ is calculated from the following equation [12].

$$\tan\theta = \frac{0.5(R - D)}{L}$$

Where:

R - is the radius of the main collimator port and equal to 7.5 cm (See the Table 3).
D - is the diameter of a neutronic aperture and equal to 2.26 cm (See the Table 3).
L - is the length of the main collimator L and equal to 251 cm (See the Table 3).

In the calculated of the D_γ gamma dose, the input file of the MNSR reactor was run by the MCNP4C code using the E, the F_4, the FS, the SD tallies, MODE N P (N and P is indicating to Neutrons and Photos, respectively) and the KERMA factors were used to calculate the D_γ with energy from 0.01 MeV to 10 MeV [12]. The calculated value of the D_γ is $(2.236 \mp 0.012) \times 10^{-2}$ mR (Millie Roentgen).

Figure 9 shows the distribution of the thermal neutron flux along the X and Y- axis at the collimator port using MCNP4C code.

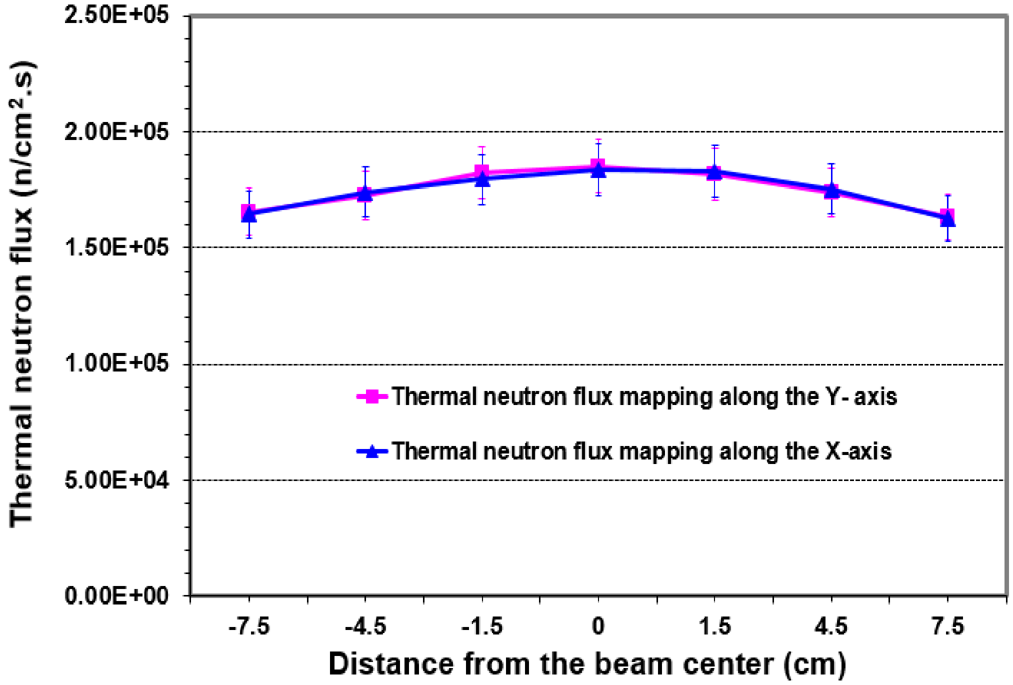

Figure 9: Distribution of the thermal neutron flux along the X and Y- axis at the main collimator port.

From Figure 9, can be seen that there is a good agreement between the calculated values of the thermal neutron flux along the X and Y-axis. The maximum difference between the calculated values is 3.74 %.

The good homogeneity between the calculated values of the thermal neutron flux is very important for getting a clear image of the studied sample.

5. Conclusions

The MCNP4C code used to design the neutron collimator of the NR facility at the MNSR reactor in the vertical irradiation sites. The reflector of beryllium was used at the bottom of the beam tube of the neutrons collimator and for the walls of the ICCC to increase the value of the neutron flux at the main collimator port. The value of the thermal neutron flux estimated at the beam exit was about 1.853×10^{5} n/cm^2.s. The construction of this beam will open the door for many scientific applications of the NRG in MNSR reactor.

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