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نمذجـة مفـاعل البحث منسـر المنخفض الإسـتطاعة باسـتخدام الكود MCNP-4C2

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□ ملخّص □

أُستخدم الكود MCNP-4C2 لنمذجة المفاعل منسر المنخفض الإستطاعة و المحمل بوقود عالي الإغناء باليورانيوم ٢٣٥. و تم حساب الوسطاء الحرجية مثل: معامل التضاعف الفعال k_{eff}، و التفاعلية ρ_{ex}، و وثوقية قضيب التحكم (CRW) و هامش الإغلاق (SM). بالإضافة إلى ذلك، حساب تدفق النترونات الحرارية في موقع التشعيع الداخلي (IIS) و موقع التشعيع الخارجي (OIS). و قورنت القيم المحسوبة لكل من الوسطاء الحرجية و تدفق النترونات الحرارية في موقع التشعيع الداخلي (IIS) و موقع التشعيع الخارجي (OIS) مع القيم التجريبية فوجد تطابق جيد. توضح هذه النتائج مدى التوافق القريب ما بين نظرية النقل العشوائية أي طريقة مونتي كارلو المبنين على أساسها الكود MCNP-4C2 و القياسات التجريبية و المتعلقة بمفاعل البحث منسر المنخفض الإستطاعة.

الكلمات المفتاحية: المفاعل منسر، الوسطاء الحرجية و النترونية و الكود MCNP-4C2.

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Simulation of the Low-Power Research MNSR Reactor by Using the MCNP-4C2 code

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□ABSTRACT □

The MCNP-4C2 code has been applied to simulate the low power Miniature Neutron Source Reactor (MNSR), fueled by the High Enrich Uranium – 235 (HEU). The critically parameters such as: the effective multiplication factor k_{eff}, Reactivity ρ_{ex}, Control Rod Worth (CRW) and Shutdown Margin (SM) were calculated. In addition, the thermal neutron flux in the Inner Irradiation Site (IIS) and Outer Irradiation Site (OIS) was calculated. The calculated values of the critically parameters and thermal neutron flux in the IIS and OIS were compared with experimental values and good agreements was found. These results illustrate the close agreement between stochastic transport theory (Monte Carlo method - MCNP-4C2 code) and the experimental measurements conducted of the low-power research MNSR reactor.

Key words: MNSR reactor, critically and neutronic parameters, MCNP-4C2 code

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1. Introduction

1.1 Miniature Neutron Source Reactor (MNSR).

The MNSR reactor belongs to the class of tank-in-pool research reactors, with thermal power rated at 30 kW. MNSR reactor uses light water as moderator, coolant and shield and beryllium as reflector. Vertical and frontal cross-sections of the MNSR reactor are shown in Figure 1 and Figure 2.

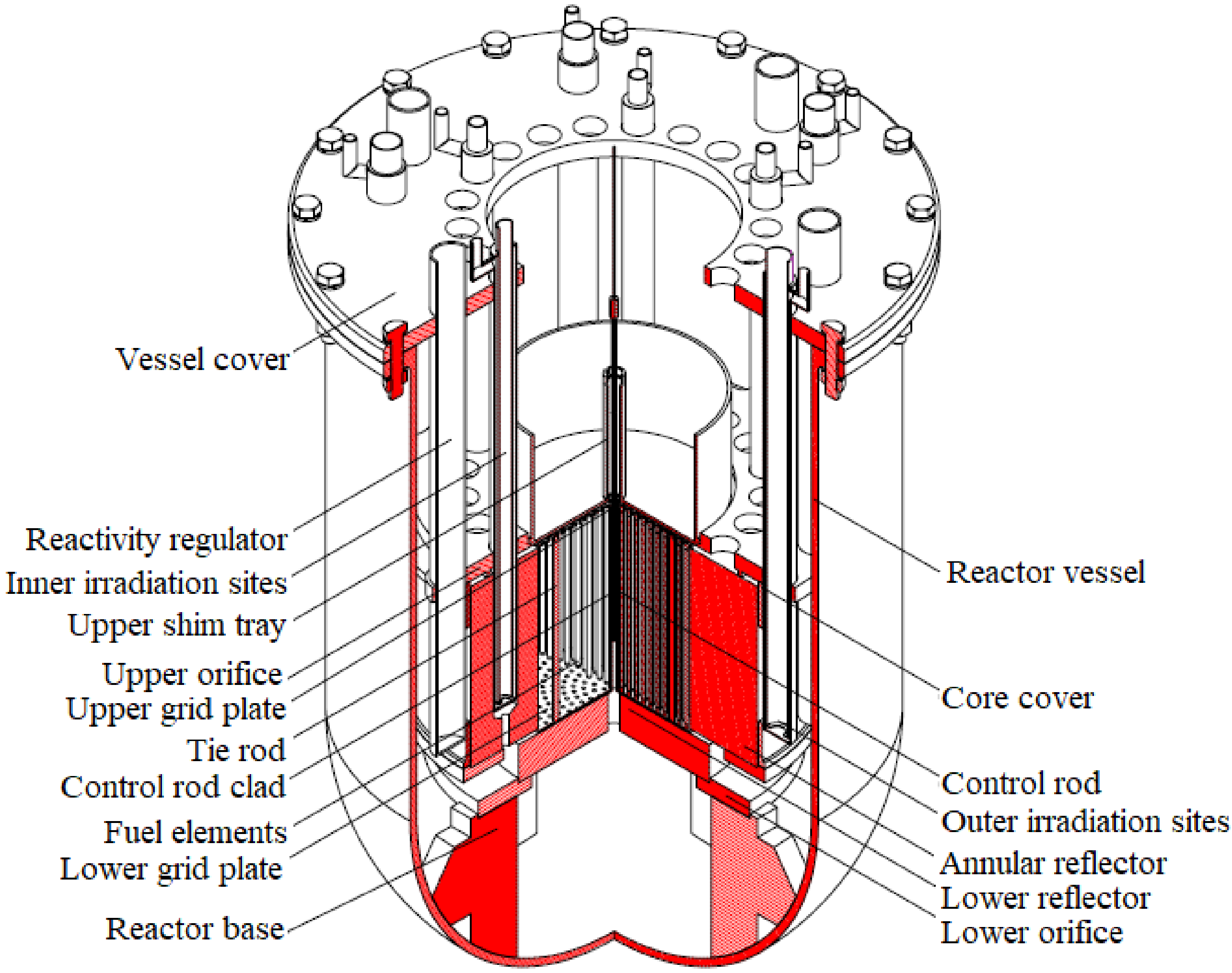

Figure 1: Vertical cross - section of the MNSR reactor by using AutoCAD and 3DMAX program.

As shown in Figure 1 and Figure 2, the reactor core consists of fuel elements which form the fuel assembly. The assembly is inside an annular beryllium reflector and rests on a lower beryllium reflector plate to minimize neutron losses and conserve neutron economy. The fuel elements are made of U-Al alloy with aluminum clad and enriched fuel with ^{235}U to 89.97%. They are arranged in ten concentric rings distributed around a central control rod guide tube at a pitch distance of 10.95 mm. The fuel elements cage consists of two grid plates, four tie rods and a guide tube for the control rod. The two grid plates and four tie rods are connected by screws. Each gird contains 350 positions for fuel elements (347) and dummy aluminum elements (3). The core is under-moderated with an H/U atom ratio of 201 and cooled by natural convection using light water [1], [2].

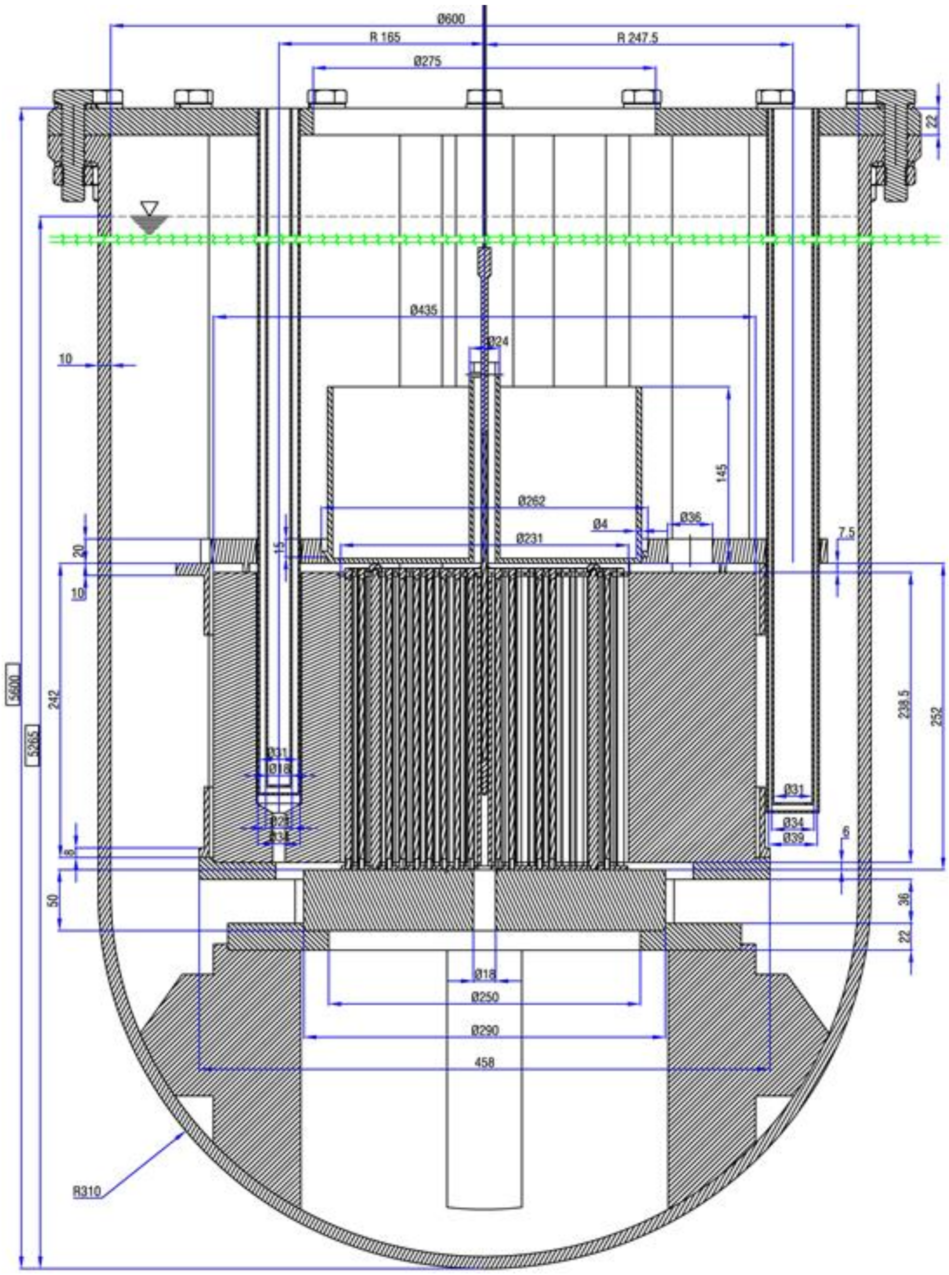

Figure 2: Frontal cross - section of the MNSR reactor by using AutoCAD 3-D program with demission's in mm.

The MNSR reactor has ten irradiation sites, five insides and five outside the annulus beryllium reflector as shown in Figure 2. 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 main specifications of the MNSR reactor are summarized in Table 1. The MNSR reactor is designed for neutron activation analysis, production of short-lived radioisotopes, and training of nuclear engineers, nuclear physicists and radiochemists [1], [2].

Table 1: Main Properties of the MNSR reactor.

	Description
	tank-in-pool
Rated thermal power	~ 30 kW
Fuel	UAl dispersed in Al base material
Parameter	89.97 %
Reactor type	Cylinder
Core diameter	23 cm
Core height	23 cm
Fuel element shape	Thin rod
Fuel element number in the core	~347
Total U 235 loading in the core	< 1 kg
Reactor continuous operating time at rated power	≥ 2.5 hours
Refuel period	More than ten years
Burnup	~1%
Temperature coefficient	~ −0.10 mk/c^o (Average)
Control rod(Cd)	One in the center of the core
Total number of irradiation sites	10
Number of inner irradiation sites	5 (small)
Thermal neutron flux in inner irradiation sites	$1\text{x}10^{12}$ n/cm^2.s – at rated power
Number of outer irradiation sites	5 (large)
Thermal neutron flux in outer irradiation sites	~$5\text{x}10^{11}$ n/cm^2.s – at rated power
Reactor cooling mode	Natural convection

1.2 The MNSR reactor fuel element

U-Al alloy has been used as fuel meat in many low-power research reactors, high flux engineering test-reactors and material testing reactors all over the world. The diameter of the fuel rod meat is 4.3 mm. The total length of the element is 248 mm and the active length is 230 mm. The clearance between pellet and end plug is 0.5 mm which allows for the thermal expansion of the pellets. There are 347 fuel elements in the core. Table 2 contains a list of the characteristics of the fuel element using in the MNSR reactor [1], [2].

1.3 Composition of U-Al alloy in the fuel meat

The meat weight of U in the alloy is 27.63%. The weight of U- 235 in the fuel element is 2.9 g. The fuel enrichment is 89.87%. The U-Al alloy of uranium 27.63 wt % is mainly UAl_4 + Al at room temperature. Therefore, the U-Al alloy is dispersion of UAl_4 in Al which possess irradiation resistance stability. The properties of U-Al alloy fuel are given in Table 3 [1].

Table 2: Characteristics of the fuel element.

Item	Quantity
U-Al alloy	
Uranium	27.63 wt %
Enrichment	89.87 %
Density	3.456 g/cm^3
Diameter	4.3 mm
Length	230 mm
Uranium loading of each rod	2.9 g
Number of fuel rods loaded	347
Number of fuel rod position	350
Cladding material	Al alloy 303 -1
Cladding thickness	0.6 mm
Dummy element material	Al
Number of dummy elements	3

Table 3: properties of U-Al alloy fuel meat.

Porosity ε %	3
Wt. % Enrichment of uranium in alloy	89.87
Wt. % of uranium in alloy	27.63
Density of U-Al alloy (ρ_{U-Al})	8.0146 g/cm^3
Density of Al (ρ_{Al})	2.7

2. The main problem of this research

The main problem of this research is simulating the research MNSR reactor using the MCNP-4C2 code and determination the criticality parameters and the thermal neutron flux in the IIS and OIS.

3. The importance of this research

The importance of this research is using the Monte Carlo method (MCNP4-C2 code) in neutronic and criticality calculations in low – power research reactors, this helps in:

1. Increasing 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.)

4. Importance of the MCNP-4C2 code

The MCNP-4C2 code is a popular, versatile multipurpose Monte Carlo particle transport code used worldwide. It has the capability to model and treat different geometries in 3-D, and also simulate the transport behavior of different particles and using a continuous energy cross section treatment as opposed to a multi-group approach thereby eliminating the errors in formulating few group cross sections. Additionally, MCNP-4C2 has the ability to treat complex nuclear interaction processes [3]. Therefore, in this work, the MCNP-4C2 code was used to:

- simulate the low-power MNSR reactor.
- evaluate the critically parameters such as: effective multiplication factor (k_{eff}), reactivity (ρ_{ex}), Control Rod Worth (CRW) and Shutdown Margin (SM).
- evaluate the neutron flux in the Inner Irradiation Sites (IIS) and Outer Irradiation Sites (OIS).

5. Simulation the MNSR reactor by using the MCNP-4C2 code

The low power MNSR reactor was simulated by using MCNP-4C2 code with three-dimensional detail to reduce possible systematic errors due to inexact geometry simulation. Therefore, this model of the low power MNSR reactor represents in detail all components of the core with literally no physical approximation. In the simulation of the low power MNSR reactor, the nuclear data for the fissile and the non-fissile materials such as:

- the fuel meat, fuel clad, coolant and moderator water, control rod and the reflector were taken from the ENDF/B-VI.1 nuclear data library.
- the thermal particle scattering $S(\alpha,\beta)$ was applied to treat the thermal scattering in both graphite and hydrogen of the coolant and moderated water.

The 3-D Monte Carlo MCNP-4C2 plot of the MNSR reactor core configuration is shown in Figure 3 and Figure 4.

6. Calculation the criticality parameters by using MCNP-4C2 code.

The $3-D$ Monte Carlo MCNP-4C2 model of the low power MNSR reactor was used to estimate the nuclear criticality parameters such as: effective multiplication factor (k_{eff}), reactivity (ρ_{ex}), Control Rod Worth (CRW) and Shutdown Margin (SM). In particular, neutron transport simulations were made for a fresh fuel (zero burn-up).

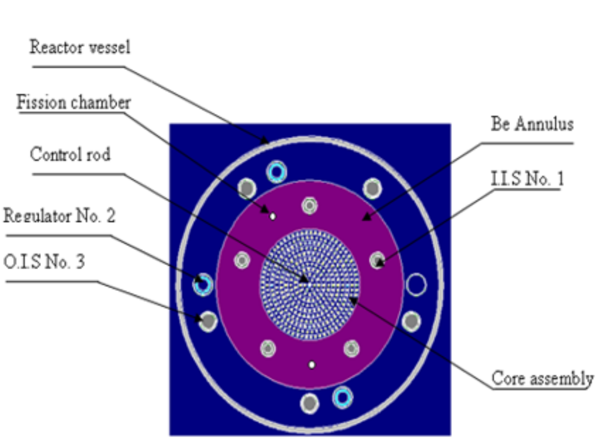

Figure 3: Cross section of the MNSR reactor by using MCNP-4C2 code

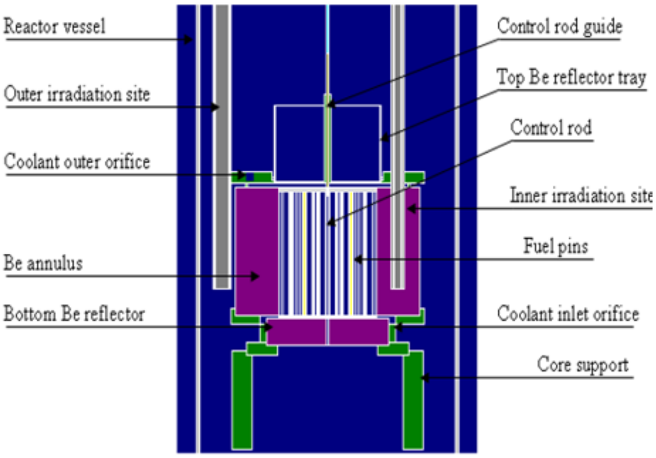

Figure 4: Vertical cross section of the MNSR reactor with the control rod totally inserted in the core by using MCNP-4C2 code

6.1 Calculation the multiplication effective factor (k_{eff})

The effective multiplication factor k_{eff} is defined as the ratio of the number of fission or fission neutrons in the generation divided by the number of fissions or fission neutrons in the preceding generation [3]. In the equation form,

$$k_{eff} = \frac{Number\ of\ fissions\ in\ one\ generation\ (i+1)}{Number\ of\ fissions\ in\ preceding\ generation\ (i)}$$

The formula of k_{eff} comes directly from the time-integrated Boltzmann transport equation without external source. k_{eff} can be written as follow [3]:

k_{eff}

This equation is derived from integro-different neutron transport equation with energy (E), flux (Φ), direction of neutron (Ω), reactor volume (V), neutron velocity (v), time (t) and neutron current flux (J), material or atom density (ρ_a) by meant of σ_c, σ_f and σ_m are microscopic cross section for capture (n, pn) fission and multiplicity (n, xn) respectively.

The effective multiplication factor k_{eff} is computed in MCNP−4C2 based on the calculation of three different estimators a collision−based k_{eff}^{C}, an absorption−based k_{eff}^{A} and a track length based k_{eff}^{TL} [3].

The effective multiplication factor k_{eff} was performed by the KCODE criticality source card [3] to determine the k_{eff} and corresponding excess core reactivity ρ_{ex} using all fuel elements as fission source points. In this analysis,

- 5×10^{6} neutron histories were used to run the MCNP−4C2 code,
- 10000 cycles with 100 passive cycles before the active cycles begin,
- initial value of the k_{eff} effective multiplication factor is 1,
- the cross − sections of all the isotopes formed in the reactor core were taken from the ENDF−VI.1 nuclear data library.

The calculated and experimental value of the effective multiplication factor k_{eff} is shown in Table 4.

6.2 Calculation the excess core reactivity (ρ_{ex}), Control Rod Worth (CRW) and Shutdown Margin (SM).

The excess core reactivity ρ_{ex}, the Control Rod Worth (CRW) and Shutdown Margin (SM) were calculated using the following equations [4], [5].

$$\rho_{ex} = (k_{eff} - 1)/\ k_{eff} \quad (1)$$

$$Control\ Rod\ Worth\ (CRW) = \frac{k_{eff}^{out} - k_{eff}^{in}}{k_{eff}^{out} \times k_{eff}^{in}} \quad (2)$$

Where:

k_{eff}^{out} – is the effective multiplication factor with control rod is fully withdrawn from reactor core and equal to k_{eff}.

k_{eff}^{in}− is the effective multiplication factor with control rod is fully inserted inside reactor core and equal to 0.99683 ± 0.00091, this value was calculated by using MCNP−4C2 code with same conditions mentioned in the item 4.1 used to estimate the k_{eff}.

$$SM = CRW - \rho_{ex} \quad (3)$$

Table 4: Calculated and experimental values of the effective multiplication factor (k_{eff}), excess core reactivity (ρ_{ex}), Control Rod Worth (CRW) and Shutdown Margin (SM).

Experimental and calculated value			Relatively error %
Parameter	Experimental	Calculated	
Effective multiplication factor (k_{eff})	(1.00396 ± 0.0001)	(1.00386 ± 0.00093)	0.001
Reactivity (ρ_{ex})	(3.94 ± 0.01) mk	(3.845 ± 0.003) mk	2.41
Control Rod Worth (CRW)	(−7.00 ± 0.01) mk	(−7.025 ± 0.013) mk	0.36
Shutdown Margin (SM)	(−3.06 ± 0.01) mk	(−3.180 ± 0.008) mk	3.92

7. Calculation the thermal neutron flux in the IIS and OIS by using MCNP−4C2 code.

To calculate the neutron flux in the IIS and OIS, the input file of the MNSR reactor was run by the MCNP−4C2 code using the F_5 tally and the FM cards [4], [5].

In the MCNP−4C2 code, tallies are normalized per source particle except in criticality calculations. 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 $\tilde{\nu} = 2.43$ neutron ($\tilde{\nu}$ − is the average number of neutrons released per fission, its value being listed in the MCNP−4C code output file), the reactor operating power (P = 30 kW) and the MCNP−4C2 flux tally normalized per source neutron [3]:

$$\varphi = \varphi_{MCNP} \times P \times \tilde{\nu} \left(\frac{neutrons}{seconds}\right) \times \left(\frac{1MeV}{1.6022\times10^{-13}Joules}\right) \times \left(\frac{fission}{200\ MeV}\right) \qquad (4)$$

The power in equation (4) is in the units of Joules/seconds, to give the normalized flux in the correct unit of neutrons/cm^2.s [4]. Therefore, to produce P watts of power, one needs $3.1203\text{x}10^{10}$P fissions per second. This produces $3.1203\text{x}10^{10}$x P x$\tilde{\nu}$ neutrons/s, which is thesource strength for this power level. The source strength (normalization) should be written in the tally on the FM card to calculate the thermal neutron flux in the IIS and OIS [4].

In addition, to calculate the thermal neutron flux in the IIS and OIS, the MCNP-4C2 code was run for $15x10^{11}$ particle histories (5×10^{7} particles and 30000 criticality cycles) and two thousand cycles were initially skipped before the actual runs.

The neutronics calculations were performed using three energy groups as:<0.625 eV for thermal neutrons, (0.625eV to 5.53keV) for epithermal neutrons and up to 20 MeV for fast neutrons [4], [5]. Calculations were also normalized to the steady-state power level of 30 kW. The validity of these calculations for the UAl-Al original fuel is verified by comparing them with the experimental values as shown in the Table 5.

Table 5. Calculated and experimental values of the thermal neutron flux in the Inner Irradiation Site (IIS) and Outer Irradiation Site (OIS).

Parameter	Thermal neutron flux n/cm^2.s		Relatively error %
	Experimental values	Calculated values	
Inner Irradiation Site (IIS)	$(1.000 \pm 0.02) \times10^{12}$	$(1.016 \pm 0.031) \times10^{12}$	1.6
Outer Irradiation Site (OIS)	$(5.000 \pm 0.02) \times10^{11}$	$(4.965 \pm 0.014) \times10^{11}$	0.7

8. Results and discussion

From Table 4, the Monte Carlo MCNP-4C2 calculations for the k_{eff} and excess core reactivity $\boldsymbol{\rho_{ex}}$ differ from the experimental results by about 0.001% and 2.41%, respectively. Similarly, the difference in the control rod worth and shutdown margin is 0.36% and 3.92%, respectively. The differences in values between the experimental and MCNP-4C2 calculations are good and accetaple.

From Table 5, the differences between the experimental values and MCNP-4C2 calculation for thermal neutron flux IIS and OIS is about 1.6% and 0.7%, respectevily. The good agreement between the experimental measurements and simulated results for the nuclear criticality k_{eff} , excess core reactivity $\boldsymbol{\rho_{ex}}$, control rod worth and thermal neutron flux confirm the accuracy of the 3-D Monte Carlo model of the MNSR reactor using MCNP-4C2 code.

Conclusion

The MCNP−4C2 code was used to simulate and calculate the criticality and the neutronic parameters of the MNSR reactor using a UAl−Al original fuel. The obtained results for the criticality and the neutronic parameters showed a good agreement with the reference values. This work provides the evidence that MNSR reactor model using MCNP−4C2 can be used for predict the negative effect on the criticality and neutronic parameters of the reactor in the event of an emergency accident in the reactor or any change in the reactor core.

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