Neutronic analysis for core conversion (HEU–LEU) of the low power research reactor using the MCNP4C code

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Abstract. Comparative studies for conversion of the fuel from HEU to LEU in the miniature neutron source reactor (MNSR) have been performed using the MCNP4C code. The HEU fuel (UAl$_4$-Al, 90% enriched with Al clad) and LEU (UO$_2$ 12.6% enriched with zircaloy-4 alloy clad) cores have been analyzed in this study. The existing HEU core of MNSR was analyzed to validate the neutronic model of reactor, while the LEU core was studied to prove the possibility of fuel conversion of the existing HEU core. The proposed LEU core contained the same number of fuel pins as the HEU core. All other structure materials and dimensions of HEU and LEU cores were the same except the increase in the radius of control rod material from 0.195 to 0.205 cm and keeping the outer diameter of the control rod unchanged in the LEU core. The effective multiplication factor (k_{eff}), excess reactivity (ρ_{ex}), control rod worth (CRW), shutdown margin (SDM), safety reactivity factor (SRF), delayed neutron fraction (β_{eff}) and the neutron fluxes in the irradiation tubes for the existing and the potential LEU fuel were investigated. The results showed that the safety parameters and the neutron fluxes in the irradiation tubes of the LEU fuels were in good agreements with the HEU results. Therefore, the LEU fuel was validated to be a suitable choice for fuel conversion of the MNSR in the future.

Key words: MNSR reactor • HEU fuel • LEU fuel • MCNP4C code • safety parameters

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Introduction

The miniature neutron source reactor (MNSR) is a tank-in-pool type research reactor [1]. It is one of the low power research reactors that uses highly enriched uranium as fuel, light water as moderator, and beryllium as reflector. Heat generated in the core is removed through natural convection and is transferred to the pool through the vessel walls. The core consists of 1 central control rod, 347 fuel rods, 4 tie rods, and 3 dummy rods. The diameter and the active height of the core are 23.0 cm. The beryllium reflector in the reactor can be divided into three sections: the side annulus surrounding the reactor core, the bottom plate, and the top beryllium shims. The MNSR reactor has 10 irradiation tubes: five are called 'inner' and are uniformly located inside the annular beryllium reflector as can be seen in Fig. 1. The other five are called 'outer' and surround the annulus reflector externally. The nominal reactor power is 30 kW [1].

Potential LEU fuels for the MNSRs

Efforts have been made for core conversion of MNSRs from the highly enriched uranium (HEU) to low-enriched uranium (LEU) fuel [2]. Recently,
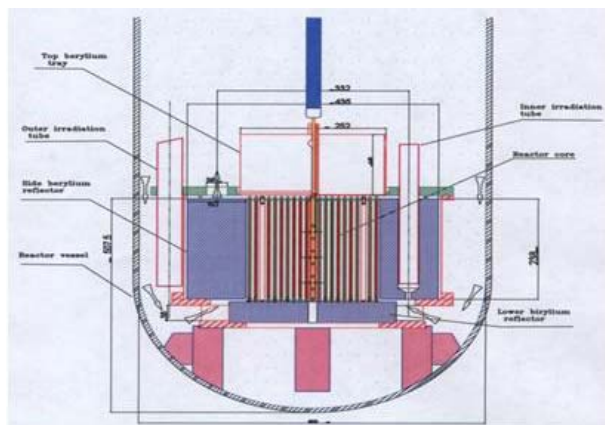
Fig. 1. Vertical cross section of the MNSR reactor.

a feasibility study has been performed for a generic MNSR to identify the potential LEU fuels as proposals for conversion of this reactor [3]. The HEU MNSRs use fuel pins containing UAl$_4$-Al alloy as fuel meat with 0.94 g/cm^3 uranium density and 90% ^{235}U enrichment. Table 1 shows the uranium density, meat outer diameter (OD), clad thickness, number of fuel pins, grams of ^{235}U in the fuel pin, and grams of ^{235}U in the core for the HEU and LEU cores.

Calculation of safety parameters of the Syrian MNSR and the neutron fluxes in the irradiation tubes using the MCNP4C code for the potential LEU fuel (UO$_2$, 12.6% enriched) and the existing HEU fuel (UAl$_4$-Al, 90% enriched) is presented in detail in the following paragraphs.

Methodology

The MCNP4C Monte Carlo code is a powerful and versatile tool for particle transport calculations. It can be used for transport calculations of neutrons, photons, and electrons. Transport calculations of neutrons in the reactor are required for reactor physicists to design the reactor core. The MCNP4C code can be used to calculate the effective multiplication factor, reaction rate, and flux distributions in the reactor core. It can be used to design any complex core geometry without any approximation. The MCNP4C code is provided in seven standard tallies [4]. All tallies are normalized to one starting particle. The effective multiplication factor is one of the most important properties of the reactor. The KCODE card in the MCNP4C code is usually used for criticality calculation in the reactor. Since the MCNP4C results are normalized to one source neutron, the result has to be properly scaled in order to get the absolute flux, reaction rate, fission density, etc.

A Monte Carlo simulation of the MNSR reactor was carried out previously using the MCNP4C code and continuous energy cross-section data from the
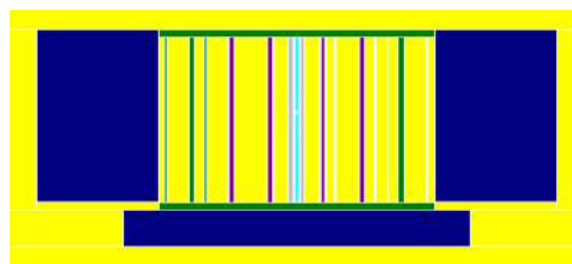

Fig. 2. Vertical cross section of MNSR reactor using the MCNP4C code.
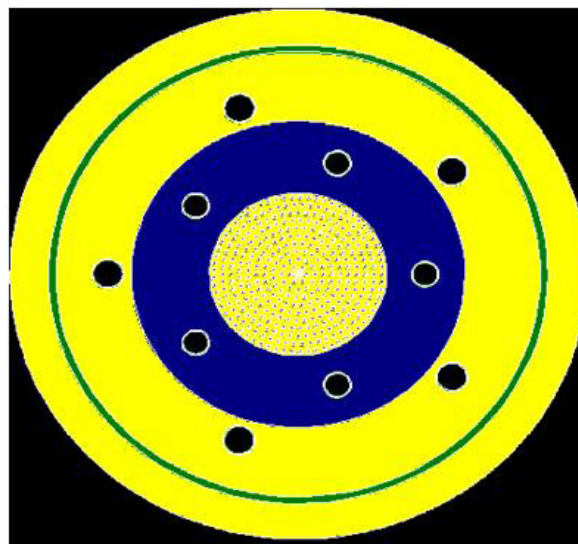

Fig. 3. Horizontal cross section of MNSR reactor using the MCNP4C code.

ENDF/B-VI library [5] and [6]. Our model was in three dimensions and it consisted of all the reactor components. In our 3D model, we used the LEU fuel with the zircaloy-4 alloy clad and increased the radius of control rod material from 0.195 to 0.205 cm keeping the outer diameter of the control rod unchanged. Figures 2 and 3 show the vertical and horizontal cross sections using the MCNP4C code implemented in our model. The calculation was conducted using the full continuous energy cross section available at the MCNP4C library at 20°C. This model was used in this study to calculate the effective multiplication factor (k_{eff}), excess reactivity (ρ_{ex}), control rod worth (CRW), shutdown margin (SDM), safety reactivity factor (SRF), and delayed neutron fraction (β_{eff}) using the KCODE card, which is usually used for criticality calculation in the reactor using the MCNP4C code.

The F5 tally of the MCNP4C code was used to calculate the axial thermal, epithermal, and fast neutron fluxes distributions in the irradiation tubes. The upper energy boundaries for the three energy groups were selected as follows: 0.625 eV for the thermal group, 0.5 MeV for the epithermal group, and 20 MeV for the fast group in the MNSR reactor.

Table 1. The physical characteristic of the HEU and LEU fuels

Fuel type	U density [g/cm^3]	OD of meat [mm]	Clad mat./thick. [mm]	No. of fuel pins	g of ^{235}U per pin	g of ^{235}U in core
UAl alloy (90%)	0.94	4.3	Al/0.6	347	2.83	981
UO$_2$ fuel (12.6%)	9.328	4.3	Zr-4/0.6	347	3.924	1362

The scaling factor in units of fission neutrons per unit time was calculated for the MNSR reactor and found to be $S = 2.224 \times 10^{15}$.

The effective (β_{eff}) in Syrian MNSR reactor was calculated by the MCNP4C transport code using the following equation [2]:

$$(1) \qquad \beta_{eff} = \frac{k_{eff} - k_p}{k_{eff} \cdot k_p}$$

where, k_{eff} is the effective multiplication factor taking both prompt and delayed neutrons into account that is acquired in the usually calculation mode of the MCNP4C code, using the KCODE card. In the KCODE mode, the mean values of both prompt and delayed neutrons are used in criticality calculations. To prevent the influence of the delayed neutrons, the TOTNU data card with entry NO was used to obtain the value of k_{eff} for prompt neutrons (k_p) only.

The reactor reactivity is calculated using the following equation:

$$(2) \qquad \rho = \frac{k_{eff} - 1}{k_{eff}}$$

It represents the fraction of neutrons created that are in excess of those required to keep the neutron population constant.

The reactor ρ_{ex} calculation is the reactivity of the core when the control rod is fully withdrawn from the reactor core. The reactor ρ_{ex} can be calculated using the following equation:

$$(3) \qquad \rho_{ex} = \frac{k_{eff}^{out} - 1}{k_{eff}^{out}}$$

where, k_{eff}^{out} is the reactor multiplication factor obtained when the control rod is fully withdrawn from the reactor core.

The SDM calculation is the reactivity of the core when the control rod is fully inserted in the reactor core. This is given by the following mathematical expression:

$$(4) \qquad SDM = \frac{k_{eff}^{in} - 1}{k_{eff}^{in}}$$

where, k_{eff}^{in} is the reactor multiplication factor obtained when control rod is fully inserted in the reactor core.

There is only one central control rod for the Syrian MNSR serving as a shim rod. The reasons for having one control rod only are the inherent safety feature of the reactor and the fact that the reactor exhibits a very strong self-regulating power capability. The control rod has a total length of 440 mm and its diameter is 5 mm. The length and the diameter of the control rod absorber are 256 mm and 3.9 mm, respectively. The thickness of the stainless steel cladding is 0.5 mm. The CRW can be calculated using the following equation:

$$(5) \qquad CRW = \frac{k_{eff}^{out} - k_{eff}^{in}}{k_{eff}^{out} \cdot k_{eff}^{in}}$$

The SRF is the ratio between CRW and the reactor ρ_{ex} of the core, which can be given using the following equation [7]:

$$(6) \qquad SRF = \frac{CRW}{\rho_{ex}} \geq 1.5.$$

Results and discussion

Table 2 shows the: k_{eff}^{out}, k_{eff}^{in}, ρ_{ex}, SDM, CRW, and SRF of the MNSR using the UAl$_4$-Al and UO$_2$ (12.6% enrichment with zircaloy-4 alloy clad) fuels for the same core dimensions keeping the diameter of fuel pins and the control rod unchanged. It is observed from this table that the SRF value for the UO$_2$ fuel type is less than 1.5. Therefore, this conversion would require a modification in the control rod dimension to maintain the safety margins ≥ 1.5 as in the current HEU core. Consequently, the thickness of the cladding of the control rod was reduced and the cadmium radius was increased from 0.195 to 0.205 cm. As can be seen from Table 2, good agreements are noticed between our calculated results and the results mentioned in references [1] and [8] with maximum relative deference less than 3%.

Table 3 shows the: k_{eff}^{out}, k_{eff}^{in}, ρ_{ex}, SDM, CRW, and SRF of the UAl$_4$-Al and UO$_2$ fuels after modification in dimensions of the control rod material and keeping the outer diameter of control rod unchanged.

Table 4 shows the calculated MCNP4C results for the β_{eff} for the reference HEU and proposed LEU MNSR core. As can be seen from this table, the β_{eff} value of the LEU UO$_2$ fuel is 7.5×10^{-3}. It is lower than the β_{eff} value of the HEU fuel, which is 7.54×10^{-3}. As can be seen from the results that appear in Tables 3–5, good agreements are noticed between our results and the results mentioned in the literatures [1, 9], and [8] with maximum relative deference less than 3%.

Figures 4–7 show the axial thermal, epithermal, and fast neutron flux distributions in the inner and outer irradiation tubes for the UAl$_4$-Al (90% enrich-

Table 2. Safety parameters of the MNSR facility using the HEU and LEU fuels for the same core dimensions

Fuel type	Reactor status			Safety parameters				
	k_{eff}^{out} Adjusted*	k_{eff}^{in} Adjusted*	ρ_{ex} [mk]	SDM [mk]	CRW [mk]	SRF [mk] (in this work)	SRF [mk] [1]	SFR [mk] [8]
UAl alloy (90%)	1.00397 ± 0.00007	0.99742 ± 0.00007	3.954 ± 0.069	–2.587 ± 0.070	6.541 ± 0.139	1.654 ± 0.045	1.67	1.580
UO$_2$ fuel (12.6%)	1.00392 ± 0.00007	0.99845 ± 0.00007	3.905 ± 0.069	–1.552 ± 0.070	5.457 ± 0.139	1.398 ± 0.043	–	1.357

* Adjusted: the reactor multiplication factor after adding three reactivity regulators to the reactor core.

Table 3. Safety parameters of the MNSR facility using the HEU and LEU fuels after modification in control rod material diameter keeping its outer diameter constant

Fuel type	Reactor status		Safety parameters			
	k_{eff}^{out} Adjusted*	k_{eff}^{in} Adjusted*	ρ_{ex} [mk]	SDM [mk]	CRW [mk]	SRF [mk]
UAl alloy (90%)	1.00397 ± 0.00007	0.99742 ± 0.00007	3.954 ± 0.069	–2.587 ± 0.070	6.541 ± 0.139	1.654 ± 0.045
UO₂ fuel (12.6%)	1.00392 ± 0.00007	0.99829 ± 0.00007	3.905 ± 0.069	–2.717 ± 0.111	6.622 ± 0.163	1.696 ± 0.052

* Adjusted: the reactor multiplication factor after adding three reactivity regulators to the reactor core.

Table 4. The delayed neutron fraction for the reference HEU and proposed LEU in the MNSR core

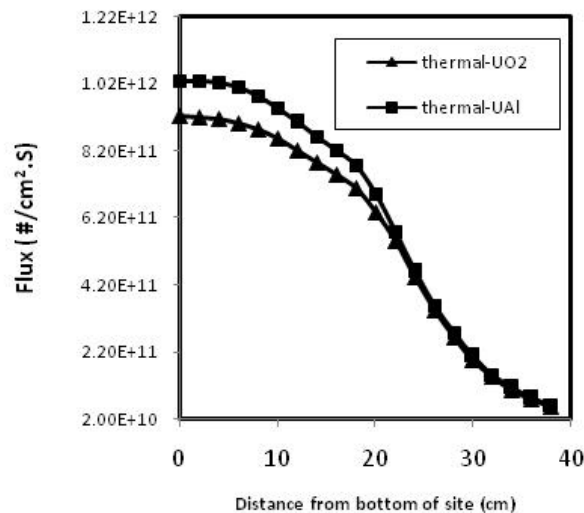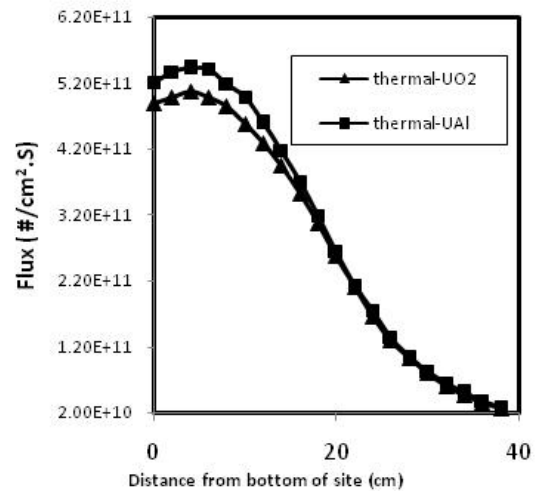| Fuel type	k_{eff} (total)	k_{eff} (delayed)	β_{eff} MCNP4C	β_{eff} [9]	β_{eff} [1]
UAl alloy (90%)	1.00510 ± 0.00007	0.99750 ± 0.00007	$7.54 \times 10^{-3} \pm 1.37 \times 10^{-4}$	7.32×10^{-3}	8.08×10^{-3}
UO₂ fuel (12.6%)	1.00508 ± 0.00007	0.99752 ± 0.00007	$7.5 \times 10^{-3} \pm 1.38 \times 10^{-4}$	–	–

Fig. 4. Comparison of axial thermal flux distributions for the HEU and LEU fuels in the inner irradiation tube of the MNSR reactor.

Fig. 6. Comparison of axial thermal flux distributions for the HEU and LEU fuels in the outer irradiation tube of the MNSR reactor.
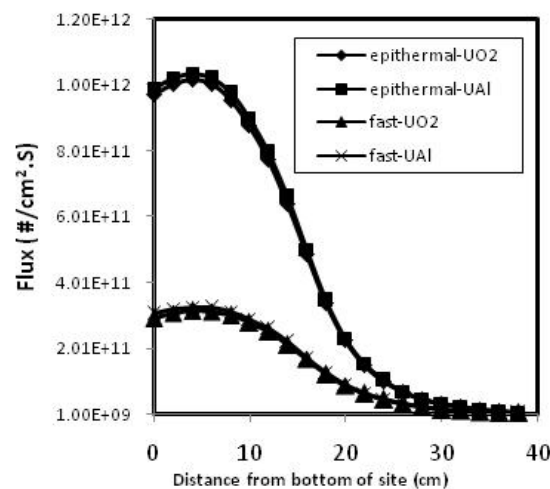
Fig. 5. Comparison of axial epithermal and fast flux distributions for the HEU and LEU fuels in the inner irradiation tube of the MNSR reactor.
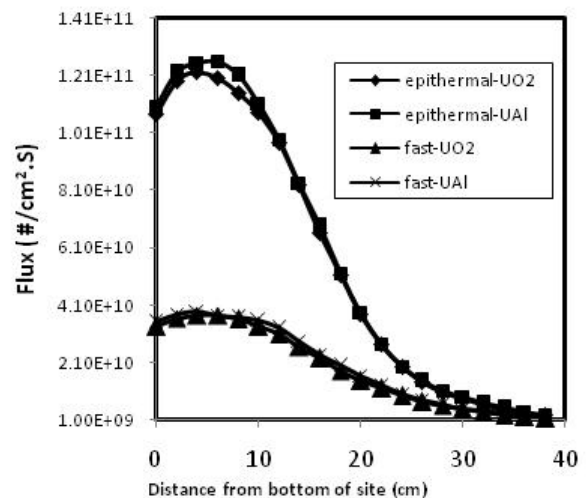
Fig. 7. Comparison of axial epithermal and fast flux distributions for the HEU and LEU fuels in outer irradiation tube of the MNSR reactor.

Table 5. The axial neutron fluxes in the bottom of the inner irradiation tubes for the reference HEU and proposed LEU fuels in the MNSR core

Fuel type	Fast (0.5–20 MeV) [n/cm²·s]	Epithermal (0.625 eV – 0.5 MeV) [n/cm²·s]	Thermal (0–0.625 eV) [n/cm²·s]	Thermal [1]	Thermal [8]
UAl alloy (90%)	3.07E+11	9.92E+11	1.03E+12	1.00E+12	1.02E+12
UO₂ fuel (12.6%)	2.92E+11	9.75E+11	9.23E+11	–	9.36E+11

Table 6. The axial neutron fluxes in the bottom of the outer irradiation tubes for the reference HEU and proposed LEU fuels in the MNSR core

Fuel type	Fast (0.5–20 MeV) [n/cm²·s]	Epithermal (0.625 eV – 0.5 MeV) [n/cm²·s]	Thermal (0–0.625 eV) [n/cm²·s]	Thermal [1]	Thermal [8]
UAl alloy (90%)	3.51E+10	1.10E+11	5.21E+11	5.00E+11	5.30E+11
UO₂ fuel (12.6%)	3.34E+10	1.07E+11	4.90E+11	–	4.98E+11

ment) and UO_2 (12.6% enrichment) fuels. The value decreases vs. the height of the irradiation tube starting from the bottom of the tube. Comparison of axial thermal flux distribution in the outer–inner irradiation tubes, both HEU and LEU fuels exhibit similar axial flux patterns except that flux value is slightly lower in the LEU fuel due to spectrum hardening.

Tables 5 and 6 show the values of the axial thermal, epithermal, and fast fluxes in the inner–outer irradiation tubes for the reference HEU and proposed LEU fuels in the MNSR core.

Conclusion

The MCNP4C code was used in this paper to calculate the safety parameters: the k_{eff}, ρ_{ex}, CRW, SDM, SRF and β_{eff} for the potential LEU (UO_2 12.6% enriched with zircaloy-4 alloy clad) and the existing HEU (UAl_4-Al, 90% enriched with Al clad) fuels in the MNSR using the KCODE card, which is usually used for criticality calculation.

The results showed that the MNSR fuel can be converted from the HEU to LEU using the same dimensions of the MNSR core except the increase in the radius of control rod material from 0.195 to 0.205 cm and keeping the outer diameter of the control rod unchanged. Good agreements between the thermal neutron flux distributions in the inner and outer irradiation tubes for the HEU and LEU were noticed with maximum relative deference less than 10%. Therefore, the LEU UO_2 with zircaloy-4 alloy clad fuel can be a suitable choice for the fuel conversion project for the MNSR in the future.

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