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Determination of the LEU core safety parameters of the MNSR reactor using the MCNP4C code

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ABSTRACT

Comparative studies for the conversion of the fuel from HEU to LEU in the Miniature Neutron Source Reactor (MNSR) have been performed using the MCNP4C code. The safety parameters like: 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}) for the existing HEU fuel (UAl_4–Al, 90% enriched) and the potential LEU fuels (U_3Si_2–Al, U_3Si–Al and the U9Mo–Al, 19.75% enriched) were investigated in this paper for the MNSR using the MCNP4C code. The results showed for the existing HEU fuel and the potential LEU fuels that the control rod shutdown margins were: 2.587, 2.295, 2.325 and 2.165 mk. The control rod worths were: 6.54, 6.16, 6.09 and 6.05 mk. The safety reactivity factors were: 1.648, 1.588, 1.611 and 1.551. The delayed neutron fractions were calculated to be: 7.54×10^{-3}, 7.48×10^{-3}, 7.29×10^{-3} and 7.57×10^{-3} for the UAl_4–Al, U_3Si_2–Al, U_3Si–Al and the U9Mo–Al fuels respectively. Finally, the safety parameters of the LEU fuels are in good agreements with HEU results and all types of the LEU fuels can be suitable candidates for the fuel conversion in the future.

1. Introduction

The Miniature Neutron Source Reactor (MNSR) is a tank in pool type research reactor (CIAE, 1993). It is one of the low power research reactors which use 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 one central control rod, 347 fuel rods, four tie rods and three 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 ten irradiation sites: five are called 'inner' and uniformly located inside the annulus beryllium reflector as can be seen in Fig. 1. The other five are called 'outer' and surround the annulus reflector externally. The reactor nominal power is 30 kW (CIAE, 1993).

2. Potential LEU fuels for the MNSRs

Efforts have been made for core conversion of MNSRs from the HEU to LEU fuel (Khamis and Khattab, 1999). Recently, a feasibility study has been performed for a generic MNSR to identify the potential LEU fuels as candidates for conversion of this reactor (Matos and Lell, 2005). The HEU MNSR use fuel pins containing UAl_4–Al alloy fuel meat with a uranium density of 0.94 g/cm^3 and 90% enrichment. The study indicated that the acceptable LEU fuels should have uranium densities greater than 5 g/cm^3 for the same core design. The list of acceptable potential LEU fuels is given in Table 1. Among of these are the: U9Mo–Al, U_3Si–Al and U_3Si_2–Al fuels being the most promising candidates and have been selected in this work for calculating the safety parameters of the MNSRs.

Calculation of safety parameters of the Syrian MNSR using the MCNP4C code for the potential LEU fuels (U–9Mo, U_3Si–Al and U_3Si_2–Al, 19.75% enriched) and with the existing HEU fuel (UAl_4–Al, 90% enriched) can be seen precisely in the following paragraphs.

3. 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 and power distributions. It can be used to design any complex core geometry without any approximation. The MCNP4C code is provided with seven standard tallies (Briesmeister, 2000). 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

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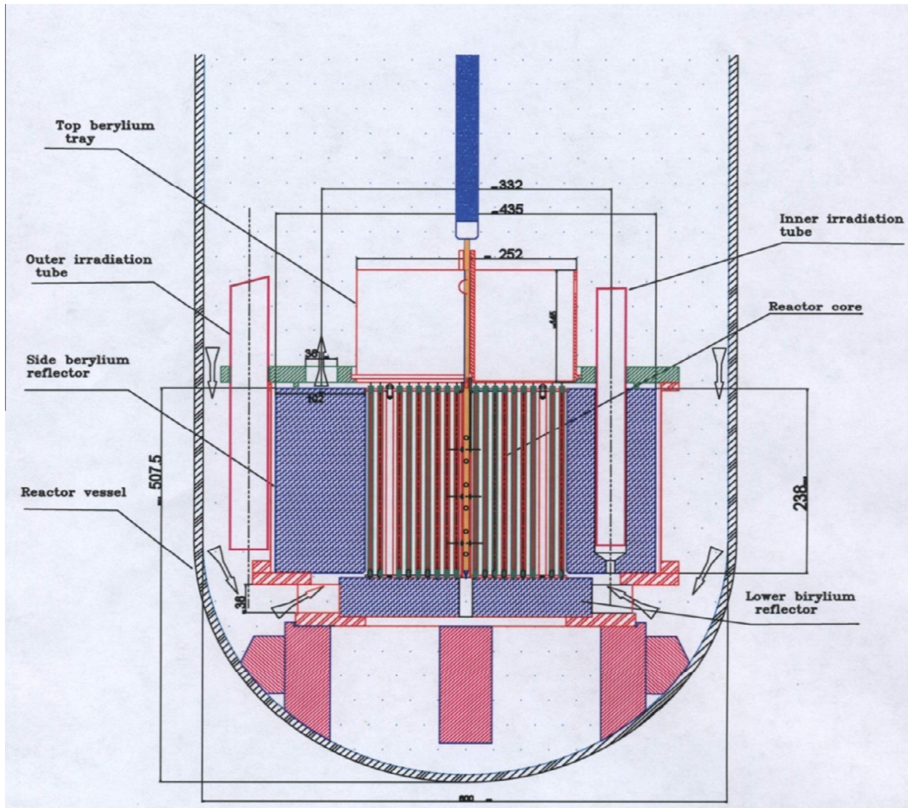

Fig. 1. Vertical cross section of the MNSR reactor.

Table 1
The physical characteristic of the acceptable potential LEU fuels.

Fuel type	U dens. (g/cm^3)	OD meat (mm)	Clad mat./thick. (mm)	No. of pins	g ^{235}U per pin	g ^{235}U in core
UAl Alloy (90%)	0.94	4.3	Al/0.6	347	2.83	981
U_3Si_2–Al (19.75%)	4.29	4.74	Al/0.38	347	3.44	1194
U_3Si–Al (19.75%)	5.37	4.3	Al/0.6	347	3.54	1229
U9Mo–Al (19.75%)	5.76	4.3	Al/0.6	347	3.8	1319

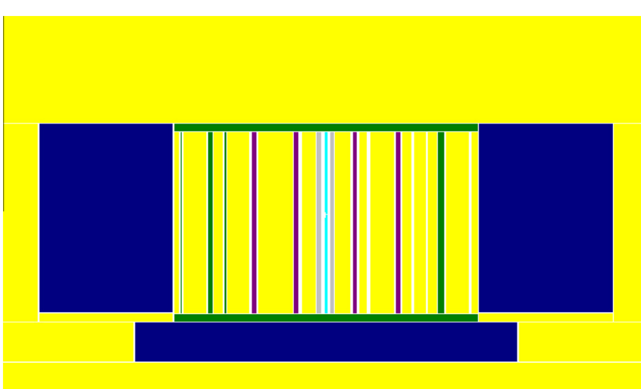

Fig. 2. Vertical cross section of MNSR reactor using the MCNP4C code.

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 ENDF/B-VI library (Khattab and Sulieman, 2009). This model was in three dimensions and it consisted of all the reactor components which were: 347 fuel rods, four tie rods, three dummy rods, control rod and its guide tube, reactor vessel, top and bottom aluminum alloy grids, annular beryllium reflector, bottom beryllium reflector, five inner and five outer irradiation sites and the reactor pool (Khattab and Sulieman, 2010). All the fuel elements, beryllium reflector and the irradiation sites were represented as cylinders of appropriate materials and dimensions positioned at exact location in the model (Khattab and Sulieman, 2011). Figs. 2 and 3 show the components of the reactor core implemented in our model. The calculation was conducted using the full continuous energy cross section available at the MCNP-4C library at 20 °C. This model was used in this paper to calculate the effective multiplication factor, excess reactivity, control rod worth, shutdown margin, safety reactivity factor-and delayed neutron fraction using the KCODE card which is usually used for criticality calculation in the reactor using the MCNP4C code.

3.1. β_{eff} calculation

The effective delayed neutron fraction β_{eff} in Syrian MNSR reactor is calculated by the MCNP4C transport code using the following equation (Bretscher, 1997):