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Numerical Modeling and Calculation of the Fuel Cycle for the IRT-Sofia Research Reactor

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Abstract. The IRT-Sofia research reactor at the Nuclear Scientific Educational and Experimental Centre of the Institute for Nuclear Research and Nuclear Energy is under reconstruction in order to achieve the contemporary national and international standards for safety and reliability. The reactor is foreseen to operate with low enriched nuclear fuel (less than 20% U-235) according to the project for reconstruction.

A model of the IRT reactor core is presented in this paper to determine the level of fuel burn-up using the MCNPX code, which performs neutron-physical calculations by the Monte Carlo method without applying approximations either for description of the core composition and geometry or for the neutron cross-sections energy dependence. The reactor core is modeled taking into account the control rods positions and the actual rounded profile of the IRT-4M type fuel assemblies and, with this input data, the nuclear fuel consumption of the IRT-Sofia research reactor is determined.

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1 Introduction

The first nuclear facility in Bulgaria – the IRT-2000 research reactor, was commissioned in September 1961. The reactor was operated at nominal power of 2000 kW with thermal neutrons. It was water moderated and water cooled heterogeneous, pool type of research reactor that was built under a standard design developed in the former Soviet Union. Low enriched nuclear fuel assemblies of the EK-10 type (with enrichment 10% U-235) were used in a period of 28 years of accident-free operation and in the last four years (from 1986 to 1989) the reactor core was mixed with high enriched nuclear fuel assemblies of the C-36 type (enrichment 36% U-235). In 2001, the Council of Ministers of the Republic of Bulgaria issued Resolution No 552 for modernization of the IRT-2000 research reactor and its reconstruction into a civil-type reactor with a rated power of 200 kW by improving the physical protection and safety in order to meet the strengthened international requirements. The feasibility project for the nuclear facility refurbishment has been developed in due time [1] and IRT-Sofia was adopted as the research reactor designation.

Neutron-physical calculations are recognized as an indispensable tool to justify the parameters of nuclear research reactors and their biological shielding. Fuel cost could be an important issue and the use of state-of-the-art software, consistent with the characteristics of a particular nuclear facility, enables fast and reliable performance of important tasks, such as conducting variant analysis for core loading selection, determining the excess reactivity, optimizing the fuel consumption, specifying the operational resource of the fuel assemblies, etc.

In the frame of the **R**educed **E**nrichment for **R**esearch and **T**est **R**eactors (RERTR) international program, fuel cycle calculations were performed using the computer code REBUS-PC [2]. The results for the IRT-Sofia fuel cycle lead to the conclusion that the initial core configuration of 16 fresh IRT-4M fuel assemblies (four 8-tube and twelve 6-tube) can be operated continuously for about four calendar years.

The aim of this study was to improve the modeling of the core and further specify the nuclear fuel operational resource of the IRT-Sofia research reactor. The calculations of the resource of the enhanced core model have been performed using the Monte Carlo MCNPX code [3], which allows for the material burn-up feature and delayed particle production.

2 IRT-Sofia Core Description

According to the project for reconstruction [1], the reactor IRT-Sofia is foreseen to operate with low enriched uranium (LEU) fuel assemblies of the IRT-4M type (enrichment $19.7 \pm 0.3\%$; maximum 19.95% U-235). This nuclear fuel is a tubular type of fuel assembly (FA) and as shown schematically in Figure 1 it consists of six or eight coaxial tube-shaped fuel elements (FE) with square cross section and rounded corners. Figure 2 shows an expanded view of the eight-tube assembly. The fuel meat is enriched UO_2 – Al dispersion fuel. The basic

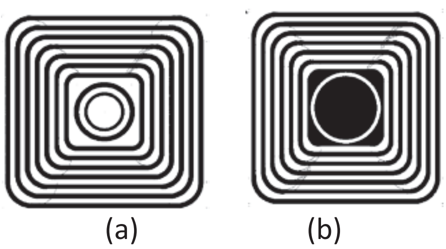

Figure 1. Transverse cross section view of IRT-4M tubular fuel assemblies: (a) eight-tube, (b) six tube.

Table 1. Basic technical characteristics of the IRT-4M type FA

IRT-4	6-tube	8-tube
Total height, mm	882	882
FE active length, mm	600	600
FA cross section, mm	69.6×69.6	69.6×69.6
Head end cross section, mm	71.5	71.5
FE thickness, mm	1.6	1.6
Fuel meat thickness, mm	0.7	0.7
FE cladding thickness, mm	min 0.3	min 0.3
Nominal FE clad thickness, mm	0.45	0.45
Coolant passage hole width, mm	min 1.35	min 1.35
Coolant pressure at FE inlet,	min 0.156	min 0.156
Maximum allowable temperature on FE surface, $^\circ$C	102	102
Volume fraction of water in FA	0.517	0.581
Mass of U^{235}, g	263.8	300.0
Total weight, kg	5.2	6.0
Clad material	Al alloy	Al alloy
Fuel material	UO_2 – Al	UO_2 – Al

technical characteristics of IRT-4M FA are presented in Table 1. In the central hole of a 6-tube FA either a control rod or a vertical experimental channel (for isotope production or irradiation of materials to modify their properties) can be positioned.

The fuel loading of the core of IRT-Sofia was selected in a close cooperation between the Institute for Nuclear Research and Nuclear Energy, the Argonne National Laboratory (USA), and the Kurchatov Institute in Moscow (Russian Federation). This cooperation started in 2003 when Bulgaria joined the RERTR international program. As usually, the calculations related to core conversion from highly enriched fuel to a low enriched one was done by preserving the structural integrity and design of the core base. Thus, the core selection was based on the assumption that the reactor grid was the typical one for all IRT research reactors in terms of construction design, dimensions and material composition. This provides the possibility to easily compare the new core parameters with previous ones from the reactor operating history. The first results from the neutron-physical calculations of the selected core, which has a total of 54 cells, are given in [4] and more comprehensive data on the IRT-Sofia basic parameters are presented in [5].

The reactor grid has a pitch of 71.5 mm and can fix 54 fuel and beryllium reflector assemblies. The safety and control system consists of 3 safety rods, 5 shim rods and 1 automatic regulating rod. The initially loaded core with fresh LEU fuel contains twelve 6-tube and four 8-tube IRT-4M fuel assemblies. The core layout is shown in Figure 2. The positions of the fuel assemblies are asymmetrical because the reactor is intended for research as well as for medical purposes (a

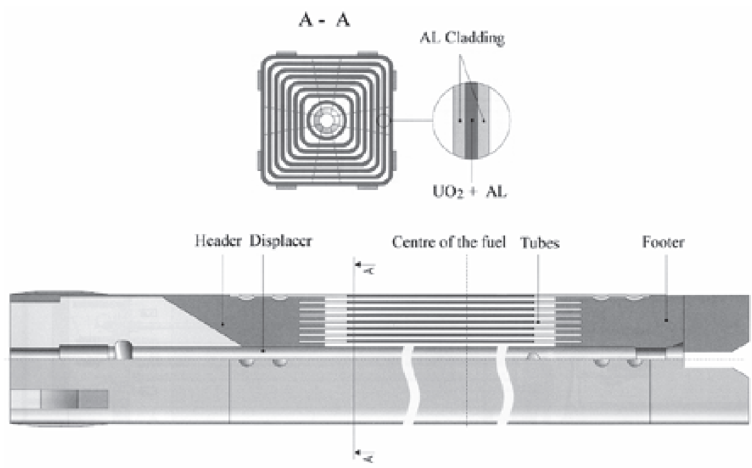

Figure 2. IRT-4M eight-tube assembly.

horizontal channel for boron neutron capture therapy (BNCT, No 1) is planned to be installed in the place of the thermal column).

The cells with diagonal-hatched lines in Figure 2 represent beryllium reflector assemblies; the F6 assembly has a hole for the automatic control rod, while vertical experimental channels with diameter of 48 mm are placed in designated holes of the assemblies in cells D3 and D4. The cells of columns H and I are filled with water reflector assemblies that have air cavity at the level of the six horizontal experimental channels (2 to 7) for better transfer of neutron fluxes. Cells A2-A5 are filled with four 8-tube fuel assemblies and there are four 6-tube fuel assemblies in cells C3, C4, E2 and E5; five shim rods (marked by a solid black circle) and three safety rods are placed in the rest eight 6-tube fuel assemblies (cells B2-B5 and F2-F5).

3 Numerical Models for the IRT-Sofia Core

In the earlier stage of determining the parameters of IRT-Sofia using LEU fuel of the IRT-4M type [4,5], the lifetime of the fuel assemblies in the selected initial core was calculated on the basis of a model, which was prepared with accounting for the capabilities of an older version of the diffusion theory code REBUS-PC [2]. This software package is developed by the Argonne National Laboratory (ANL, USA) and is designed for fuel cycle calculations of both homogeneous and heterogeneous reactors. The neutron cross sections for use in REBUS-PC are generated using the WIMS-ANL code [6]. WIMS-ANL uses a 69-energy group library based on ENDF-B/VI data collapsed to seven broad energy groups for use in REBUS-PC. Neutron-physical calculations with REBUS-PC were performed using these seven broad-group microscopic cross sections in the multi-group diffusion theory module DIF3D. The output from the calculations for each fuel cycle stage, predefined in effective days, are the effective neutron multiplication factor k_{eff} and the amount of generated fission products, as well as the residual amount of the initial fuel load.

In earlier core modeling [4,5], in order to simplify the input file, the fuel elements were modeled in REBUS-PC with square corners in slab geometry. In addition, the control rods were not included in the model of initial core. The duration of the first operational cycle (without loading additional fresh fuel) was assessed this way as four years of continuous operation at power level of 200 kW.

For the purpose of this study it was prepared a similar model of the IRT-Sofia core, presented schematically in Figure 3, which accounts for the capabilities of the MCNPX code [3]. This code is developed by the Los Alamos National Laboratory (LANL, USA) and lately is made available on request.

In addition to MCNP, which is a widespread general-purpose program for transport calculations by the Monte Carlo method of almost all particles in their almost entire energy range, the code MCNPX also features depletion/burn-up/transmutation capability when performing criticality calculations. It is able to provide neutron flux distributions in space, energy, and time, and to track isotopic changes in fuel and neutron absorbers with time. The algorithm that accounts for fissile isotopes burn-up is a consistent process starting with steady state calculations of the neutron and gamma flux parameters (identical to the calculations performed by MCNP). The calculated neutron fluxes represent input data for the CINDER90 module, which is used to calculate the fuel burn-up, and the corresponding isotopic densities of fission products are obtained as a result.

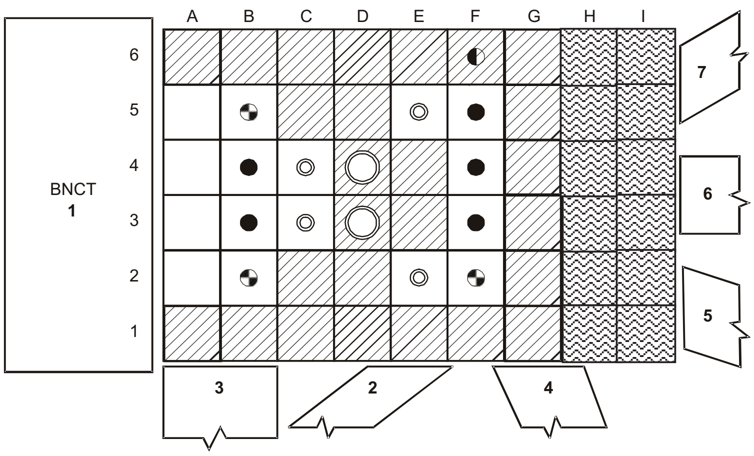

Figure 3. Initial core loading scheme of IRT-Sofia.

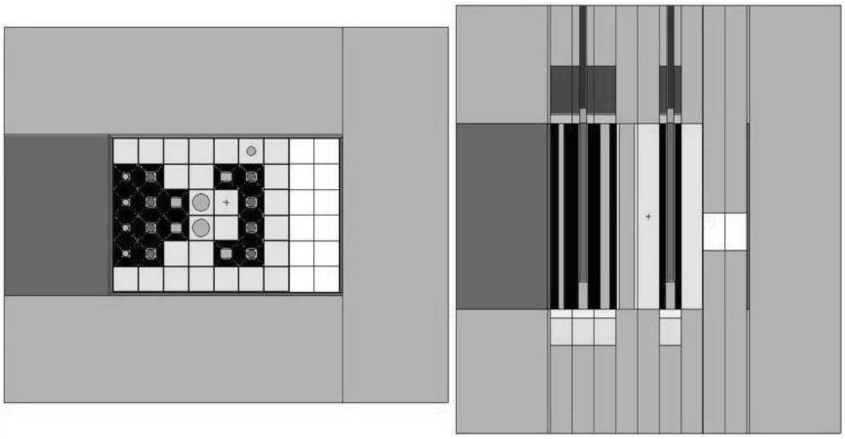

Figure 4. Visualization of the IRT-Sofia core model in MCNPX.

Geometric objects modeling in MCNPX is relatively easy: the code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. Thus, the fuel elements in the present model were defined considering their round corners. Another difference from the model in REBUS-PC is that the reactor control rods were added in the MCNPX model. In order to get a correct assessment of the core operational lifetime, the control rods were placed in their upper positions, leaving only their aluminum cylindrical followers in the IRT-Sofia core (Figure 4, on the right).

The results from the MCNPX calculations for k_{eff} are presented in Figure 5 (upper panel), where the calculated with REBUS-PC values are also shown for comparison. As shown on the ordinate, the initial value for k_{eff} (corresponding to a steady-state criticality calculation of a cold core, loaded only with fresh fuel) calculated by MCNPX is 1.06451, while the same result from REBUS-PC is 1.05607. The lower panel of Figure 5 represents the corresponding decrease of excess reactivity $\rho = (k_{\text{eff}} - 1)/k_{\text{eff}}$ derived from the calculations.

It can be summarized that the values of the effective neutron multiplication factor calculated with MCNPX are slightly greater than the values obtained by REBUS-PC. This difference varies from 0.6% to 0.8% and with a view of the assessment for initial core lifetime the results from REBUS-PC are more conservative. Furthermore, such a difference is not to be neglected, because it is comparable to one of the most important safety criteria for every research reactor – the shutdown margin, which for the case of IRT-200 is defined as at least 1% core subcriticality ($\rho = -1\%$, meaning that k_{eff} is equal to 0.99) when all safety rods and the auto regulating rod are fully withdrawn, and all shim rods are fully inserted [5]. Accordingly, the present results for k_{eff} indicate an extended

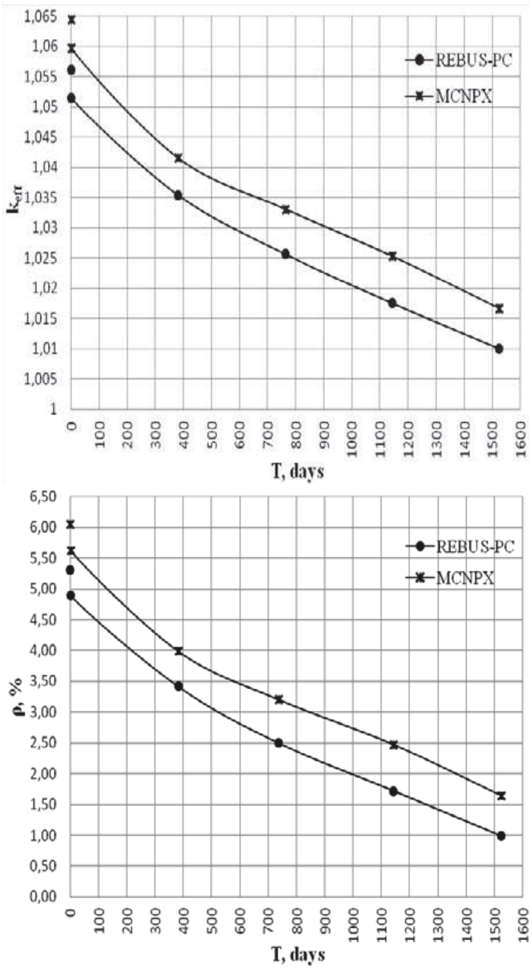

Figure 5. Changes with time in the effective neutron multiplication factor $k_{\rm eff}$ (left) and the excess reactivity (right) as a result from the calculations with REBUS-PC and MC-NPX.

core lifetime as compared to the limit of about four years of continuous core operation at 200 kW as evaluated with REBUS-PC. In addition, at least three times longer operation might be expected if a realistic duty factor (approximately 0.3 that corresponds to operation of twelve hours, five days a week and forty-one weeks per year) were taken into account.

The calculations with MCNPX were performed for continuous reactor operation at rated power level of 200 kW for 5 stages of the fuel cycle – the first is one day long and the rest four stages are with equal length of 381 days. This was intentional in order to reproduce the fuel cycle stages used for the calculations with REBUS-PC. But there is one important remark that has to be noted – REBUS-PC can divide the fuel cycle stages into subintervals in order to get accurate results for the fuel burn-up. Such a possibility is present in MCNPX as well, but in order to achieve the preset sufficiently low level of uncertainty of the results for the available computer system (standard dual-core workstation) it takes significantly long time to compute each stage of the calculations. Moreover, although the reactor is operated at a constant power, the duration of the stages cannot be sharply increased in order to get correct results for the fuel burn-up with MCNPX. The use of a multiprocessor workstation running under UNIX-based operating system is foreseen as a future activity, which is expected to reduce the time for MCNPX calculations. This will contribute to the full and in-depth feasibility study of the developed model.

4 Conclusion

A numerical model of the IRT-Sofia research reactor core was developed accounting for the capabilities of the MCNPX computer code for fuel cycle calculations. The model preserves the basic features of the earlier core model, used for preceding fuel cycle calculations using the REBUS-PC computer code, but takes into account additional details of the core composition and actual geometry of the IRT-4M fuel assemblies. The results for the effective neutron multiplication factor k_{eff} obtained with MCNPX although with little but steadily are higher than the values calculated with REBUS-PC providing this way the possibility to improve the assessment of the IRT-Sofia reactivity balance and operating reactivity margins.

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