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Research Article

Neutron background of composite low-enriched uranium fuel of the IVG.1M research reactor*

Ruslan A. Irkimbekov[1], Aleksandr D. Vurim[1], Sergey V. Bedenko[2], Artur S. Surayev[1], Galina A. Vityuk[1]

1 *Institute of Atomic Energy of the Republican State Enterprise National Nuclear Center of the Republic of Kazakhstan (NNC RK), 10 Beybit Atom Str., 071100 Kurchatov, Kazakhstan*

2 *National Research Tomsk Polytechnic University (TPU), 2 Lenin Ave., 634050 Tomsk, Russia*

Corresponding author: Sergey V. Bedenko (bedenko@tpu.ru)

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Abstract

IVG.1M is a research pressurized water reactor designed to use high-enriched fuel. As part of the core conversion program, the reactor will be switched to a new low-enriched composite uranium fuel. Further operation of the reactor is determined by the availability of fresh fuel to replace the core after the next campaign and the possibility of ensuring safe storage of irradiated spent nuclear fuel (SNF) unloaded from the core. The SNF storage conditions are assessed in terms of ensuring nuclear and radiation safety.

Radiation safety of the research reactor fuel storage is achieved, first of all, by solving problems of protection against γ-radiation, while neutron radiation, as a rule, is not considered due to its significantly lower intensity compared to γ-radiation. As for the new low-enriched fuel of the IVG.1M reactor, which is characterized by a set of elements with low and medium atomic masses, on which the (α, n) reaction is possible, the assessment of the neutron component is a necessary procedure to ensure safe fuel storage.

The authors of the article propose a procedure for calculating the neutron component of the radiation characteristics of fresh and irradiated composite fuel of the IVG.1M reactor, and also estimate the (α, n)-component. The results of the research will be useful in selecting SNF storage and transportation technologies as well as in providing scientific justification for the possibility of using neutron radiation to control burnup.

The research was carried out using verified computational codes MCNP5 and Sources-4C, high-precision experimental EXFOR and computational ENDSF data, as well as evaluated nuclear data libraries.

Keywords

The IVG.1M research reactor, low-enriched cermet-based uranium fuel, dosimetry, (α, n)-reaction

Introduction

In 2010, within the framework of the Kazakhstan-USA cooperation under the auspices of the IAEA, the IVG.1M reactor was included in the program for converting research reactors cores to low-enriched uranium fuel. In February 2021, fresh low-enriched uranium fuel was delivered to the National Nuclear Center of the Republic of Kazakhstan (NNC RK) (https://www.nnc.kz) from the Research and Production Association “LUCH” (Russia) (http://sialuch.com).

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It should be noted that earlier specialists from the NNC RK, together with colleagues from the Argonne National Laboratory (USA) (https://www.anl.gov) and the Russian Association "LUCH", checked the performance of this fuel and confirmed its suitability for conversion purposes.

After the conversion to low-enriched uranium is completed, the IVG.1M reactor will continue to operate; at the same time, the duration of this operation will be determined by the availability of fresh fuel to replace the core after the next campaign and the possibility of ensuring the safe storage of spent nuclear fuel unloaded from the core. The SNF storage conditions are assessed in terms of ensuring nuclear and radiation safety.

Radiation safety of research reactor fuel storage is achieved, first of all, by solving problems of protection against γ-radiation, while neutron radiation, as a rule, is not considered due to its significantly lower intensity compared to γ-radiation. As for the new low-enriched fuel of the IVG.1M reactor, which is characterized by a set of elements with low and medium atomic mass, on which the (α, n) reaction is possible, the assessment of the neutron component is a necessary procedure to ensure safe fuel storage.

Therefore, the goal of this research is to estimate the neutron radiation level of fresh and irradiated fuel of the IVG.1M reactor and to develop recommendations for safe long-term SNF storage.

To achieve the goal of the work

- a full-scale computational 3D model of the reactor was created;
- neutron studies of the core and the evolution of the elemental composition of the fuel were carried out; and
- neutron radiation levels of the fresh and irradiated fuel were calculated.

The research was carried out using verified computational codes MCNP5 and Sources-4C (Wilson et al. 2009), high-precision experimental and computational data (Bulanenko 1979, Murata and Shibata 2002, Dulin and Zabrodskaya 2005, Vlaskin et al. 2015, Pigni et al. 2016, Simakov and Berg 2017, Vlaskin and Khomyakov 2021) as well as evaluated nuclear data libraries.

The IVG-1M reactor facility and methods for studying its characteristics

The IVG.1M reactor facility (Batyrbekov et al. 2015, Irkimbekov et al. 2019, Zhanbolatov et al. 2022) is a research pressurized water reactor with the core composed of 30 water-cooled technological channels arranged in three rows along concentric circles. Each such channel contains one fuel assembly (Fig. 1a) consisting of 468 fuel elements (Fig. 1b) (Zhanbolatov et al. 2022).

In the interchannel space, three types of beryllium displacers are installed. The reactor is controlled by 10 control drums located around the circumference behind the third row of the water-cooled technological channels. Each drum consist of an absorber, which occupies a sector of 120° on the circumference, and a reflector on the remaining surface. When the drum is turned towards the core, the material of the reflector or absorber introduces, respectively, positive or negative reactivity.

Neutronic model of the IVG.1M reactor

The computational model (Fig. 2) of the reactor core is made in a three-dimensional formulation with the preservation of the dimensional and mass parameters and material properties of each element of the reactor core specified in the design documentation; it is intended to determine the neutronic characteristics of the reactor (effective multiplication factor k_{eff} and other neutronic functionals) and the nuclide composition of the fuel.

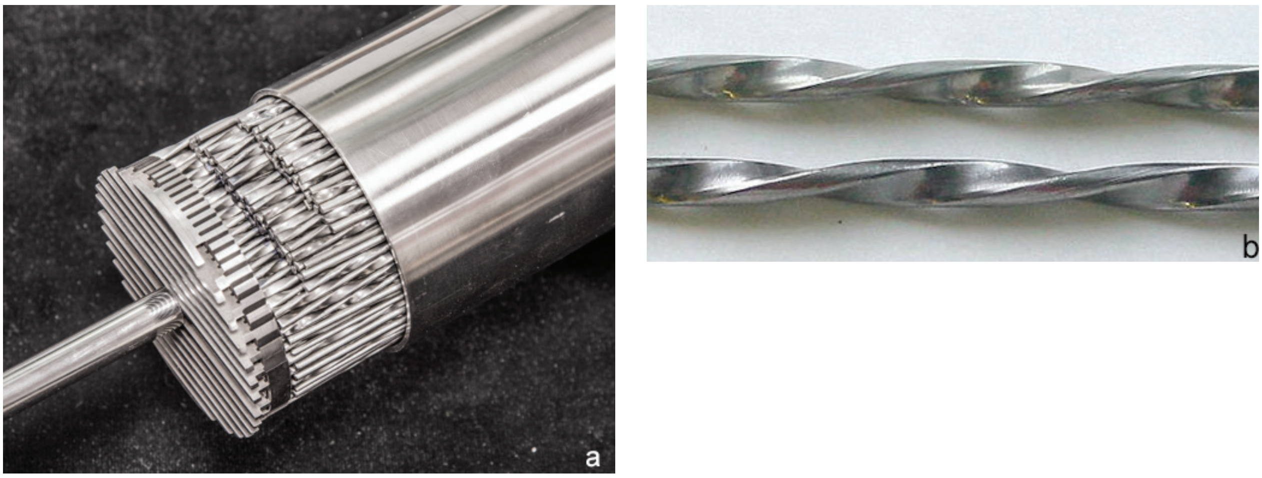

Figure 1. IVG.1M reactor fuel: **a.** Fuel assembly (Zhanbolatov et al. 2022); **b.** Fuel element

Calculation of the IVG.1M reactor neutronic characteristics

The neutronic studies and calculation of the fuel nuclide composition were performed using the MCNP5 (Monte Carlo *N*-Particle Transport Code System) code, based on the ENDF/BVII.0 evaluated nuclear data files (Evaluated Nuclear Data Library Descriptions) to solve the particle transport equations for the core volume.

To save computing resources:

1. the dimensions of the core elements, such as radius or length, were set as averages between the maximum and minimum values laid down in the design;
2. the tolerances for alignment or ovality of elements were not taken into account;
3. the small details of the core (fastening locks, bolts, etc.) were not modeled;
4. the absorbing elements of the control drums burned up evenly;
5. the impurities in the materials corresponded to the average values of the tolerances for the content of each element in accordance with the grade of the material; and
6. the SNF nuclide composition was obtained during continuous operation of the reactor for four years without fuel refueling.

In the calculation, $1{\cdot}10^7$ histories were played, which made it possible to ensure the accuracy of the desired functionals equal to 0.1%.

Calculation of the IVG.1M reactor fuel element

The IVG.1M reactor fuel element belongs to the plate-type fuel and has a complex spiral-shaped composite structure (see Figs 1b, 3).

The fuel kernel was made of 133 metal uranium filaments (https://stranarosatom.ru/2020/01/20/133-uranovye-niti-npo-luch-sozdalo-inn), the density of the core in the calculations was assumed to be 7.74 g/cm^3. Each uranium filament is coated with a thin layer of E-110 zirconium alloy with a density of 6.53 g/cm^3. The kernel coating is made of the same alloy as the uranium filament coatings. The computational model for the MCNP5 code is shown in Fig. 3b.

To calculate the source of alpha particles $N_\alpha(E)$, the neutron yield $Y_n(E)$ and their energy distribution $S_n(E)$, the real fuel element configuration (Fig. 3a) is replaced by an equivalent model. The equivalent model (calculation model for the Sources-4C code) is a homogenized cylindrical rod with a diameter $d = 1.66$ mm, height $h = 600$ mm, and coating thickness = 0.25 mm.

Calculation of the neutron background of the IVG.1M reactor fuel element

The neutron component of the radiation characteristics of the IVG.1M reactor fuel was calculated by jointly using the Sources-4C code (Wilson et al. 2009) and an analytical model that made it possible to calculate the transfer of alpha particles in the material of the cylindrical rod and in its coating.

The calculation was performed according to the following algorithm (Bedenko et al. 2018, Ghal-Eh et al. 2019):

1. the parameters of the source of alpha particles are determined (E_α is the energy of alpha particles produced in the rod, MeV; $R_\alpha(E)$ is their range, μm; $N_\alpha(E)$ is the yield of alpha particles, $\alpha{\cdot}s^{-1}$; N_{nucl} is the concentration alpha emitters, cm^{-3});
2. the transfer of alpha particles is calculated, i.e., the fraction of the remaining radiation in the rod ($1 - P_\alpha(E)$) and the emitted $P_\alpha(E)$ from its lateral surface is determined (here, the value of $P_\alpha(E)$ can be obtained by using the principles of geometric probabilities proposed for the first time by Buffon and Gaines; examples of application are given in (Bedenko et al. 2018, 2019, 2019a, Ghal-Eh et al. 2019));
3. the spectrum of alpha particles inside the rod $N_\alpha(E){\cdot}(1 - P_\alpha(E)), \alpha{\cdot}s^{-1}$ is calculated;
4. the spectrum on the lateral surface of the rod is calculated $n_\alpha(\mathrm{E}) = -d(P_\alpha(E,E_\alpha))/dE$, $\alpha{\cdot}(s{\cdot}\mathrm{MeV})^{-1}$;
5. a multigroup spectrum of alpha particles $F^{(i)} = \int n_\alpha(E)\, dE$, $\alpha{\cdot}s^{-1}$ is formed for the Sources-4C code; and
6. the neutron yield $Y_n(E)$ and their energy distribution $S_n(E)$ are calculated using the *Homogeneous Mixture Problems* and *Beam Problems* models (Wilson et al. 2009).

Results and discussion

The proposed tools make it possible to evaluate the dose of neutron radiation from the fuel and to revise the traditional procedures for handling fresh and irradiated fuel; the found value of $Y_{\alpha n}/Y_{sf}$, using the Rossi-alpha method, will allow us to estimate k_{eff} (the method for determining k_{eff} in subcritical systems by the Rossi-alpha method with a known contribution of (α, n)-neutrons is given by the authors in (Dulin and Matveyenko 2002, Dulin and Zabrodskaya 2005, Grabezhnoy et al. 2021) related to a beam of irradiated fuel assemblies prepared for transport and technological operations and "dry" storage.

Fig. 4 shows the normalized distribution $\chi_n(\mathrm{E})$ (Fig. 4a) and the total neutron spectrum $\chi_n(E)$ (Fig. 4b) of fresh and irradiated fuel of the IVG.1M reactor. The established dependences can be useful for neutron methods of passive non-destructive analysis of SNF.

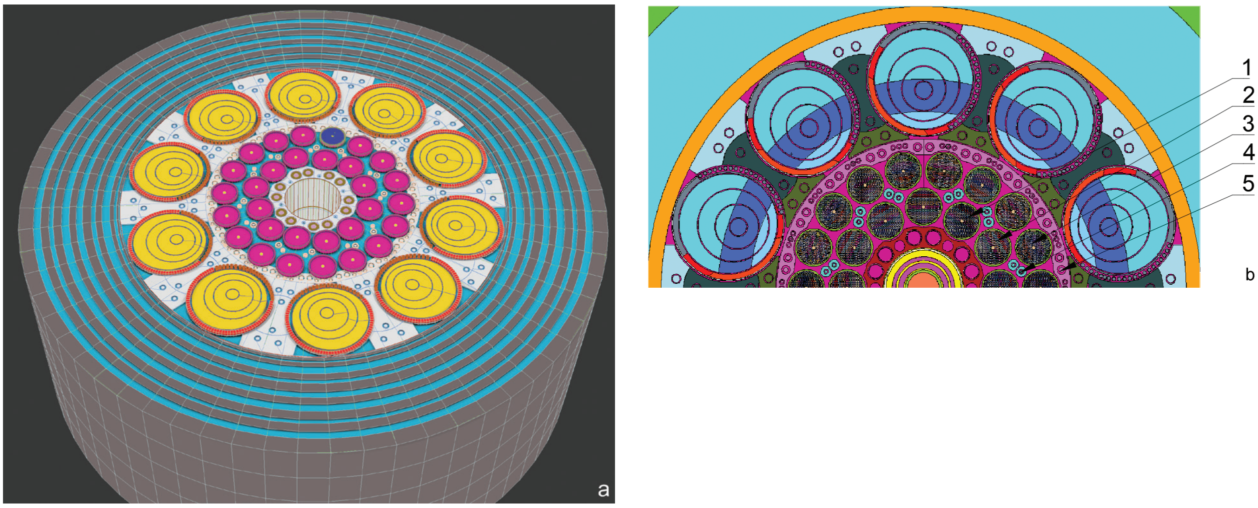

Figure 2. Computational model of the reactor core for the MCNP5 code (Irkimbekov et al. 2019): **a.** 3D model of the IVG.1M reactor core; **b.** Cross section of the 3D model (1, 2, 3 – water-cooled technological channel of the first, second and third row; 4 – interchannel beryllium displacers; 5 – side beryllium displacer)

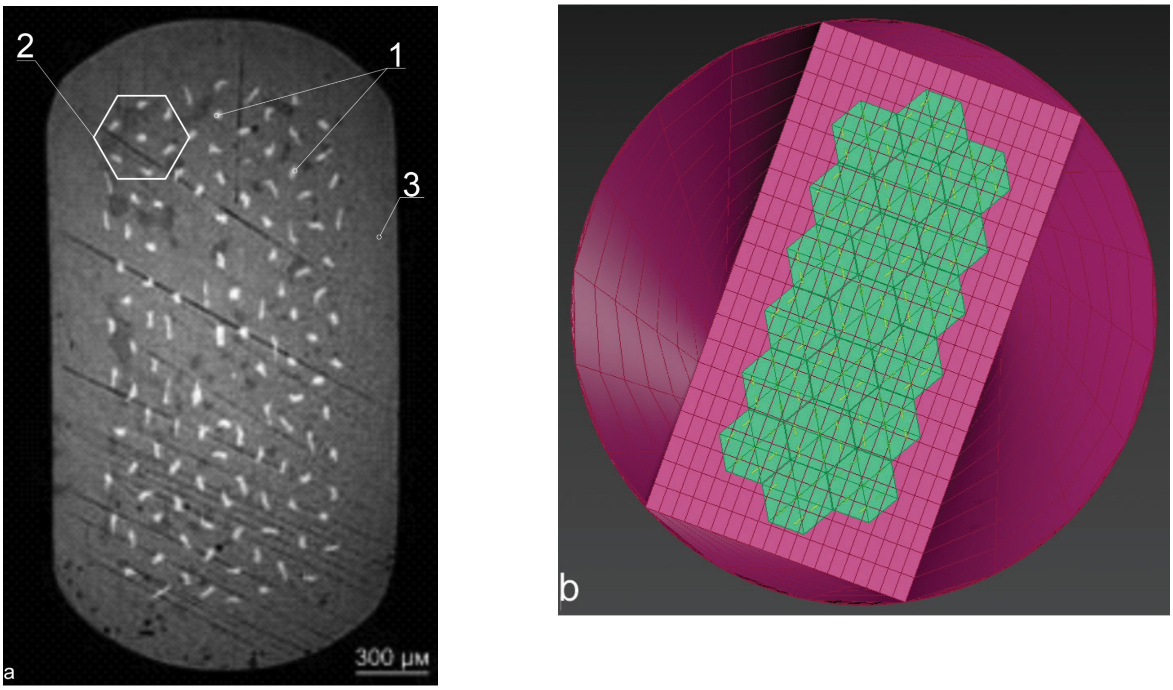

Figure 3. IVG.1M reactor fuel element: **a.** Actual configuration (1 – uranium filaments; 2 – cell with uranium filaments; 3 – fuel element coating); **b.** Scheme of the computational model for the MCNP5 code

The integral neutron yield of fresh fuel from the entire core $Y_n = 3.28{\cdot}10^2$ n·s^{-1} and the contribution of neutrons from the (α, n) reaction $Y_{\alpha n}/Y_{sf}$ (the ratio of the neutron yield from the reaction channel (α, n) Yαn to the neutron yield spontaneous fission Y_{sf}) are 0.0403 (~ 4.03%).

The normalized distribution of the neutron spectrum in the energy range from $1{\cdot}10^{-3}$ eV to 5.6 MeV can be approximated by a function of the form $\chi_{sf}(E) = s{\cdot}\exp(-E/0.827)\ \sinh(4.445{\cdot}E)^{1/2}$ (99,81% of spontaneous neutrons are formed as a result of spontaneous fission of ^{238}U), the observed high-energy part of this spectrum for neutron energies above 5.6 MeV is formed by neutrons during the (α, n) reaction on ^{9}Be (Fig. 4a, curves 1, 3 and Fig. 4b) when the alpha particles, which are formed during the radioactive decay of isotopes $^{234,235,236,238}U$, interact. It should be noted that 94.2% of all the alpha particles are formed as a result of ^{234}U decay with a half-life $T_{1/2} = 2.455{\cdot}10^5$ years, i.e., within the framework of our problem, this radionuclide is assumed to be stable, which gives grounds to consider the (α, n)-component of the neutron background to be constant in time.

For the irradiated fuel, the integral neutron yield is 2.28 times higher than for the unirradiated one and is equal to $7.49{\cdot}10^2$ n·s^{-1}; the $Y_{\alpha n}/Y_{sf}$ contribution is 0.1029 (~ 10.29%). The high-energy part of the spectrum is also formed by neutrons in the (α, n) reaction on ^{9}Be, which is part of the cermet-based fuel, and practically does not change with time.

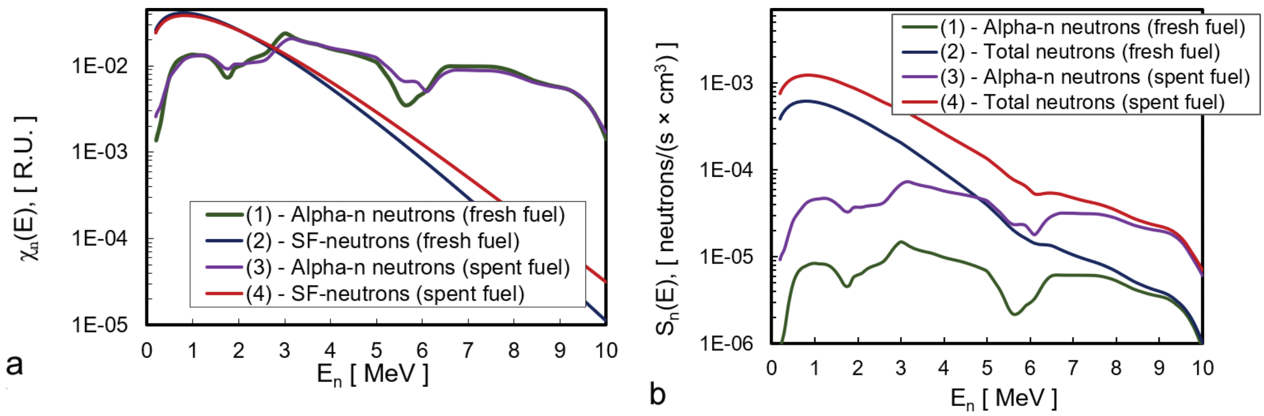

Figure 4. The neutron component of the radiation characteristics of the IVG.1M reactor fuel: **a.** The normalized distribution of neutrons (1 – (α, n)-neutrons of fresh fuel; *sf*-neutrons of fresh fuel; 3 – (α, n)-neutrons of irradiated fuel; 4 – *sf*-neutrons of irradiated fuel); **b.** The neutron spectrum (1 – fresh fuel; 2 – irradiated fuel)

The spontaneous fission neutron source (see Fig. 4, curves 2 and 4), in contrast to the fresh fuel, is formed by three nuclides: ^{238}U ($T_{1/2}$ = 4.468·10^{9} years, the contribution = 40.69%), ^{240}Pu ($T_{1/2}$ = 6564 years, the contribution = 43.86%) and ^{242}Cm ($T_{1/2}$ = 162.8 days, the contribution = 14.69%). All these radionuclides can be considered stable, except for the short-lived ^{242}Cm, which will almost completely decay after 3.57 years (~ 8·$T_{1/2}$). However, in calculating the personnel dose loads, we recommend observing the principle of conservatism, i.e., disregard the decay of this isotope.

The irradiated assemblies of the IVG.1M reactor are stored in a “dry” way, three pieces in a bundle in a specialized storage facility located on the territory of the reactor complex of the National Nuclear Center of the Republic of Kazakhstan. Previously performed neutronic calculations showed that the k_{eff} of a beam from three fresh assemblies is 0.01. Consequently, the neutron multiplication ~ $(1 - k_{eff})^{-1}$ in this beam due to stimulated fission will be 1.01 (i.e., the contribution of stimulated fission neutrons will not exceed ~ 1%) and may be left out of account in calculations of the neutron leakage spectrum of both fresh neutrons and irradiated fuel assemblies.

Conclusions

The authors propose a procedure for calculating the neutron component of the radiation characteristics of the innovative nuclear fuel of the IVG.1M reactor, and also estimate the (α, n)-component $Y_{\alpha n}/Y_{sf}$. The research results showed that the integral neutron yield of the fresh fuel from the entire core is Y_n = 3.28·10^{2} n·s^{-1}, and the contribution of $Y_{\alpha n}/Y_{sf}$, is 4.03%. For the SNF (burnup 8 MWt·day/kg (U), exposure 180 days), the integral neutron yield is 2.28 times higher, and $Y_{\alpha n}/Y_s$ = 10.29%. The high-energy component of the neutron spectrum, both in the first and in the second case, is formed by neutrons in the (α, n) reaction on ^{9}Be, which is part of the fuel, and practically does not change with time.

Based on the results of the performed studies, input data sets were prepared on the neutron yield Y_n and their spectral distribution $S_n(E)$ for the MCNP5 code and subsequent calculation of the SNF integral and differential dosimetric characteristics. This will make it possible to revise the procedures and regulations for handling fuel after its operation, established by regulatory documents.

The recommendations for safe long-term SNF storage announced for the purpose of this study will be developed in the second phase of the project after completion of reactor experiments and CFD modeling of transport packages.

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