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Study on an open fuel cycle of IVG.1M research reactor operating with LEU-fuel

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ABSTRACT

The fuel cycle characteristics of the IVG.1M reactor were studied within the framework of the research reactor conversion program to modernize the IVG.1M reactor. Optimum use of the nuclear fuel and reactor was achieved through routine methods which included partial fuel reloading combined with scheduled maintenance operations. Since, the additional problem in planning the fuel cycle of the IVG.1M reactor was the poisoning of the beryllium parts of the core, reflector, and control system. An assessment of the residual power and composition of spent fuel is necessary for the selection and justification of the technology for its subsequent management. Computational studies were performed using the MCNP6.1 program and the neutronics model of the IVG.1M reactor. The proposed scheme of annual partial fuel reloading allows for maintaining a high reactor reactivity margin, stabilizing it within 2–4 β_{eff} for 20 years, and achieving a burnup of 9.9–10.8 MW × day/kg U in the steady state mode of fuel reloading. Spent fuel immediately after unloading from the reactor can be placed in a transport packaging cask for shipping or safely stored in dry storage at the research reactor site.

1. Introduction

The IVG.1M research reactor was constructed through modernization of the IVG.1 reactor, and used to test fuel assemblies (FA) of nuclear rocket engine reactors and nuclear power propulsion systems [1]. The design of the IVG.1M research reactor was mainly based on the use of highly-enriched uranium fuel (HEU-fuel) (i.e., 90% of ^{235}U). Therewith, the loading of the IVG.1M reactor with the ^{235}U was 4.6 kg.

The IVG.1M reactor with HEU-fuel was actively used to study both the behavior of materials in nuclear and thermonuclear technology - fuel rods, fuel assemblies, and structural materials under reactor irradiation, as well as in transient and emergency conditions of its operation [2,3], and to study the physical characteristics of reactor radiations [4] and its effects on biological materials [5].

In 2010, within the framework of Kazakhstan-USA collaboration, the IVG.1M reactor was included in the program for conversion of research reactors to low-enriched uranium fuel (hereinafter—conversion) implemented under the auspices of the IAEA. The conversion of the IVG.1M reactor is currently conducted jointly with the National Nuclear Center of the Republic of Kazakhstan, Argonne National Laboratory (ANL, USA), Idaho National Laboratory (INL, USA), and Scientific Production Association LUCH (Russia).

Preliminary calculations and analytical evaluations confirmed the theoretical feasibility of IVG.1M reactor conversion to LEU-fuel and also the successful practical implementation of the technology for its manufacture [6,7] predetermined the technical feasibility of this conversion. In 2021, for the conversion of the IVG.1M reactor, a batch of fuel assemblies (FA) was finally manufactured, limited to a single load of the core. The FAs with LEU-fuel were delivered to the National Nuclear Center of the Republic of Kazakhstan. In April 2022, the loading of the reactor core with fresh LEU-fuel was completed, and on May 6, 2022, the physical start-up of the IVG.1M reactor was successfully conducted. The power start-up of the IVG.1M reactor is scheduled for 2023 when the reactor is supposed

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to become operational.

An important stage in the conversion was the equipping of the reactor with a coolant cooling system. In this regard, it became possible to increase the intensity of the reactor operation by reducing the duration of forced downtime. These downtimes are necessary to reject the heat that accumulates in the coolant during the reactor start-up. In this regard, the problem of maintaining the reactivity margin of the reactor at an acceptable level by periodically loading fresh fuel into the reactor becomes urgent. For this purpose, the rationale for the effective refueling organization of the IVG.1M reactor, the assessment of its service life, and the spent nuclear fuel management are being implemented.

2. Background

2.1. The IVG.1M research reactor complex

The IVG.1M reactor is a research pressurized heterogeneous thermal neutron reactor using a light water moderator and coolant and also a beryllium neutron reflector (see Fig. 1). The core of the IVG.1M reactor consists of 30 water-cooled fuel channels (WCFC) located in 3 rows on concentric circles of different radii (see Fig. 2). The FA consists of a thin-walled cylindrical case (AMg-5 alloy: Al – 95%, Mg – 5%), permeable end grids and a stack of fuel rods (spiral-rod, two-blade profile) laid on a triangular grid and sealed with cylindrical fillers. The WCFC is loaded into cells located along three concentric circles with a radius of 156 mm (1st row), 163.5 mm (2nd row), and 239 mm (3rd row), respectively. The diameter of the

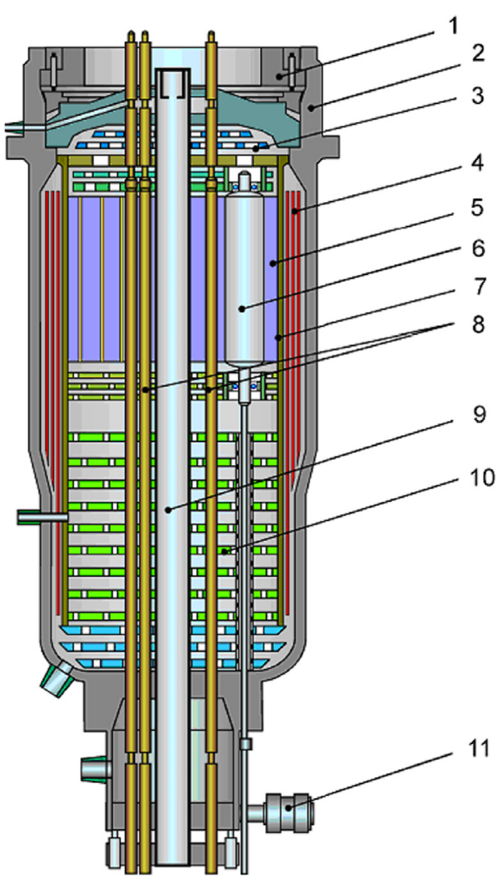


Fig. 1. Scheme of the vertical section of the IVG.1M reactor: 1 – Lid; 2 – Vessel; 3 – Top shields; 4 – Side shields; 5 – Reflector; 6 – Control drums; 7 – Central assembly; 8 – Fuel channels; 9 – Loop experimental channel; 10 – Iron-water protection; 11 – Control drum drive.

fuel channel is 76 mm. The FAs are placed in the WCFC at the core level. The FAs of the first and second rows are identical. The length of their active part is 800 mm, the content of ^{235}U is 0.212 kg, and the mass of all fuel rods is 10.046 kg. The length of the active part of the fuel assemblies of the third row is 600 mm, the content of ^{235}U is 0.178 kg, and the mass of all fuel rods is 7.647 kg. The radial profiling of the uranium load with a stepped division into two zones is provided. The numbers of fuel rods in the WCFC in the central and peripheral zones are 276 and –192, respectively (see Fig. 3). In fuel rods of LEU-fuel, the uranium in the form of thin metal filaments (see Fig. 4) is evenly distributed in the zirconium matrix, where, in total, one fuel rod contains 133 uranium filaments (see Fig. 4a and b). Such fuel is unique in its thermophysical and mechanical properties [11], and is not used in other reactor facilities.

The reactor is controlled by a group of 10 rotating control drums that compensate for the loss of reactivity due to the processes of burnup, poisoning, and slagging off the core, as well as the temperature effects and also the effects associated with the accumulation of ^{6}Li and ^{3}He isotopes in the beryllium elements of the core (see Fig. 2). The efficiency of the control system and the characteristics of the reactor cooling system altogether provide the possibility of implementing single start-ups with an energy output of 36 MW $\times$ h (1.5 MW $\times$ day) at a stationary power level of up to 10 MW with a thermal neutron flux density in the empty central experimental channel up to 2.5×10^{14} neutron $\times$ cm^{-2} $\times$ s^{-1}.

The IVG.1M reactor with LEU-fuel has a designed reactivity margin of 6.4 β_{eff}, which is 3.2 β_{eff} higher than the reactivity margin of the reactor with HEU-fuel. Increasing the reactivity margin allows for an increase in fuel burnup. The reactor is equipped with a forced cooling system for the primary coolant, which is used between reactor start-ups. This technology makes it possible to increase the frequency of reactor start-ups [8], but at the same time, the reactor fuel lifetime is inevitably reduced. The latter means that the long-term use of the reactor for a time exceeding its campaign is possible only if the composition of the core is periodically renewed by loading fresh fuel.

Since the fuel in the IVG.1M reactor has not been not regularly reloaded for 30 years, the problem of optimizing this process has never been resolved. The transport and technological operations with spent fuel assemblies (SFA) of the IVG.1M reactor consist of a set of technological measures for unloading, transportation, temporary storage, cutting, and loading SFA into a transport packaging set for further transportation to the SFA storage site. The long-term container storage of SFAs is conducted at a special site located on the territory of the National Nuclear Center of the Republic of Kazakhstan. An important feature of the IVG.1M research reactor complex is the absence of a spent fuel pool. In this regard, fuel aging should be implemented directly in the reactor core.

The transport and technological operations with SFA of the IVG.1M reactor are conducted according to the following scheme. After the shutdown of the IVG.1M reactor, the SFAs are conditioned in order to reduce the activity of the irradiated fuel rods. After a decrease in activity, the water-cooled fuel channels (WCFC) with spent nuclear fuel are unloaded from the IVG.1M reactor core and placed in an intermediate dry storage, where they can be kept for up to one year and subject to cutting. The SFAs are separated from other structural elements of the WCFC. After cutting, the SFAs are placed in hermetically sealed canisters. Three canisters can be grouped and put in a case of the transport and packaging set. In this form, SFAs are placed on the site of the research reactor complex for temporary dry storage. The transportation to a special site for long-term storage is conducted in special transport and packaging sets. The entire activity is performed in accordance with the legislation of the Republic of Kazakhstan on the transportation of irradiated

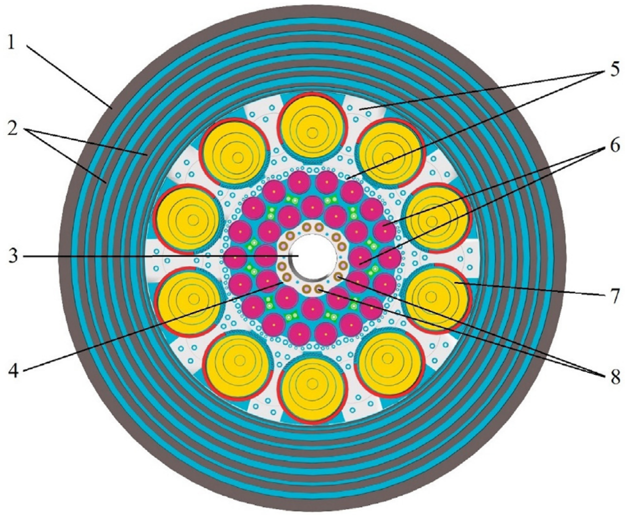


Fig. 2. Scheme of the cross-section of the IVG.1M reactor: 1 – Vessel; 2 – Side shields; 3 – Loop channel; 4 – Central displacer; 5 – Reflector; 6 – WCFC; 7 – Control drum; 8 – Reactivity compensation rods.

nuclear fuel [9]. The main issues during the storage and transportation of SFAs are related to criticality, thermal state, and the activity of some elements.

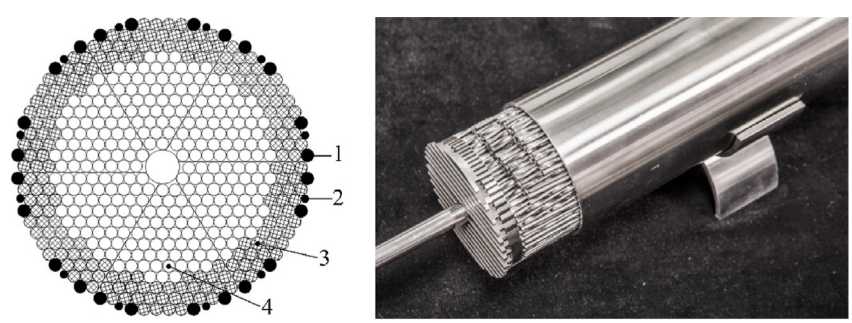
The systems of the IVG.1M reactor are subject to annual checks during the period of routine maintenance when the reactor startups are not implemented. They include checking the operation of the reactor control and protection system (CPS), and the instrumentation and equipment of the transport and technological systems. The check of the CPS is conducted within one month and cannot be combined with other types of routine maintenance or with fuel reloading. Therefore, it is assumed that the CPS should be checked immediately after the reactor shutdown. The duration of routine maintenance at the shutdown of IVG.1M reactor is up to 70 days. During the first 30 days of routine maintenance (i.e., CPS verification), all SFAs selected for replacement with fresh fuel may remain in the reactor in order to reduce their activity. Over the next 40 days, the fuel can be reloaded without adversely affecting the efficiency of the reactor, which is sufficient to reload six fuel assemblies.

The obvious advantage of this approach is that the fuel will be reloaded in parallel with other tasks as an integral part of the annual schedule of the IVG.1M reactor operation.

2.2. Model of the IVG.1M reactor

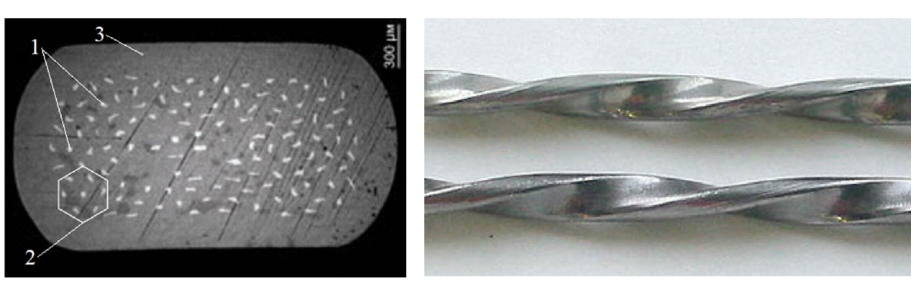
The calculations of neutronic characteristics of the reactor were undertaken with the Monte Carlo code, MCNP6.1 (ENDF/BVII.0) [10], based on the model previously developed by Irkimbekov et al. [11]. The neutronics model was made in three-dimensional settings (see Fig. 5) and maximally corresponded to the actual reactor configurations in terms of geometry and material composition.

The model can be used to calculate the effective neutron multiplication factor (k_{eff}), reactivity margin (ρ_{eff}), neutron flux density, reactor thermal power distribution, and the spent fuel activity for different elemental compositions of fuel and structural materials. In addition, the calculation model implements the possibility of rotating the control drums to the required angle and installing reactivity compensation rods to simulate a specific critical state.



(a) (b)

Fig. 3. FA of the IVG.1M reactor:(a) Map of the WCFCs: 1, 2 – Filler; 3 – Fuel rods of the peripheral area; 4 – Fuel rods of the central area; (b) General view of the reactor FA.



(a) (b)

Fig. 4. Nuclear fuel rod of the IVG.1M reactor [11]:(a) Cross-section of the fuel rod: 1 – uranium filaments, 2 – a cell with uranium filaments, 3 – fuel rod cladding; (b) General view of the fuel rod.

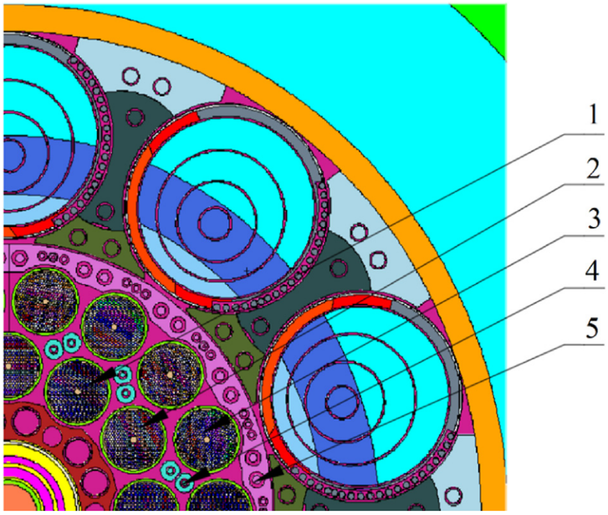


Fig. 5. Cross-section of the IVG.1M reactor 3D-model:1, 2, 3 – Water-cooled fuel channel of the first, second and third rows, respectively; 4 – Intra-channel beryllium displacers; 5 – Lateral beryllium displacers.

To simulate heterogeneous fuel burnup, each FA is spatially divided into separate volumes with uniformly-distributed burnup. The fuel rods of the FAs are combined into 16 groups. In terms of height, the FA is divided into 6 and 8 parts with 600 mm and 800 mm active channels, representing 96 and 126 groups, respectively (see Fig. 6).

To save computational resources, some simplifications and approximations were taken into account: (1) the impurities in materials were considered as the average values for the content of each element in accordance with the brand of material; (2) the dimensions of the core elements, such as radius or length, were set as the averages between maximum and minimum values of the design; (3) the coaxiality or ovality of elements were not taken into account; (4) the small details of the core (fastening locks, bolts, etc.) were not modeled; (5) the absorbing elements of the adjusting drums were assumed to be burnup evenly.

In the burnup calculation, 10^7 histories (for each burnup step) were recorded, which allowed ensuring the standard deviation of the calculated k_{eff} values equal to 0.00050 ($\Delta\rho < 0.07\ \beta_{eff}$). It is assumed that the control elements are deployed by absorbing elements 180° from the active zone. The reactivity effects associated with the heating of the core parts, the movement of control elements, or the accumulation of xenon during the operation of the reactor were not taken into account.

The calculation model of the reactor was validated by the procedures listed below.

The model was validated by k_{eff} using the results of reactor start-ups with fresh and burned-out highly enriched fuel (1990–2016), as well as using the results of start-ups to study two assemblies with LEU-fuel (2016–2019) [16]. As part of the testing of two fuel assemblies with LEU-fuel in the IVG.1M reactor, data were obtained on the excess power of fuel assemblies with LEU-fuel, which was confirmed by calculations using the presented model.

Feasibility study of the conversion of the IVG.1M reactor to LEU-fuel and justification of the IVG.1M reactor safety with LEU-fuel [12]. As part of the work to justify the safety of the IVG.1M reactor with LEU-fuel, the results were validated using the example of the reactor operation mode with power maneuvering. Using the presented model, data were calculated on the effect of heating of individual elements of the core on the reactivity of the reactor. These results were used to calculate the kinetics of the IVG.1M reactor during power up and down.

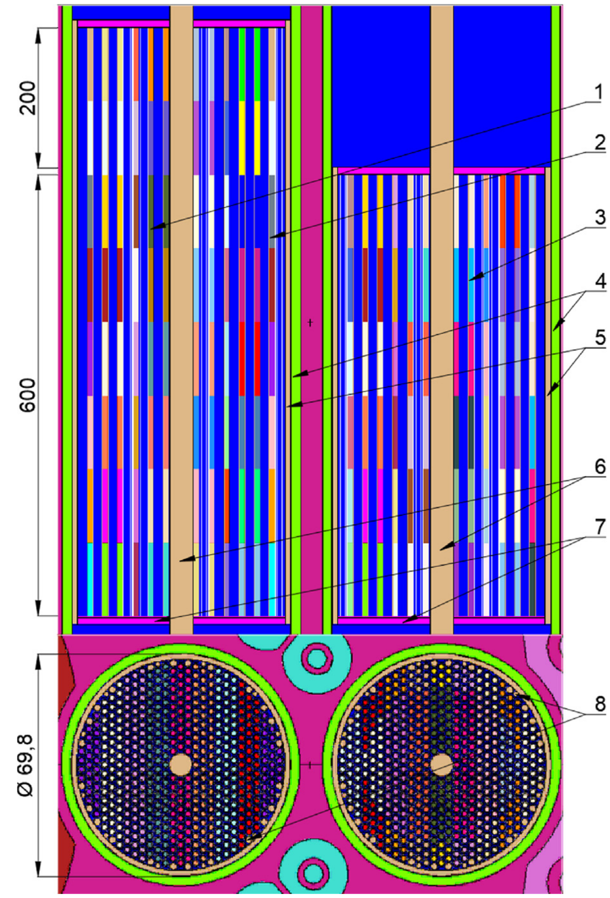


Fig. 6. Slicing planes (vertical and horizontal views) of a computational model for the WCFC of the IVG.1M reactor:1, 2 – Nuclear fuel rod of the first and second rows with the length of the active part of 800 mm; 3 – Nuclear fuel rod of the third row with an active part length of 600 mm; 4 – Vessel; 5 – Clip; 6 – Rod; 7 – End grate; 8 – Filler.

Study of the characteristics of the spent HEU-fuel of the IVG.1M reactor [13]. When studying the characteristics of spent HEU fuel from the IVG.1M reactor, the problem of calculating the main isotopes that contribute to the spent fuel activity was successfully solved.

Estimation of neutron component of SFA [13]. As part of the work on assessing the neutron component of SFAs, only computational work was carried out to study the possibility of controlling the burnup of fresh and spent fuel.

Reactor parameters calculation for physical start-up [14]. Before the physical start-up of the IVG.1M reactor, a number of parameters were calculated, such as the power distribution over individual fuel rods in the fuel assemblies and the rate of neutron reactions with activation detectors. These results are awaiting experimental confirmation.

2.3. Conditions for selecting a reactor campaign scenario

The development and calculation justification of possible scenarios for the operation of the IVG.1M reactor fuel campaign with LEU were conducted based on both the above-described possibility of annual reloading of SFAs for fresh fuel and the efficient use of fuel by increasing the maximum burnup [15].

For calculations of the reactor fuel campaign, the following conditions were taken into consideration:

(1) The reactor campaign is a multiple of one year, during which 40 start-ups are conducted each with a period of 7 days; (2) Annual power output at the reactor is 60 MW × day; (3) Fuel replacement is linked to routine maintenance and should be performed every year or at intervals that are multiples of one year; (4) At the end of the reactor campaign, the reactivity margin should be at least 2 β_{eff} (this value was acquired based on the experience of operating the reactor with HEU-fuel); (5) The layout of the reactor core can be changed only by loading fresh fuel because the rearrangement of FAs is impossible due to the different lengths of the inner and outer rows of FAs.

Based on the above-mentioned conditions and after providing calculations of 3–5 reloadings, a partial fuel replacement scenario, which allows increasing fuel usage efficiency by increasing maximum burnup, was selected [20]. Results of studies for this scenario of the IVG.1M reactor with LEU fuel campaign for a long period are presented below. Fig. 7 shows the proposed map for replacing FA groups for partial reloading of the core. According to the map, one group of 6 FAs is simultaneously replaced for each reload. Groups one through five are replaced sequentially at the end of the next reactor campaign.

3. Results of calculations

3.1. Reactor campaign

The reactivity margin of the reactor is 6.42 β_{eff} when the entire core is initially loaded with fresh fuel. However, over 4 years (with power generation of 240 MW × day), the reactivity margin will decrease to 2.16 β_{eff}, and by the end of the fifth year of operation to 1.5 β_{eff}. Thus, the first reactor campaign will last 4 years.

The reactivity margin increases to 6.27 β_{eff} with a one-time fuel replacement after a four-year reactor campaign. During the next four-year campaign, the reactor reactivity will drop to 1.94 β_{eff} (see Fig. 8). Thereafter, only limited use of the reactor for experimental purposes will remain possible, as shown in the example of the third load. Meeting the minimum reactivity margin requirement will become impossible already in the 12th year of reactor operation. After 15 years, the fuel campaign should be reduced to three years, as shown in the example of the fourth fuel load.

According to the conditions considered for the fuel campaign calculations, the switch to partial core fuel reloading should begin after the completion of the first campaign of the reactor with an energy output of 240 MW × day (see Fig. 9, point 3). Hereafter, every time after the power output of 60 MW × day, the fuel is reloaded according to the map (see Fig. 7).

The replacement of the initial load fuel is accompanied by a constant increase in the reactivity margin compared to the previous load in the period from 240 to 480 MW × day. This is due to the fact that the fuel lifetime of the first load exceeds the fuel lifetime during the period of the established fuel campaign. The last replacement of the fuel assemblies belonging to the fifth group of the initial load will be conducted upon reaching the power output of 480 MW × day. The fuel burn-out in the fifth group of FAs will be 15.8 MW × day/kg U. After the last FAs are unloaded from the initial load, the steady-state fuel campaign will be 5 years (300 MW × day). Fig. 9 shows that each time the refueling efficiency is different. The refueling efficiency of the first and second rows is always higher than the refueling efficiency of the third row. This can be traced if each increase in the reactivity margin is associated with the number of rows in which fuel assemblies are replaced (2, 3, 1, 3, 3, etc). The replacement point of the FA group of the first row is marked for example (see Fig. 9, point 5).

Calculations show that the annual replacement of six fuel assemblies leads to an increase in the duration of the fuel campaign under the accepted assumptions for power generation and the value of the residual reactivity margin (see Fig. 9). As far as this fuel campaign organization is met, the maximum value of the reactor reactivity margin does not exceed 4 β_{eff}. It is possible to stabilize the reactor reactivity margin (see Fig. 9, line 1) above 2 β_{eff} (see Fig. 9,

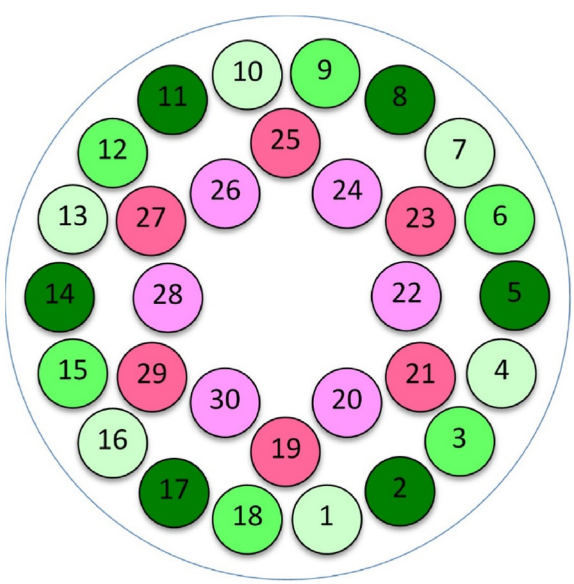


Fig. 7. Scheme of core reloading:Group 1 – WCFC of the 2 row Nos. 19, 21, 23, 25, 27, 29; Group 2 – WCFC of the 3 row Nos. 1, 4, 7, 10, 13, 16; Group 3 – WCFC of the 1 row Nos. 20, 22, 24, 26, 28, 30; Group 4 – WCFC of the 3 row Nos. 2, 5, 8, 11, 14, 17; Group 5 – WCFC of the 3 row Nos. 3, 6, 9, 12, 15, 18.

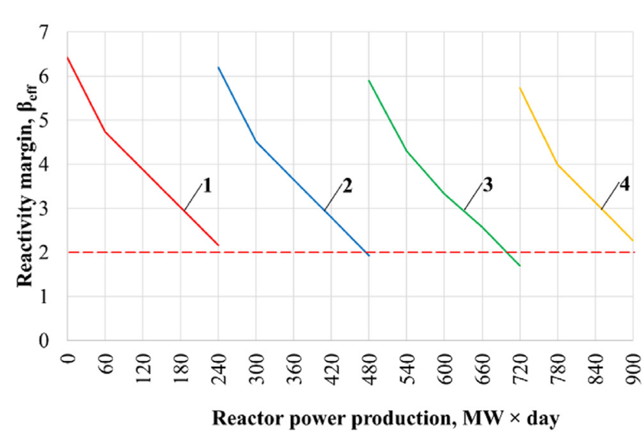


Fig. 8. Breeding properties of the IVG.1M reactor core for options:1 – First reactor campaign (4 years); 2 – Second reactor campaign (4 years); 3 – The third reactor campaign (4 years), The time of limited use of the reactor is 6 months; 4 – Fourth reactor campaign (3 years).

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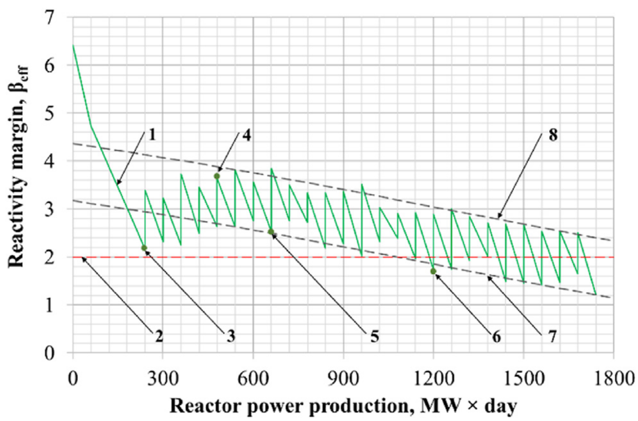

Fig. 9. Breeding properties of the IVG.1M reactor core during partial fuel reloading: 1 –change in reactivity margin; 2, 3 – recommended lower limit of reactivity margin end of the first reactor campaign; 4 – The first fuel load was fully replaced; 5 – The moment of unloading the FAs of the first row; 6 – Reduction of the reactivity margin below $2\beta_{eff}$; 7, 8 – The net effect of beryllium poisoning.

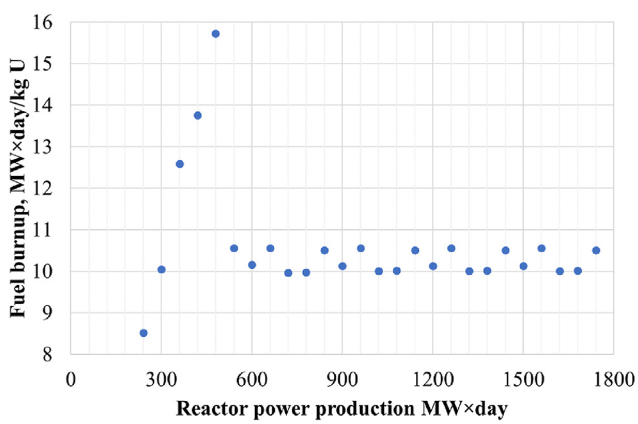

Fig. 10. Burnup of unloaded fuel.

line 2) for at least 20 years (up to reactor power output of 1200 MW × day, see Fig. 9, point 6). The downtrend of the reactivity margin in Fig. 9 is associated with the accumulation of isotopes with a high neutron absorption cross-section in the beryllium moderator.

Fig. 10 shows the burnup behavior of the unloaded fuel in each reactor campaign. The FA burnup between the first, second, and third rows is equalized. The fuel burnup of the first row will reach 10.2–10.8 MW × day/kg U, the second row - 10.3–10.7 MW × day/kg U, and the third row - 9.9–10.3 MW × day/kg U.

3.2. Beryllium poisoning in the core

The decrease in reactivity margin is due to both the fuel burnup and the accumulation of lithium and helium isotopes in the beryllium parts of the core. At the same time, if the reactivity loss due to the fuel burnup can be compensated by periodically loading fresh fuel into the reactor, the reactivity loss due to beryllium poisoning will constantly increase and eventually lead to the need to replace beryllium or to stop the reactor operation.

The effect of ^{6}Li and ^{3}He accumulations in beryllium on the reactor reactivity margin is shown in the calculation data in Fig. 11.

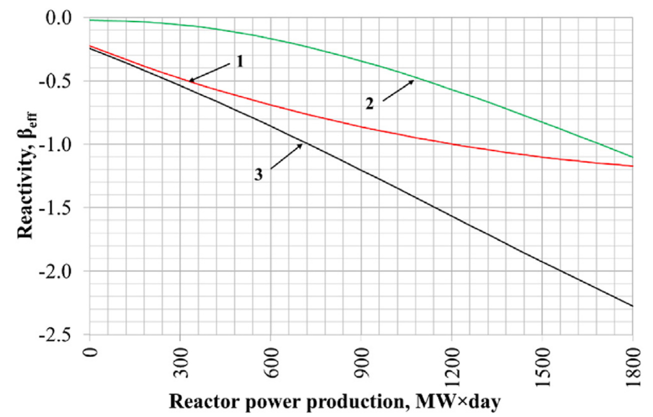

Fig. 11. Beryllium poisoning effect:1 – ^{6}Li poisoning, 2 – ^{3}He poisoning, and 3 – Total beryllium poisoning effect.

It can be seen that the ^{6}Li accumulation rate decreases, while the ^{3}He accumulation rate, on the contrary, increases. However, the effect of beryllium poisoning increases almost linearly at a rate of about 0.07 β_{eff}/year (for a calculation period of 30 years). The effect of beryllium poisoning is shown by lines (7, 8) in Fig. 9. These lines were obtained by parallel transfer to the points of the minimum and maximum values of the reactivity margin at a power output of 540 MW × day. The fluctuations of the reactivity margin in the corridor between these lines show the consistency of the results.

In addition to a negative effect on the reactivity, the accumulation of helium and tritium in beryllium during prolonged irradiation can lead to a change in its mechanical properties - radiation swelling of beryllium, a change in its strength and plasticity [16–20]. In a specific case, the radiation swelling of the material is of prime importance, since the IVG.1M reactor core contains movable beryllium components of the control system. Recent studies on the mechanical properties of beryllium samples installed in the core of the IVG.1M reactor confirm these conclusions [21].

3.3. SFAs of the IVG.1M reactor after irradiation

Two groups of FAs were selected to evaluate the characteristics of SFAs after irradiation in the reactor with a steady-state fuel campaign. The first group consists of FAs of the third row with a fuel burnup of 15.8 MW × day/kg U. This group of FAs is unloaded when the reactor output reaches 480 MW × day. This choice is due to the fact that the burnup in these FAs significantly exceeds the burnup in other FAs and therefore requires additional attention. The second group of FAs from the first row, according to the calculation conditions, will be unloaded at the reactor power output of 660 MW × day. The fuel burnup will reach 10.8 MW × day/kg U. This choice is caused by the fact that the FAs of the first row, at the end of their campaign, have the highest burnup in the steady-state fuel replacement mode.

A neutronics calculation was performed with the MCNP 6.1 code to determine the change in the isotopic composition of fission and decay products during a two-year period of fuel conditioning. The neutronics model presented in Figs. 6 and 7 was used in the calculations. The change in the decay heat over time was estimated using the Wey-Wigner formula [22,23].

The analysis of the calculation results for the activity and power of the decay heat in irradiated fuel was performed by taking into account 300 different isotopes that are in spent fuel in the form of

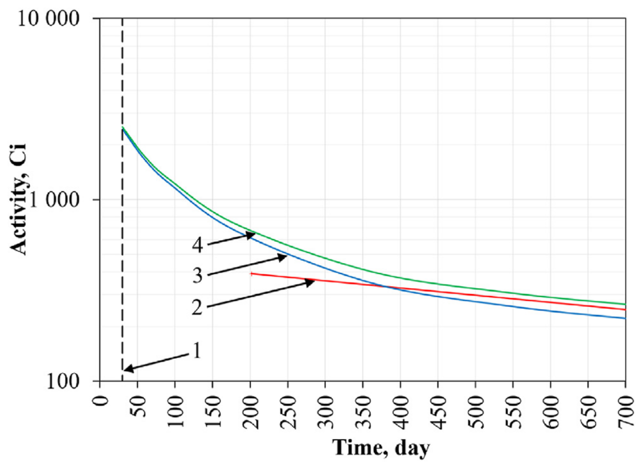

Fig. 12. The activity of the 800-mm high SFA after irradiation:1 – Moment of fuel unloading after conditioning in the core for 30 days; 2 – Activity of an SFA with HEU-fuel, 3 – SFA activity with LEU-fuel (10.8 MW × day /kg U), 4 – Spent assembly activity with LEU-fuel (15.8 MW × day /kg U).

uranium transmutation products, fission, and decay products. In 30 days after the reactor shutdown, short-lived radionuclides will decay, which decreases the activity (see Fig. 12) and decay heat (see Fig. 13) by 8 times. For comparison, the same figure shows the characteristics of FAs with HEU fuel with a burnup of a maximum of 35 MW × day/kg U. In general, the radiation and thermal parameters of SFAs with LEU-fuel exceed those of similar SFAs with HEU-fuel (see Fig. 12, lines 1 and 4).

According to [14], the average heat flux on the surface of a package or transport cask (i.e., transport packaging set) should not exceed 15 W/m^2. The value of the heat flux can be estimated by taking the transport packaging set TUK-19 with an outer surface area is 3.5 m^2 used for shipping the IVG.1M fuel as an example.

The total permitted power [14] of three FAs placed in TUK-19 can reach 52.5 W, or an average power of 17.5 W per FA. Fig. 13 shows the conditioning period of FAs in the reactor. It can be seen that the power of the FAs drops below 17.5 W during the conditioning. Thus, the thermal state of SFAs of the IVG.1M reactor allows shipping them using TUK-19 after unloading the fuel from the reactor. Therefore, further activity calculations for the fission and decay products were performed for SFAs after their conditioning for 30 days.

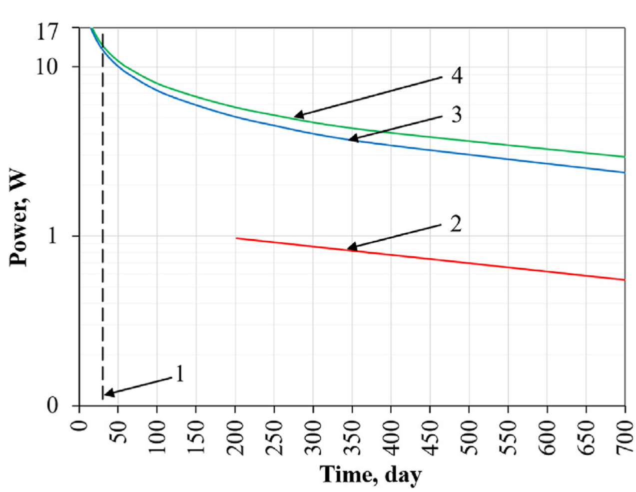

Fig. 13. The power of the heat decay of the 800-mm high SFA after irradiation:1 – Moment of fuel unloading after conditioning in the core for 30 days; 2 – Decay heat of assembly with HEU-fuel; 3 – Decay heat of assembly with LEU-fuel (10.8 MW × day/kg U); 4 – Decay heat of assembly with LEU-fuel (15.8 MW × day/kg U).

Table 1
Activity of SFA isotopes (in Curie unit) placed inside the transport packaging container.

	Calculated value for three SFAs		Permissible value, according to [14].	
			Min	Max
Burnup (MW × day/kg U)	15.8	10.8	-	-
^{3}H	29.8	20.5	1080	3200000
^{85}Kr	15.1	10.5	10	30000
^{89}Sr	581.9	581.9	16.2	48000
^{90}Sr	143.6	93.8	8.1	24000
^{90}Y	93.8	93.8	8.1	24000
^{91}Y	749.4	749.4	16.2	48000
^{95}Zr	851.5	851.5	54.1	162000
^{95}Nb	1054.0	1054.0	27	81000
^{103}Ru	343.3	343.3	54.1	162000
^{106}Ru	55.4	54.3	5.4	16000
^{131}I	44.3	44.3	81.1	243000
^{133}Xe	37.4	37.4	540.5	1600000
^{137}Cs	145.8	95.0	54.1	162000
^{140}Ba	243.5	243.5	13.5	40000
^{140}La	280.4	280.4	10.8	32000
^{141}Ce	592.1	592.1	540.5	1600000
^{144}Ce	705.1	699.8	5.4	16000
^{143}Pr	284.6	284.6	81.1	243000
^{147}Nd	69.0	69.0	1080	3200000
^{147}Pm	254.1	219.0	1080	3200000

According to the calculation results, 103 isotopes out of 300 have non-zero activity at the time of SFA unloading, and 33 of them are actinides. The total activity of SFAs at the time of unloading will not exceed 2511 Ci, whereas the activity of actinides is not more than 1.5 Ci. At least 99% of the contribution to the SFA activity is made by 19 fission and decay isotopes: ^{3}H, ^{89}Sr, ^{90}Sr, ^{90}Y, ^{91}Y, ^{95}Zr, ^{95}Nb, ^{103}Ru, ^{106}Ru, ^{131}I, ^{133}Xe, ^{137}Cs, ^{140}Ba, ^{140}La, ^{141}Ce, ^{144}Ce, ^{143}Pr, ^{147}Nd, ^{147}Pm.

Table 1 summarizes the calculated and allowable content of active isotopes in SFAs for their transportation in B(U) packages [14]. According to the results, 30 days after the reactor shutdown, the spent nuclear fuel in the selected configuration can be placed in transport packaging sets for shipping to reprocessing or permanent storage sites.

3.4. Nuclear physics studies on the shipping possibility of IVG.1M Reactor's SFAs

To justify the nuclear safety of SFA storage, the breeding characteristics of a three-FA package placed in a case of a transport packaging set were assessed.

SFAs are stored at a special site of the research reactor complex in special cases of transport packaging sets. Three spent FAs are placed in one case and the SFA cases are kept in cells of triangular-shaped lattice with a pitch of 250 mm.

The 3D modeling of the case for loading and spacing of spent fuel assemblies, as shown in Fig. 14, was carried out in terms of geometry and composition, as close as possible to the real design (Fig. 5). Nuclear physics studies were performed in the MCNP6.1

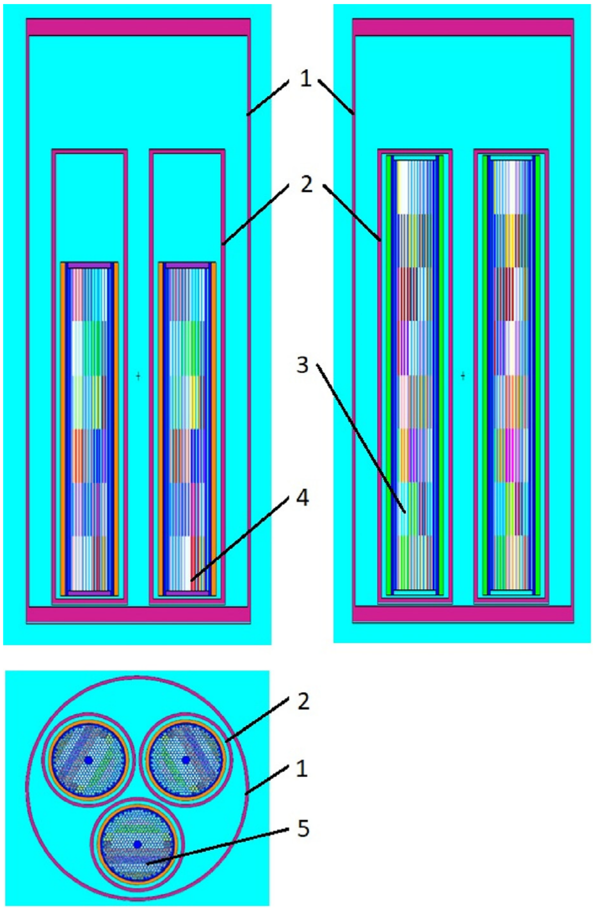

Fig. 14. Calculation model of a case with three SFAs:1 – A case of a transport packaging; 2 – Hermetically sealed canisters; 3 – Longitudinal section of 800 mm fuel assemblies; 4 – Longitudinal section of 600 mm fuel assemblies; 5 – Cross-section of fuel assemblies.

program code (ENDF/BVII.0) for fresh fuel assemblies (i.e., without taking into account the decrease in the breeding properties of the fuel during burnup) with two different heights of 800 mm and 600 mm and ^{235}U mass of 0.212 kg and 0.178 kg, respectively.

The results of nuclear physics studies of a dual-purpose universal case of a transport packaging set containing three SFAs of different types are given in Table 2. The calculations were performed for the conditions of normal and abnormal transport and technological situations, where the complete flooding of the case with water was provided. In addition, the calculations were made for normal and abnormal situations during the storage of a group of cases for the transport packaging set, in which the complete water flooding of the storage with simultaneous convergence of cases for the transport packaging set could be considered.

The calculation results showed that the SFA case is a deeply subcritical breeding system even when it is completely flooded with water ($k_{eff} = 0.35$). A storage facility with a flooded group of SFA cases is also subcritical ($k_{eff} = 0.53$). This result confirms that, under the considered conditions, the case of the transport packaging set with SFA is a nuclear-safe system.

4. Conclusions

Based on the results of the implemented studies, an effective organization of the fuel campaign for the IVG.1M reactor was substantiated, the service life was estimated, and the technology for the safe management of spent nuclear fuel of the reactor was considered.

Partial reloading of FAs (i.e., six assemblies per reactor campaign) allows for fuel replacement during scheduled maintenance at the research reactor complex which exhibits the most efficient use of the downtime of the reactor.

The calculation results of the changes in the isotopic composition in the core elements show that the selected mode of fuel reloading will allow for the uninterrupted operation of the reactor over the next 20 years. The limiting factor, in this case, will be the effect of beryllium poisoning in the core. At the end of this period, it will be necessary to revise the requirements for the fuel composition in the direction of increasing the uranium content. Moreover, during this period, it is necessary to monitor the condition of the moving beryllium parts of the CPS.

The maximum burnup at the end of the initial load fuel campaign with initially loaded fuel is 15.8 MW × day/kg U, such that the fuel campaign of these FAs will reach 8 years. However, the fuel burnup in the steady replacement mode will not exceed 10.8 MW × day/kg U. The relative difference in fuel burnup between the FAs in different rows will be 8%.

Calculations of the composition of two groups of FAs with maximum fuel burnup show that their immediate handling after unloading from the reactor will be safe within the established norms. The isotopic composition of fission products and the decay heat of the assembly make it possible to safely handle spent fuel at all stages of transport and technological operations at the research reactor complex.

From the results of nuclear physics studies on the possibility of spent fuel transport, one may deduce that under normal operating conditions and in the event of an abnormal situation, the case of the transport packaging set with the studied spent fuel assemblies and the storage facility within such cases is a deeply subcritical nuclear-safe system.

Finally, the results of the computational and analytical investigations on the characteristics of the fuel campaign for the IVG.1M reactor proved the possibility of implementing the specified fuel reloading mode using the available technological schemes for handling nuclear fuel at the IVG.1M research reactor complex.

Table 2
Breeding properties of the case with SFAs of various types.

Fuel assembly height (mm)	Quantity (pc)	^{235}U mass in a case (kg)	Effective multiplication factor	
			Normal situation	Abnormal situation
600	3	0.533	0.010	0.35
800	3	0.637	0.009	0.34

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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