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IN-PILE AND OUT-OF-PILE TESTS AND RESEARCH ON FAST CRITICAL ASSEMBLIES OF HIGH DENSITY URANIUM ZIRCONIUM CARBONITRIDE LEU FUEL

Abstract. UZrCN fuel is a high-density, high-temperature fuel that has potential for application in different type reactors. In the past, reactor tests using UZrCN HEU (96% U-235) fuel have been performed to low burnup. However, reactor-testing data are still needed at high burnup to confirm the optimal performance of this-type fuel. The SM-3 research reactor, which is a high-flux reactor located at the State Scientific Center – Research Institute of Atomic Reactors, Dimitrovgrad, Russia, will be used to test a UZrCN LEU (19.73% U-235) fuel to ~40% of burnup. The fuel will then be examined to determine its performance during irradiation.

On the "Giacint" and "Kristal" critical facilities located at the Joint Institute for Power and Nuclear Research – SOSNY of the National Academy of Sciences of Belarus, Minsk, Belarus, criticality experiments on multiplying systems modeling physical features of cores with UZrCN LEU (19.75% U-235) fuel have been prepared for use in works on fast reactors with gaseous and liquid-metal coolants. Critical assemblies represent uniform hexagonal lattices of fuel assemblies, each of which consists of 7 fuel rods and has no clad. The active fuel length is 500 mm. Clad material is stainless steel or Nb. Three types of fuel assemblies with different matrix material (air, aluminum and lead) are investigated. These are side radial, top and bottom reflectors – beryllium (internal layer) and stainless steel (external layer).

This article desribes the design of the experiment that will be performed in the SM-3 reactor and discusses the results of different calculations that have been performed to show that the experiment design will meet all objectives. The description of construction and composition of critical assemblies with UZrCN fuel and the calculation results are also presented.

Keywords: nuclear fuel, uranium zirconium carbonitride, in-pile and out-of-pile tests, critical assemblies

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РЕАКТОРНЫЕ И ПОСЛЕРЕАКТОРНЫЕ ИСПЫТАНИЯ И ИССЛЕДОВАНИЯ НА БЫСТРЫХ КРИТИЧЕСКИХ СБОРКАХ ВЫСОКОПЛОТНОГО НИЗКООБОГАЩЕННОГО УРАН-ЦИРКОНИЕВОГО КАРБОНИТРИДНОГО ТОПЛИВА

Аннотация. Топливо UZrCN представляет собой высокоплотное высокотемпературное топливо, которое может применяться в реакторах различных типов. В прошлом реакторные испытания ВОУ (96% U-235) UZrCN-топлива были выполнены только с низким выгоранием. Вместе с тем данные реакторных испытаний необходимы при высоком выгорании для подтверждения оптимальных характеристик этого типа топлива. Высокопоточный исследовательский реактор СМ-3, расположенный в Государственном научном центре – Научно-исследовательский институт атомных реакторов (г. Димитровград, Россия), будет использоваться для испытания НОУ (19,73% U-235) UZrCN-топлива до ~40 % выгорания. Затем топливо будет исследоваться для определения его характеристик после облучения.

На критических стендах «Гиацинт» и «Кристал» в Объединенном институте энергетических и ядерных исследований – Сосны Национальной академии наук Беларуси (г. Минск, Беларусь) осуществляется подготовка к экспериментам по критичности на размножающих системах, моделирующих физические особенности активных зон с НОУ (19,75% U-235) UZrCN-топливом для использования в работах по новому поколению быстрых реакторов с газообразными и жидкометаллическими теплоносителями. Критические сборки представляют собой однородные гексагональные решетки топливных сборок, каждая из которых состоит из семи топливных стержней и не имеет оболочки. Длина активной части топливного стержня составляет 500 мм. Материал оболочки – нержавеющая сталь или ниобий. Будут исследованы три типа топливных сборок с различным материалом матрицы в них (воздух, алюминий и свинец). Боковой радиальный, верхние и нижние отражатели – бериллий (внутренний слой) и нержавеющая сталь (внешний слой).

В настоящей статье описываются проектные данные эксперимента, который будет осуществлен на реакторе СМ-3, и обсуждаются результаты расчетов, призванные показать, что эксперимент будет отвечать всем поставленным целям. Также представлены описания конструкции и состава критических сборок с топливом UZrCN и результаты их расчетов.

Ключевые слова: ядерное топливо, уран-циркониевый карбонитрид, реакторные и послереакторные испытания, критические сборки

Для цитирования. Реакторные и послереакторные испытания и исследования на быстрых критических сборках высокоплотного низкообогащенного уран-циркониевого карбонитридного топлива / С. Н. Сикорин [и др.] // Вес. Нац. акад. навук Беларусі. Сер. фіз.-мат. навук. – 2017. – № 4. – С. 104–112.

For many years, the USSR researchers have developed and tested a high density, high temperature uranium zirconium carbonitride UZrCN HEU fuel for potential application in different types of reactors. As part of this effort, reactor tests using UZrCN HEU (96% U-235) have been performed to low burnup of 0.6 % FIMA that show the fuel has optimal irradiation performance characteristics. In general, UZrCN fuel has better thermo-physical properties than UO_2 fuel. [1] UZrCN is a high uranium density fuel (that requires relatively low U enrichment) that can be employed at relatively high operating temperatures (≥2500 K). The zirconium in the fuel stabilizes the phase composition, and the carbon blocks relatively low temperature dissociation typical for UN. The fuel has a thermal conductivity almost 10 times higher, a strength limit almost 3 times higher, and volumetric swelling almost 3 times lower than UO_2; has high resistance to overheating during accidents (4 times higher than UO_2); has lower fission gas emissions and swelling compared to UN fuel; and, has a relatively smooth transition to a corium dioxide phase during extreme accidents. The main disadvantage for UZrCN is the limited amount of data on the irradiation performance of the fuel, particularly at high burnups. The optimal characteristics of UZrCN make it an attractive candidate for use in different types of reactors, including fast and high temperature gas-cooled reactors.

In order to produce the high-burnup data that is needed for demonstrating the optimal characteristics of UZrCN, a reactor test will be performed up to ~40% burnup (with some UZrCN pellets being removed after 5 and 15% burnup) in the high-flux SM-3 research reactor located in The State Scientific Center-Research Institute of Atomic Reactors, Dimitrovgrad, Russia. The fuel that will be tested has a composition of $U_{0.9}Zr_{0.1}C_{0.5}N_{0.5}$, a density of 12.5 g/cm^3, an enrichment of 19.73% (uranium-235), and an uranium density of 11.3 g/cm^3. Over 1200 effective days of irradiation will be required to achieve the targeted burnup.

It is planned to use the “Giacint” and “Kristal” critical facilities located in The Joint Institute for Power and Nuclear Research – Sosny of the National Academy of Sciences of Belarus (Minsk, Belarus) to carry experiments at critical assemblies on fast neutrons, simulating physical features of the cores of fast reactors, cooled by gas and liquid-metal coolants.

This article will describe the design of the experiment and will discuss the results of performed calculations that show a planned reactor test in the SM-3 research reactor will meet all the testing objectives. Also describes the design and the composition of the critical assemblies, the results of calculation of K_{eff}, and the program of experiments at the critical assemblies.

1. In-pile and out-of-pile tests. UZrCN LEU fuel at the SM-3 research reactor. Experiment design. Experimental capsules have been designed to allow for reactor testing of UZrCN pellets to 5, 15, and 40% burnup contained in an irradiation device. An experimental capsule is a leak-tight cylindrical canister with 2-mm-thick walls that is welded at the top and bottom with plugs. The capsule will be made of W or possibly a W alloy (e.g., W-4Ta). There will be a 300 μm gap between the UZrCN pellet

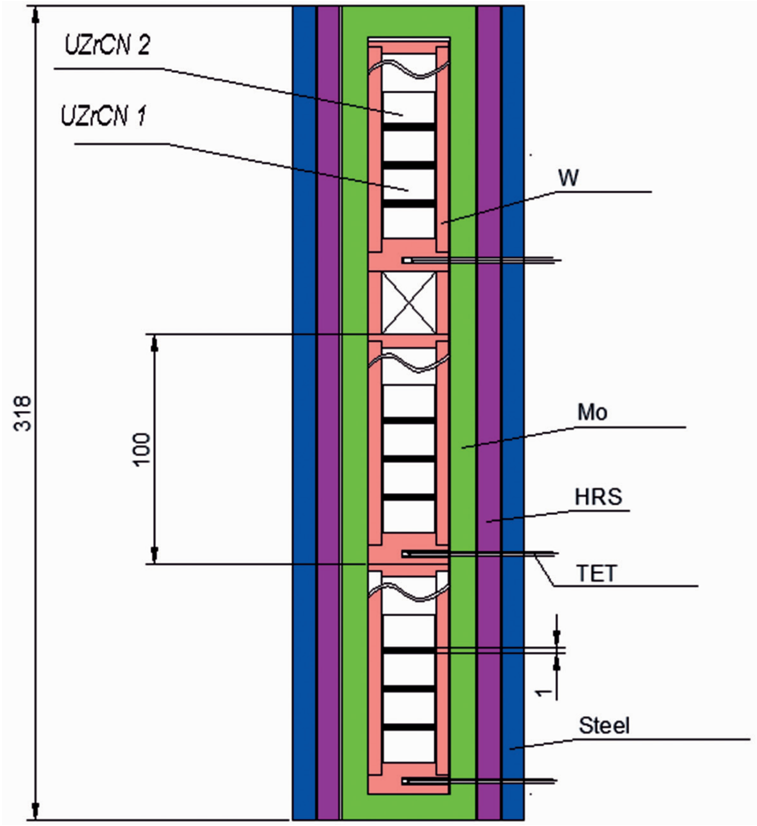

Fig. 1. Schematic diagram of the irradiation device

and the capsule wall, and the capsule will be filled with pure He (with an excess pressure of ~0.01 MPa). Each CNF pellet will be ~100 mm long and four pellets will be contained in a capsule.

The irradiation device, as shown in Fig. 1, consists of three experimental capsules Mo shells with 4-mm-thick walls contained in protective heat resistant steel (EP 912-VD or stainless steel X12H10T). This experimental assembly is held in X12H10T steel. The Mo shell has grooves for six thermocouples. The steel casing will be exposed to water at a temperature between 50 and 70 °C during the irradiation. The irradiation device also has an external neutron absorbing Hf screen to equalize the energy released during the irradiation test. It is planned to perform the reactor tests of the experimental capsules in the designed irradiation devices in positions 10 and 11 of the second reflector row of the research reactor SM-3 (Fig. 2).

Supporting Calculations. The neutron physics conditions for the experiments were calculated using the MCU-RR computer program, and the geometric parameters that were employed are listed in Tab. 1.

Table 1. **Geometric parameters of the experimental capsule and irradiation device**

Designation	Beginning Campaign		End of Campaign	
	Scheme 1	Scheme 2	Scheme 1	Scheme 2
Gap between CNF pellet and cladding (microns)	300	300	150	150
Dimensions of W cladding (mm)	12.6×2.0	12.6×2.0	12.6×2.0	12.6×2.0
Gap between cladding and cover (microns)	100	100	100	100
Dimensions of Mo cover (mm)	22.0×4.6	20.6×4.0	22.0×4.6	20.6×4.0
Gap between Mo cover and protective case (microns)	500/300	100	500/300	100
Dimensions of protective case (mm)	34.0×5.5/33.6×5.5	28.0×3.5/28.1×3.5	34.0×5.5/33.6×5.5	28.0×3.5/28.1×3.5
Gap between protective case and steel case (microns)	500/300	300/350	500/300	300/350
Dimensions of Irradiator Unit steel case (mm)	42.0×3.5	35.6×3.5	42.0×3.5	35.6×3.5
Thickness of experimental capsule (mm)	5	5	5	5
Height of experimental capsule (mm)	100	100	100	100
Height of Irradiator Unit (mm)	318	318	318	318

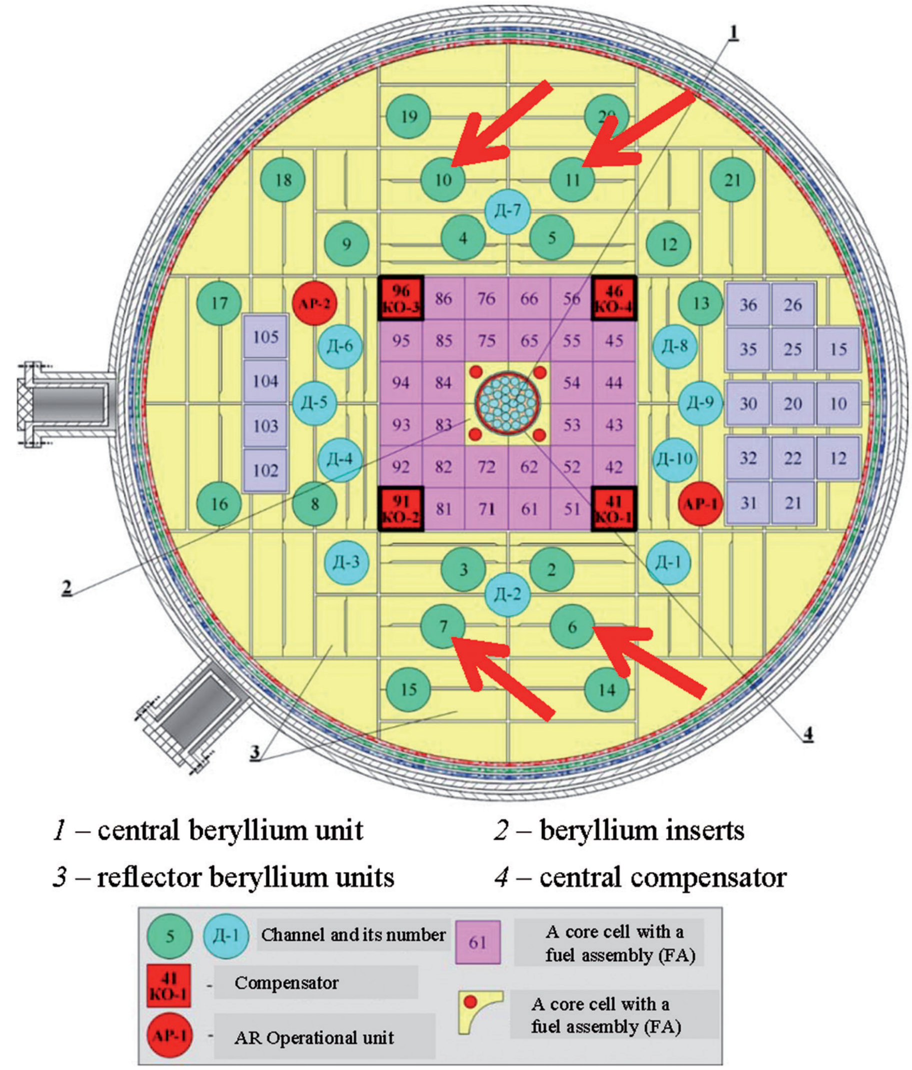

Fig. 2. Diagram showing a cross-section of the SM-3 research reactor core

The program implemented an algorithm for solution of the neutron transfer equation by the Monte Carlo method. The goal of the neutron physics calculations was to determine geometric characteristics of the absorbing hafnium screen, ensuring maximum energy release in UZrCN fuel (up to 500 W/cm^3) in the irradiation device with a heat-resistant steel capsule. The calculations considered the conditions of the core, correspondent to the midpoint and the end of the campaign. The neutron physics calculation results were used to determine the outer diameter of the hafnium screen equaling 59.5 mm, and the wall thickness equaling 2.5 mm. The screen dimensions were selected to ensure heat release with the first circuit washed with water in a uniform manner.

Based on the calculated values of the neutron flux density in the experimental capsules for the SM-3 reactor operating at 90 MW, the duration of the tests was estimated. To reach 40% U-235 burnup, around 25 or 26 SM-3 campaigns are needed (more than 1200 days). Based on the calculated value of energy release for the UZrCN pellets, it was shown that the target energy release of 500 W/cm^3 could be met by slightly increasing the thickness of the Hf screen.

During irradiation of the UZrCN, the thermal physical properties of the gaseous medium in the gap between the fuel and the ampoule cladding will change. The egress of volatile fission products from the fuel (mainly, Xe^{133} and Kr^{89}) results in the formation of a gaseous mixture He–Xe–Kr in the gap between the fuel and the cladding, whose thermal physical properties deteriorate compared to the original properties of helium specified during the design of thermal physics calculations. A mixture of He–Ne, simulating the mixture of gaseous fission products is fed to the original gas gaps in order to maintain the UZrCN and ampoule temperature at a quasi-stationary level during the irradiation. Nitrogen released due to dissociation of UZrCN at elevated temperatures is also considered.

To aid in the selection of the optimum conditions for testing of the UZrCN in the reactor, it was necessary to predict thermal physical properties of gaseous media in the irradiation device depending on the content of individual components and the temperature. It was determined that during the irradiation the heat conductivity coefficient of the mixture He–Xe–Kr varies in a range from 0.10 W/(m K) to 0.46 W/(m K) with the content up to 0.6 relative fractions of Xe–Kr and the temperature 1500 K. The broad range relationship of the fission products changes the heat conductivity coefficient of the mixture He–Xe–Kr insignificantly.

The stable heat conditions of the experiment capsules and irradiation devices are ensured by varying the compositions of the gas media used to fill the ampoule and capsule gaps. The predicted temperatures for the UZrCN pellets and claddings were used to determine the gas compositions in the gaps. These temperature calculations were performed using the codes ANSYS, COMSOL, and PARAM-TG. The calculations were made with iterations, taking into account changes in the model geometry due to heat expansion, given the dependence of thermal physical properties of solids, liquids and gases on temperature. The thermal condition of the fuel section was simulated, taking into consideration variations of the UZrCN fuel geometry and fission product gas egress. The swelling of UZrCN pellets and the composition of gaseous media are taken as constant for the calculations. The calculations were made for two core conditions, correspondent to the beginning and end of the campaign, reaching 40% burnup of fissionable atoms. The calculations were made for three arrangements of the UZrCN pellets and spacers.

At the beginning of life, a 60% He–40% Ne mixture is used for the capsule that is removed after 5% burnup. For the capsule removed at 15% burnup, a 90% He–10% Ne mixture is used. For the capsule irradiated to 40% burnup, a 100% He gas composition is used. For the 40% burnup case, estimations show the final composition of gaseous mixture under the experimental capsule cladding is 60% (Xe+Kr)–40% He (mole). It was assumed that the fission product gases were fully egressed from the UZrCN pellets. Given the mechanisms of gaseous fission product migration and diffusion from the UZrCN, it is predicted that the share of stable Xe and Kr isotopes in the gaseous mixture will decrease to ~ 20% (mole).

2. The program of experiments on the fast multiplying systems with UZrCN LEU fuel at the critical facilities “Giacint” and “Kristal”. The critical facilities “Giacint” and “Kristal” are used to prepare the experiments on the criticality of multiplying systems simulating physical features of the core of the future fast-neutron reactors with gaseous and liquid-metal coolants. Lattices of fuel cassettes in a matrix from air, aluminium and lead were investigated.

This fast critical assembly represents a lattice (39 mm pitch) of fuel cassettes with fuel rods based on UZrCN (19.75% U-235) and with air or lead or aluminum as matrix materials, with a beryllium-steel reflector. The critical assemblies included a core, a side reflector, top and bottom axial reflectors and a control and protection system’s (CPS) rods.

The cores of critical assembly, comprising fuel cassettes, is surrounded by several rows of Be and steel reflector units. These elements of the critical assemblies are placed on the stainless-steel support grid. The neutron detectors are attached on special poles around the critical assemblies.

There are six types the fuel cassettes. The fuel cassette type 1 (type 2) with air as matrix materials is unclad and comprises 7 fuel rods of type 1 (7 fuel rods of type 2) and end pieces (Fig. 3). The fuel rods are placed in a cassette with a 14 mm pitch over the hexagonal grid and fixed by the end pieces. The turnkey size of the design model of the cassette is 38.5 mm; the total length of the design model of the cassette is 912 mm. All upper and lower end pieces of the cassettes are made from stainless steel.

The fuel cassette type 3 (type 4) with lead as matrix materials is unclad and comprises 7 fuel rods of type 1 (7 fuel rods of type 2) and end pieces (Fig. 4). The fuel rods are placed in a cassette with a 14 mm pitch over the hexagonal grid and fixed by the end pieces. The lower end piece of the cassette comprises a shank (stainless steel), a lower end piece and the lower tube plate (stainless steel); the upper end piece of the cassette comprises the upper tube plate (stainless steel), the lead end piece and the head (stainless steel). The turn-key size of the design model of the cassette is 38.5 mm; the total length of the design model of the cassette is 912 mm.

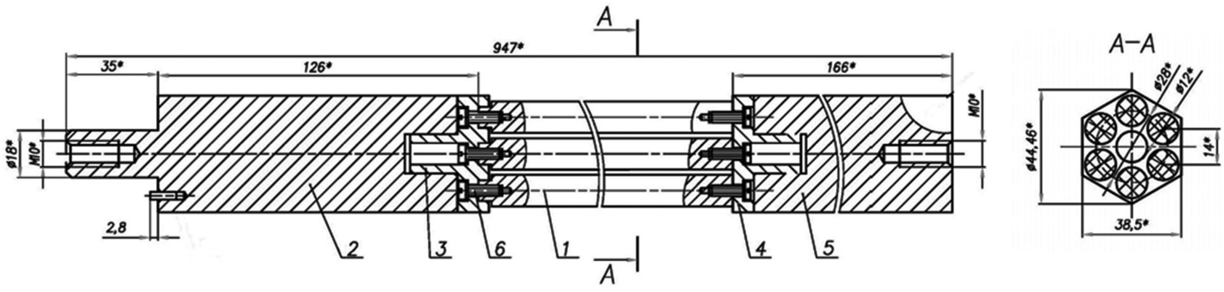

Fig. 3. The design model of fuel cassette type 1 (type 2):
1 – fuel rod of type 1 (type 2); *2* – shank; *3* – lower tube plate; *4* – upper tube plate; *5* – head; *6* – screw

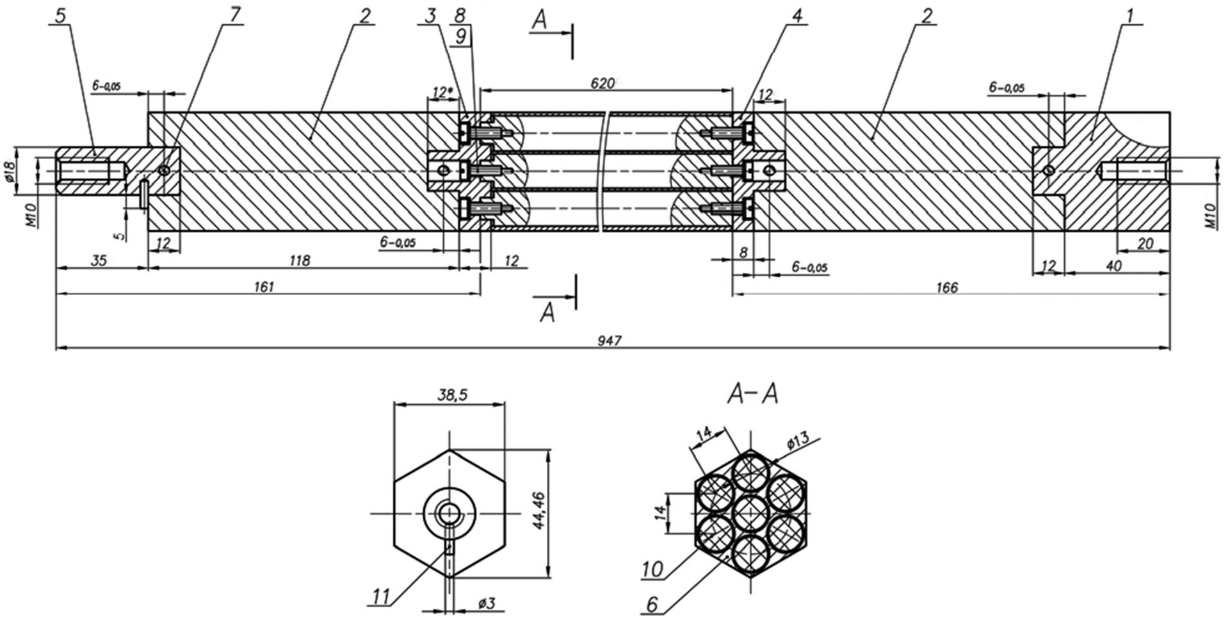

Fig 4. The design model of fuel cassette type 3 (type 4):
1 – upper steel piece; *2* – lead prism; *3* – lower tube plate; *4* – upper tube plate; *5* – lower end piece; *6* – lead matrix; *7* – pin; *8* – washer; *9* – pin; *10* – screw; *11* – the fuel rods of the type 1 or type 2

The fuel cassette type 5 (type 6) with aluminum as matrix materials is unclad and comprises 7 fuel rods of type 1 (7 fuel rods of type 2) and end pieces (Fig. 5). The fuel rods are placed in a cassette with a 14 mm pitch over the hexagonal grid and fixed by the end pieces. The turn-key size of the design model of the cassette is 38.5 mm; the total length of the design model of the cassette is 912 mm. All upper and lower end pieces of the cassettes are made from stainless steel.

The fuel rod type 1 (type 2) comprises a fuel core, cladding, and end pieces (Fig 6). The fuel rod cladding has the outer diameter 12 mm and the wall thickness 0.6 mm. The fuel rod comprises tablets, 10.75 mm in diameter and 14.7 mm in height, from uranium-zirconium carbonitride $U_{0,9}Zr_{0,1}C_{0,5}N_{0,5}$. The enrichment by U-235 was 19.75%. The gaps between the fuel rod cladding tablets and the fuel rod cladding contain He under ~0,11 MPa. The total core height is 500 mm. The total length of the fuel rod is 620 mm. The material of the fuel rod clad and end pieces (plugs) is stainless steel (fuel rod type 1) or alloy NbZr-1 (fuel rod type 2).

The side reflector of the critical assemblies is several rows of the beryllium and stainless steel reflector units. The beryllium reflector unit represents a hexagonal beryllium prism with turn-key size 38.5 mm and the length 872 mm. To the bottom of the unit is attached to the shank. The head (stainless steel) with the turn-key size 38.5 mm and the length 40 mm is fixed to the upper part of the unit. The total length of the unit is 947 mm, and the shank length is 35 mm. The steel reflector unit represents a hexagonal prism from stainless steel with the turn-key size 38.5 mm. To the bottom of the unit is attached to the shank. The total length of the unit is 947 mm, 35 mm shank length.

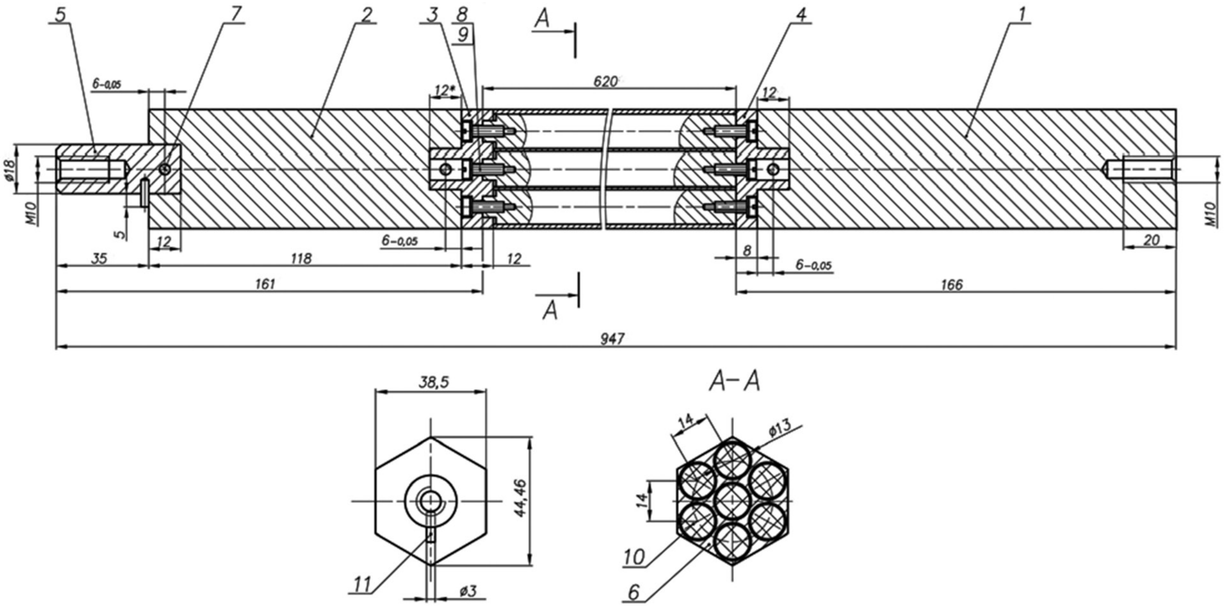

Fig. 5. The design model of fuel cassette type 5 (type 6):
1 – upper steel prism; *2* – lower steel prism; *3* – lower tube plate; *4* – upper tube plate; *5* – lower end piece;
6 – aluminum matrix; *7* – pin; *8* – washer; *9* – pin; *10* – screw; *11* – the fuel rods of the type 1 (type 2)

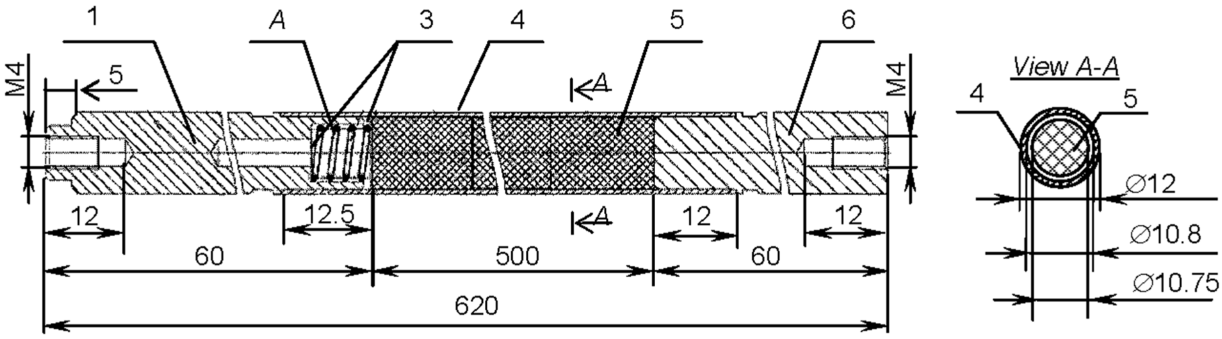

Fig. 6. Fuel rod type 1 (type 2):
1 – lower plug; *2* – spring; *3* – gaskets; *4* – cladding; *5* – fuel core; *6* – upper plug

The experiments are to be performed on three fast critical assemblies. The core and reflector compositions of these critical assemblies are presented in Tab. 2. The calculation results of K_{eff} made by the Monte Carlo method using the MCNP-4C [2] and MCU-PD [3] computation codes are presented in Tab. 3. Fig. 7 represent the loading chart of the fast critical assemblies.

Table 2. **The core and reflector compositions of the fast critical assemblies**

The critical assembly	The fuel cassette, pcs						Beryllium reflector unit, pcs	Steel reflector unit, pcs
	type 1	type 2	type 3	type 4	type 5	type 6		
Type 1	210	7	–	–	–	–	504	306
Type 2	–	–	210	7	–	–	504	306
Type 3	–	–	–	–	210	7	504	306

Table 3. **The calculation results of the fast critical assemblies**

The critical assembly	K_{eff} calculation result		β_{eff} calculation result
	MCNP-4C	MCU-PD	MCU-PD
Type 1	1,00494 ± 0,00021	1,00993 ± 0,00030	0,007408
Type 2	1,01672 ± 0,00021	1,01244 ± 0,00027	0,007362
Type 3	1,01412 ± 0,00020	1,01619 ± 0,00037	0,007367

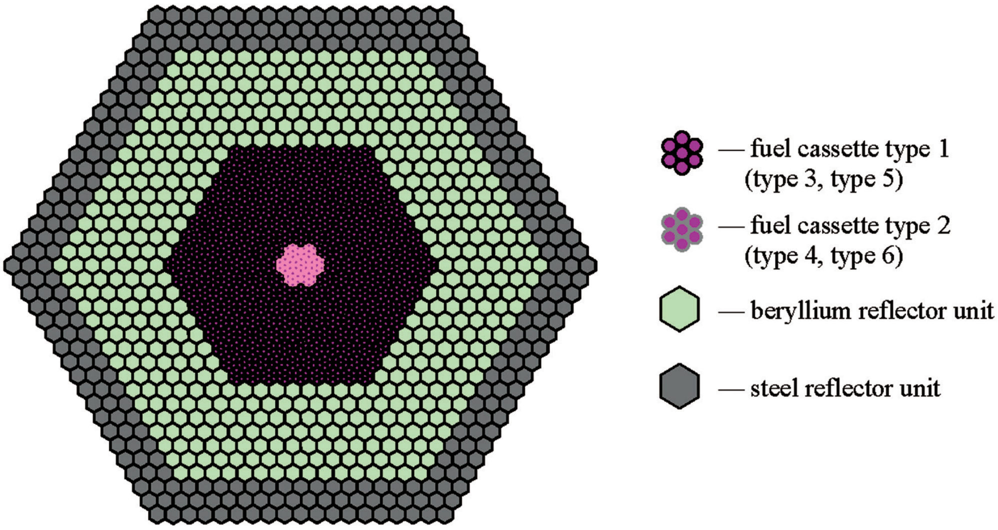

Fig. 7. Loading chart of the critical assembly type 1 (type 2, type 3)

Conclusions. This article has described the design of an experiment using UZrCN pellets that will be performed in the SM-3 research reactor up to a burnup of 40 %. Calculations have been performed that indicate the irradiation experiment will be accomplished successfully, and all testing goals will be met. Initial gas compositions have been identified for use in technological gaps that will maintain target temperatures, and the final gas compositions after different levels of irradiation have been determined.

It is planned to use the "Giacint" and "Kristal" critical facilities to carry experiments at critical assemblies on fast neutrons, simulating physical features of the cores of fast reactors, cooled by gas and liquid-metal coolants.

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