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DEVELOPMENT AND JUSTIFICATION OF FUEL EFFICIENCY FOR SUBCRITICAL INSTALLATION DRIVEN BY ELECTRON ACCELERATOR

В.С. Красноруцький, М.М. Бєлаш, Й. Гохар, А.М. Абдуллаєв, А.В. Куштим, С.О. Солдатов, **Розробка і обґрунтування працездатності палива для підкритичної установки, керованої прискорювачем електронів.** Представлено варіанти конструкції та технологічні особливості виготовлення паливних збірок для підкритичної ядерної установки «Джерело нейтронів», керованої прискорювачем електронів, з тепловидільними елементами стрижневого типу. Розроблене паливо є альтернативним паливу ВВР-М2, яке виробляється і поставляється ВАТ НЗХК (Росія) для вітчизняних дослідницьких ядерних установок. Приведено результати дореакторних випробувань, нейтронно-фізичних та тепло-гідравлічних розрахунків в обґрунтування надійності і безпечності використання розробленого палива в підкритичній установці.
Ключові слова: підкритична установка, ядерне паливо, диоксид урану, металлокерамічна композиція, працездатність, розрахункове обґрунтування

V.S. Krasnorutskii, M.M. Belash, Y. Gohar, A.M. Abdullaev, A.V. Kushtym, S.O. Soldatov,. **Development and justification of fuel efficiency for subcritical installation driven by electron accelerator.** The variants of design and technological features of manufacturing of fuel assemblies for the subcritical nuclear installation “Neutron Source” controlled by an electron accelerator, with rod-type fuel elements are presented. Developed fuel is an alternative for VVR-M2 fuel, produced and supplied by PJSC NCCP (Russia) to Ukrainian nuclear research installations. The results of pre-reactor tests, neutronic and thermo-hydraulic calculations are shown to substantiate reliability and safety of using developed fuel in a subcritical installation.
Keywords: subcritical installation, nuclear fuel, uranium dioxide, metal ceramic composition, efficiency, design-basis justification

Introduction. Nuclear Research Installations (NRIs), despite the systematic reduction of their quantity, still play an important role in the development of nuclear energetics and in other fields of science and technology. They are successfully used for carrying out a broad program of fundamental research in nuclear physics, condensed matter physics, radiation material science, radiobiology, for the development of radionuclides for medical purposes [1, 2]. In addition, applied research on the NRIs provides justification for the safety of newly developed nuclear power plants.

Most developed nuclear countries seek to build nuclear research facilities or actively participate in joint projects for their construction and use for research purposes. Much attention in the world is given to the NRI controlled by accelerators of charged particles. One of such installations is the “Neutron Source” built in the NSC KIPT (Kharkov). It represents a new type of nuclear installation in which the intensity of the nuclear reaction of the ^{235}U isotope fission in the reactor core (RC) is controlled by an electron accelerator. Caused by the accelerator electron flow in the target installation, primary neutrons are reproduced in the active zone. According to international classification of the IAEA such installations are the ADS-systems (Accelerator Driven Systems). The geometry of the installation loading and the mass of the fissile material are chosen in such a way that the effective neutron multiplication factor K_{eff} remained less than unity (K_{eff} <1), no matter what the initial events were, that is, the subcritical installation (SI) is the multiplication environment. Such a solution guaran-

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tees nuclear safety of the NRI. This is the essence of the fundamental difference between the developed NRI and research nuclear reactors that operate in self-sustaining chain reaction mode [3].

The purpose of this work is development and justification of the fuel efficiency for the subcritical installation "Neutron Source".

1. Design and characteristics of fuel for a subcritical installation. Neutron-physical characteristics of SI with BBP-M2 fuel are given in Table 1.

Table 1

Characteristic of the reactor core of the SI under routine operation [3]

Parameter	Value
Reactor core	
Maximum effective neutron multiplication factor $K_{\text{eff}}^{\max}$	≤ 0,98
Type of FA	VVR-M2
Uranium enrichment rate by isotope ^{235}U, %	19.7
Number of FA in reactor core, pcs.	36
Maximum capacity of heat generation in *RC*, kW	≤ 260
Maximum full heat generation in *SI*, kW	≤ 360
Coolant / moderator	
Material of coolant/moderator	deionized H_2O
Inlet temperature of coolant/moderator, °C	25
Outlet temperature of coolant/moderator, °C	less than 30
Coolant/moderator pressure, MPa	0,1
Loss of coolant/moderator, m^3/g	58
Neutronic characteristics	
Maximum neutron flux density in *RC*, $n/(cm^2 \cdot s)$	$2.4 \cdot 10^{13}$
Maximum neutron flux density on inner boundary of annular reflector, $n/(cm^2 \cdot s)$	$2 \cdot 10^{13}$
Maximum neutron flux density in RC in spectral interval $E_n \le 0{,}1$ MeV, $n/(cm^2 \cdot s)$	$1.5 \cdot 10^{13}$
Maximum fast neutron flux density ($E_n > 0{,}1$ MeV) in *RC*, $n/(cm^2 \cdot s)$	$1.3 \cdot 10^{13}$
Maximum fluence of neutrons on outer boundary of annular reflector, $n/(cm^2 \cdot s)$	$1.6 \cdot 10^{12}$

Its reactor core is composed from VVR-M2 fuel assemblies. The serial fuel assembly of the VVR-M2 reactor is a collection of three fuel tubes, placed one inside the other [4].

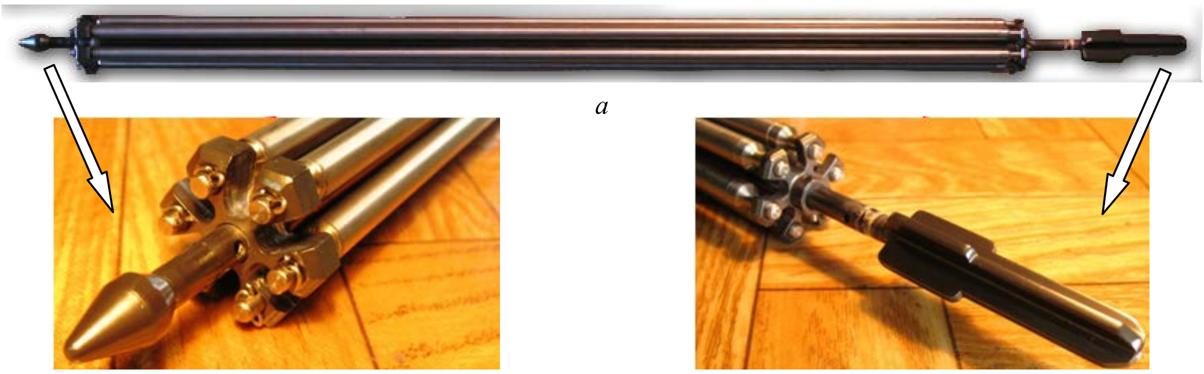
The lower and upper ends of the fuel elements are connected to the head and tail piece. The head and tail piece along the perimeter have the shape of a hexagon with a "ready-to-operate" size of 35 mm. The fuel element, which is located on the outside, has the shape of a hexagonal tube with a "ready-to-operate" size of 32 mm. Inside it, on one of its axis, there are two cylindrical tubular fuel elements with 22 mm and 11 mm diameters on the outer surface.

The length of fuel portion in nuclear fuel elements is equal 500^{+20}_{-30}. The fuel enrichment in ^{235}U is 19.7 ± 3 %of the weight, and the concentration of UO_2 is 2.53 g/cm^3. A layer of metalloceramic fuel (UO_2–Al) 1.0 mm thick is placed inside the walls of each tubular fuel element. The construction material of fuel elements, head and tail piece is aluminum alloy SAV-1.

Fuel assemblies of the VVR-M2 type have successfully proved themselves as a result of long-term exploitation in research reactors, but the technological process of their production is complex and difficult to automate, and is based on the use of expensive and bulky equipment. Therefore, when choosing alternatives for the design of alternative fuels, taking into account global trends, the fuel element frame construction was kept in mind, which is simpler and more technological in production.

In the course of designing the TVS-X fuel assembly, one of the main requirements was to ensure its compatibility with the base assembly VVR-M2 in main operational characteristics: neutronic, thermo-hydraulic, overall dimensions and the form of places of installation of fuel assemblies in the reactor core (*RC*) of *SI*.

The appearance of the fuel assembly TVS-X manufactured according to the developed technological process, is shown in Fig. 1.



а

б *в*

Fig. 1. Appearance of TVS-X: assembled TVS-X (a); upper part (head, fitting, upper grid, fuel elements) (b); lower part (tail piece, stub tube, lower grid, fuel elements) (c)

2. Neutronic calculation of fuel enrichment. Modelling of propagation properties of the TVS-X fuel in the active zone of SI was carried out using the MCNPX^{TM} computer code (Monte Carlo *N*-Particle code extended). This program is designed to calculate the radiation transport using the Monte Carlo method. It allows tracing the trajectories of neutral and charged particles in a wide range of energies. At transporting particles, their interaction with a substance is randomly determined on the basis of the possible types of nuclear reactions inherent to the transported particle and to the isotopes that are part of the substance. If secondary particles are generated in the process of interaction, then their tracing is also considered. For neutrons, the program uses libraries of intersections of energies of 20 MeV ENDF/B-VI types, which include almost all stable isotopes. The SCALE5 code system, which uses the ORIGEN ARP function module and allows us to estimate the change in the isotopic composition of fuel during its burn-out, was used to analyze the modes of regular operation.

The geometry of the designed task is determined by describing the geometric shapes and volumes (cells) that are bounded by them. This description is the geometric part of the input task file for the MCNPX code. In the input file, the materials that fill the cells are also specified, and the type and characteristics of the source of the primary particles are determined.

In order to calculate the neutron multiplication factor in SI with TVS-X the basic configuration of the reactor core consisting of 36 cassettes (Fig. 2) covered by the reflector was considered. Fuel assemblies and reflector were placed in an aluminum tank filled with cooling water (Fig. 2, *b*). The calculation with the dividing source was made for 1000 active generations of neutrons with 20.000 neutrons in each generation. This amount of statistics provides a statistical error of calculation not worse than $\sim 2 \cdot 10^{-4}$. At the boundary of the calculation model vacuum boundary conditions were established.

The results of calculating the neutron multiplication factor of SI for UO_2 fuel pellets, depending on the enrichment, are given in Fig. 3. The obtained dependence is well described by the linear equation $Y = 0,04147 \times X + 0,79175$, where X is the fuel enrichment in weight %. The analysis of the given data shows that when the fuel enrichment is 4.525% of ^{235}U weight, the value $K_{eff} = 0.97950 \pm 0.00013$ is very close to the rated value of 0.98. Any attempts to determine the enrichment at which K_{eff} will be exactly 0.98, showed that in reality, taking into account technological tolerances, it can reach a value greater than 0.98, so it is necessary to have some margin in the neutron multiplication factor that would compensate these technological tolerances. Thus, the maximum value of fuel enrichment with the use of UO_2 pellets should be chosen for reasons of not exceeding the value of $K_{eff} = 0.98$ at maximum tolerances.

The data shown in Fig. 3 indicates that the amount of enrichment of fuel pellets should not exceed 4.44 % of ^{235}U weight. From an economic point of view, it is most appropriate to use pellets enriched by 4.4 % of ^{235}U weight that are manufactured and used on an industrial scale, for example, to feed the reactor core of WWER-1000 reactor [5].

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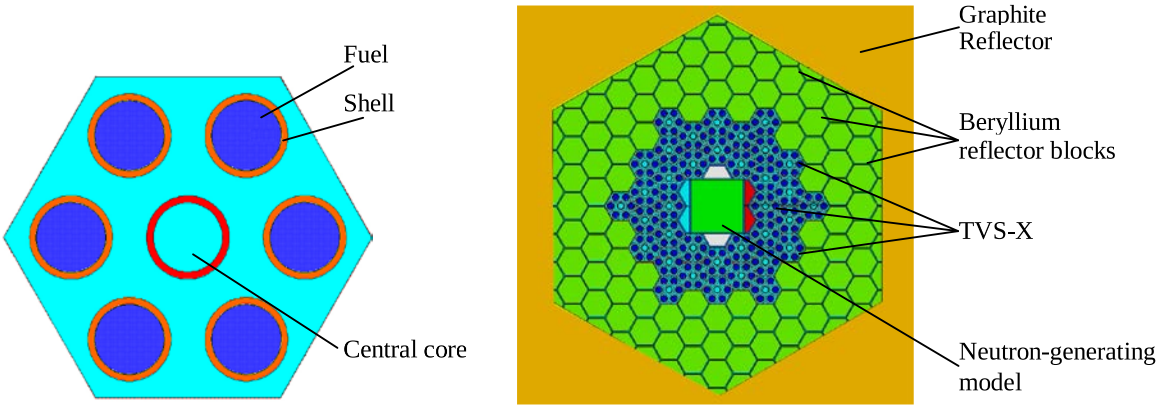


a *b*

Fig. 2. Design model for determining fuel pellet enrichment: TVS-X (a); SI (b)

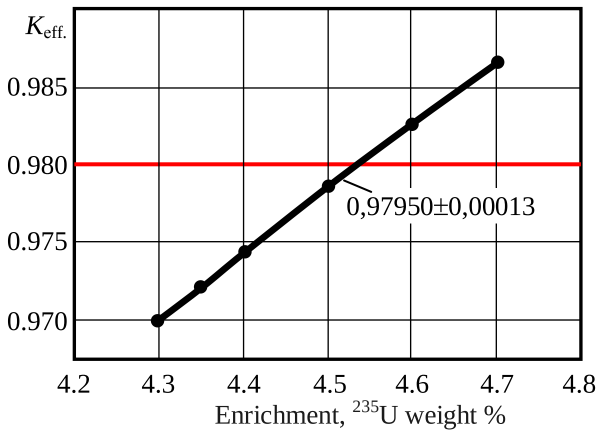


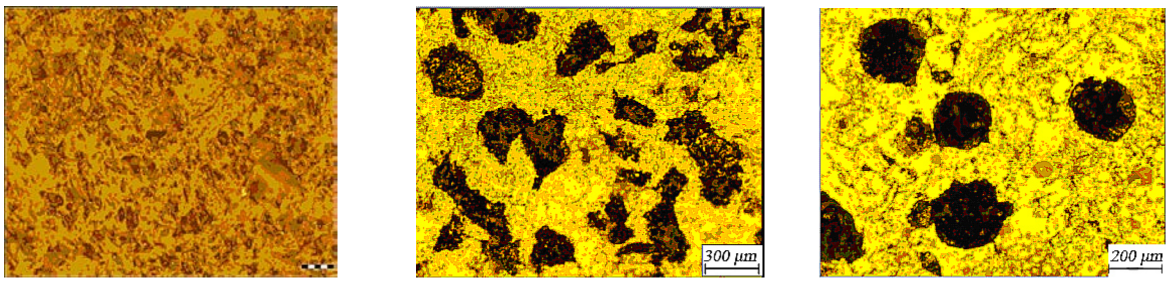
Fig. 3. Dependence of $K_{eff.}$ of SI on fuel enrichment using UO_2 pellets

When using the UO_2–Al dispersion composition pellets in the TVS-X fuel elements, it was found that the content of the UO_2 grit enriched by 19.97 % of ^{235}U weight in the metal-ceramic composition should not exceed 1.78 g/cm^3 (17.1 % vol.). In this case K_{eff}<0.97980± 0.00018.

If we compare the obtained data with the characteristics of VVR-M2 fuel, where the content of UO_2 is 2.53 g/cm^3 (~ 24 % vol.), then it can be stated that in the case of the use of metal-ceramic fuel in TVS-X fuel elements, the same energy output per volume unit is achieved at a lower concentration of ^{235}U in a fuel pellet, and, accordingly, a smaller amount of aluminum matrix in the process of radiation in a nuclear installation will be damaged.

3. Justification of efficiency of the TVS-X.

3.1. Results of materials research. Based on the developed technological processes, UO_2 fuel pellets and UO_2–Al metal-ceramic composition were produced and were used in the prototype of TVS-X fuel elements [6–8]. Their structure is shown in Fig. 4. UO_2 pellets have the density of 10.3...10.5 g/cm^3. The grain size of the pellet material is 2...7 μm (Fig. 4, *a*), the oxygen coefficient is 1.97, the microhardness is 6...7 hPa.



a *b* *c*

Fig. 4. Structure of fuel pellets material: UO_2 pellet (a); UO_2–Al pellet with grit particles (b); UO_2–Al pellet with microsphere particles (c)

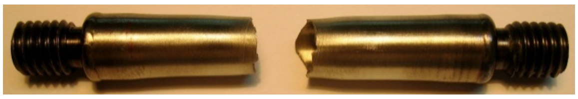
In the dispersion composition tablets, UO_2 grits and microspheres are used as fuel particles. Their characteristics are given in Table 2. The density of UO_2–Al pellets is 0.96…0.98 of the calculated maximum value. Samples of fuel pellets have a fairly even distribution of fuel particles in the matrix (Fig. 4 *b, c*). Visible traces of interaction of fuel particles material with the matrix in the process of pellets manufacturing were not found. The matrix material has a fine-grained structure with an average grain size of ~25 μm. Small pores (up to 10 microns) are observed on the grain boundaries.

Table 2

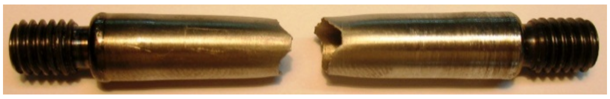
Characteristic of uranium dioxide particles

Material of particles	Shape of particles	Range of sizes, μm	Density, g/cm^3	Method of production
UO_2	microspheres	200…400	9.8	mechanical spheroidizing
UO_2	grits	112…315	10.4	mechanical grinding

Examination of the material of the welded joints shows that the melting of the material of the fuel element's shell is made on its entire thickness, the pores and shells are absent. The structure of the weld seams material is a typical metastable α′-phase Zr with a deformed high-density solid state grid formed from a β- solid solution at cooling [9]. Compared to the initial state, when the grain size is equal to 7...20 μm, a martensitic-type structure of the former β-grains of 150...500 μm in size is formed in the material of the weld seam with a characteristic internal structure in the form of martensitic type plates of 3...8 μm in width and 20...50 μm in length. The results of the mechanical tests of the fuel elements models confirm the high strength of the welded joints. In the process of testing at temperatures of 20 °C and 100 °C the destruction occurs on the shell in its central part (Fig. 5).



a



b

Fig. 5. Destruction types of fuel elements models during short-time mechanical tests at temperatures: 20 °C (a) and 100 °C (b)

The values of the margin of strength, plasticity and relative elongation at a temperature of 20 °C are 377±25 MPa, 310±18 MPa and 31±3% respectively, and at a temperature of 100 °C, correspondingly, equal to 346±30 MPa, 237±20 MPa and 31±2 %. The autoclave test of the models in the medium of bidistilled water at a temperature of 100 °C with a duration of ~5000 g also confirmed their high efficiency. All samples have retained consistency and tightness.

3.2. Calculated justification of TVS-X efficiency at standard operating conditions. The calculation of the performance characteristics of TVS-X at standard operating conditions was carried out using programs, verified on the basis of experimental data, obtained on different types of similar fuel elements during their operation in PWR and WWER reactors. SolidWorks Flow Simulation program was used to calculate the thermo-hydraulic characteristics, and the TRANSURANUS code (v1m1i09) was used for thermophysical, corrosion, deformation and strength parameters (Calculated analysis using the TRANSURANUS code (v1m1i09) was used during the period of the license agreement under the SLA № 31302). The results of calculations are given in Table 3.

The given data illustrates that in the developed designs of fuel elements and TVS-X within the design lifetime of 43.800 effective hours the integral consistency and efficiency of all elements of a construction with a high safety factor is provided.

3.3. Justification of efficiency of TVS-X at emergency processes. When substantiating the efficiency of TVS-X in the conditions of occurrence of emergency processes, the following criteria were applied:

– fuel temperature should not exceed 660 °C (melting point of the most fusible construction material of the reactor core – aluminum);

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– non-exceedance of 0.98 value for Keff. for SI at standard operating conditions, in case of violation of the standard conditions and in case of project accidents;

– non-exceedance of 0.95 value for Keff for systems of storage and treatment of fresh and discharged fuel at standard operating conditions, in case of violation of the standard conditions and in case of project accidents;

– subcriticality of SI in all conditions of stopping not less than 5 %.

Table 3

Results of calculated justification of TVS-X efficiency and reliability

Efficiency criterion	Method used for justification	Achieved result
Thermo-hydraulic calculation of reactor core	SolidWorks Flow Simulation (COSMOSFloWorks)	If the coolant temperature at the inlet equals 25 °C, the temperature of the shell of fuel element is 40.7 °C. The coefficient of hydraulic resistance (CGO) of TVS-X is similar to that of VVR-M2, which indicates the satisfactory compatibility of these assemblies when working with software.
Temperature calculation	TRANSURANUS (v1m1i09)	Temperature in the center of the fuel pallet increases with burning and reaches the maximum values at the end of the campaign at maximum power output: – ~150°C using UO_2 pellets; – ~80 °C using developed dispersion composition UO_2–Al pellets.
Gas clearance between fuel and shell	TRANSURANUS (v1m1i09)	At the maximum burn-up range of fuel element, the minimum value of diametric gap in the hot condition is 130 μm. According to this calculation for all time of operation contact interaction between fuel pellets and the shell is absent.
Shell deformation	TRANSURANUS (v1m1i09)	At the burn-up range of 10.5 MW×days/kg U the effective deformation of the shell is 0.0015 %, which is ~3 orders of magnitude less than the adopted design value (1 %) for the fuel elements of PWR and WWER reactors.
Exit of gas fission products (GFP)	TRANSURANUS (v1m1i09)	At the fuel burn-up of 10.5 MW×days/kg U the accumulation of GFP (Xe + Kr) is 67.1 cm^3 and helium is 0.04 cm^3. In this case, the concentration of GFP under the shell is 0.06...0.15 %.
Thickness of oxide film	TRANSURANUS (v1m1i09)	Thickness of the oxide film, which is additionally formed as a result of working in the SI on the outer surface of the shell, is negligible and at the end of the campaign does not exceed 1.5 μm

For the analysis of the initial events associated with the deterioration of the heat transfer in the SI, the code RELAP5/Mod.3.2 was used, which allows to simulate a wide range of transition processes in light water reactor systems. In Table 4 and 5 the results of the calculation analysis performed for the reactor core of the SI of loaded TVS-X are represented.

The results of calculations have shown that at passing transitional modes induced by analyzed exhaust-gas events there is no violation of certain eligibility criteria.

As can be seen from the results summarized in Table 5, the removal of neutron-generating targets of both types (uranium and tungsten) in the absence of absorber rods should be eliminated by the presence of appropriate blockages and the introduction of administrative measures. When the uranium target is flooded, the acceptance criterion is violated; but the system remains subcritical, that is, the acceptance criterion for design failures is fulfilled: K_{eff} <1. For other considered initial events, violations of certain acceptance criteria do not occur.

Table 4

Summary table of fulfillment of acceptance criteria, established for the analysis of group of initial events, connected with heat removal deteriorating in subcritical assembly

Initial event	Maximum fuel temperature, °C	Maximum clad temperature, °C	Acceptance crite?-rion, °C
Loss of coolant flow in reactor core	206.9	117.0	660
Loss of heat removal in coolant contour of SI	192.4	100.7	
Loss of coolant in reactor core	250.4	248.9	

Table 5

Summary table of fulfillment of acceptance criteria, established for the analysis of initial events, connected with introduction of positive reactivity

Initial event	Maximum value of $K_{eff.}$	Acceptance criterion
Removal of neutron-generating target	1.00731 ± 0.00017	0.98
Flooding the casing of neutron-generating target with water	0.98819 ± 0.00012	
Drop of TVS	0.97462 ± 0.00016	
Loading TVS in a false position	0.97744 ± 0.00017	
Loading one excessive TVS	0.98225 ± 0.00017	1.0

Conclusions. The design of the TVS-X fuel assembly with rod fuel elements for the subcritical nuclear installation “Neutron source”, which is guided by an electron accelerator (SI), was developed in the NFC STE NSC KIPT.

The characteristics of the reactor core of the SI, the fuel assembly of the basic VVR-M2 variant and the alternative TVS-X are presented.

The variants of rod fuel elements with the use of UO_2 fuel pellets and UO_2–Al metal-ceramic composition are considered, experimental batches of fuel pellets and models of fuel elements are constructed; pre-reactor tests to substantiate reliability and safety in standard operating conditions and in emergencies were carried out.

Using the MCNPX^{TM} computer code the neutron multiplication factors in SI for UO_2 fuel pellets and UO_2–Al metal-ceramic composition were determined, depending on the enrichment of ^{235}U fuel.

It was proved that in order to provide the values of $K_{eff} \le 0.98$ in the reactor core of SI, the enrichment of UO_2 pellets should not exceed 4.44 % of ^{235}U weight, and while using UO_2–Al dispersion composition pellets the content of UO_2 particles enriched at 19.97 % of ^{235}U weight should not exceed 1.78 g/cm^3 (17.1 % vol.).

Using the SolidWorks Flow Simulation and TRANSURANUS (v1m1i09) programs the calculation of justification of TVS-X efficiency in standard operating conditions was done, the RELAP5/Mod.3.2 code was used at studying emergency processes.

The results of calculations showed that at passing transitional modes caused by the analyzed initial events, violations of certain eligibility criteria do not occur.

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