LOW ENRICHED NUCLEAR FUEL BASED ON URANIUM-ZIRCONIUM CARBONITRIDE: REACTOR TESTS AND POST-REACTOR STUDIES

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1. Introduction

JSC "NII NPO "LUCH" has developed a fuel based on uranium-zirconium carbonitride (CNT). To eliminate this problem, a project is currently being implemented to study the properties and characteristics of CNTs under irradiation conditions to achieve burnups of 5, 15, and 40% fifa in the SM-3 high-flux research reactor (Dimitrovgrad, Russia).

The critical facilities “Giacint” and “Kristal”, located in the scientific institution “JIPNR – Sosny”, Minsk, Belarus are used for the preparation of critical facility experiments on the multiplication units, simulating physical characteristics of the cores using low-enriched UZrCN-based nuclear fuel (19.75% 235U) so as to use them in the reactors on fast and intermediate neutrons with gas or liquid-metal coolants.

2. Reactor Tests of Low Enrichment Nuclear Fuel Based on Uranium-Zirconium Carbonitride

Table 1 – Comparison of key characteristics of fuels

Characteristic	UO_2	UN	U-Zr-C-N
Theoretical nucleus density of U, g/cm³	10,97	13,5	12,8
Thermal conductivity, W/m·K	3,0	26,0	32,0
Melting point, K	3078	3123	3470
The content of oxygen impurity, wt.%	—	< 0,15	< 0, 1

Table 2 - Parameters of the reactor experiment

Fuel	U – Zr – C – N
T of fuel	< 1600 К
Energy release, W/cm³	до 600
Burnup (fifa), %	5, 15, ≤ 40
Duration of the reactor experiment, year	≤ 5

2.1. Description of the Investigated Fuel

To implement the program for conducting reactor experiments, a batch of fuel pellets was manufactured and produced. Fuel pellets with a height of 5 mm and a diameter of 8 mm have an enrichment of 19.75% in the U-235 isotope, the chemical composition of UZrCN (equimolar fuel composition for light components), a density of 12.3 g/cm3. It should be noted that the resulting fuel has a high purity of oxygen impurities (less than 0.1 wt. %). The results of structural-phase studies of the manufactured fuel pellets in the initial state are shown in Fig. 1.

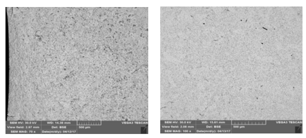

Figure 1 – General view of the cross section of the fuel pellet in the initial state: a) fracture section; b) thin section

2.2. Design of the Experimental Capsule and the Irradiation Device

To implement reactor tests of UZrCN fuel pellets, an experimental capsule (EC) design was developed (Fig. 2). EC contains 17 uranium-zirconium carbonitride pellets 8 mm in diameter and 5 mm thick. All pellets are installed by a tight fit in a non-sealed protective tungsten three-section shell. A sealed molybdenum housing is outside the tungsten shell. The EC cavity has been initially filled with helium at a pressure of 0.1-0.2 MPa. A compensation volume is provided in the lower part of the EC.

Thermoelectric transducers (TET) are installed on the molybdenum housing and in the second (counting from the top) pellet for reliable control of the CNT temperature.

The irradiation device (ID) consists of a sealed EC located inside a protective stainless steel casing. During irradiation, the outer steel casing is in contact with water in the primary circuit of the SM-3 reactor.

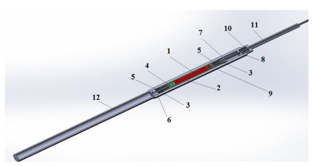

Figure 2 – Schematic representation of the experimental capsule: *1 - fuel column, 2 - tungsten shell, 3 - end springs (4 pcs.), 4 - tungsten table, 5 - Mo tube, 6 - lower molybdenum tube spacer, 7 - TET housing internal fixator (W), 8 - upper molybdenum tube spacer, 9 - sealed molybdenum housing, 10 - temperature compensation unit, 11 - Mo tube for TET output, 12 - volume for collecting gaseous fission products.*

2.3. Confirmation Calculations

The results of neutron-physical calculations of energy releases in ID materials are given in Table 3.

Thermophysical calculations have been carried out in order to determine the temperature regime of the EC during irradiation and its compliance with the requirements of the specification. The calculation results are shown in Fig. 3.

Fig. 4 shows the results of calculating the equivalent thermal stress in the Mo housing (MPa).

Table 3 - Calculated value of energy releases in ID materials

Object	Calculated energy release, W/cm³
UZrCN	509,9
W	64,1
Mo	27,83
Steel	17,71

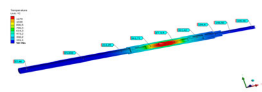

Figure 3 – Temperature field of the experimental capsule

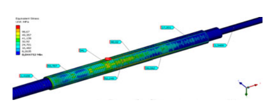

Figure 4 – Equivalent stresses in Mo housing

2.4. Methodological Experiment

A short-term methodical reactor experiment has been implemented as a preparation for long-term reactor experiments and confirmation of thermophysical calculations. Methodical reactor tests of the irradiation device with an experimental capsule with uranium-zirconium carbonitride fuel have been carried out in the reflector of the SM-3 reactor. The total exposure time was 23.28 effective days.

Changes in the temperature of the fuel, the molybdenum shell of the experimental capsule, and the pressure in the cavities of the ampoule and hanger during irradiation are shown in Fig. 5.

The average value of the volumetric energy release in the test pellets has been 516 W/cm³ during the methodical reactor tests. The burnup achieved in the test tablets is 0.63% fifa.

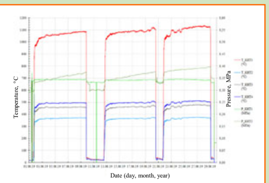

Figure 5 – Change in temperature and pressure in the irradiation device

2.5. Post-Reactor Studies

Fig. 6 shows the counting rate distributions for the fission product lines of Zr-95, Cs-137, and Nb-95 obtained by gamma-scanning the experimental capsule. The origin of coordinates on the diagram corresponds to the bottom edge of the capsule.

The X-ray patterns obtained as a result of the study make it possible to identify a cubic face-centered lattice (Fm-3m space group). No maxima belonging to other structures have been found in the X-ray diffraction patterns. The crystal lattice parameter (a) is 4.892 ± 0.0005 Å. Thus, according to the nameplate data, the phase composition of the fuel under irradiation have not changed.

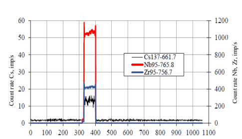

Figure 6 – Intensity distribution of the lines of some fission products along the length of the experimental

The metallographic analysis of the fuel structure of this specimen has been performed at two levels along the pellet height: near the surface and closer to the pellet center. For these purposes, after studying the structure near the pellet surface, the specimen has been again subjected to grinding to a depth of about 1 mm, then polished and chemically etched. There are no cracks in the body of the pellet, the macrostructure of the fuel is almost homogenious in the cross section of the pellet.

2.6. Reactor experiment (5% fifa)

On December 19, 2021, the reactor experiment has been started with duration of 200 eff. days. Fig. 8 represents the first data on the temperature of fuel pellets and the pressure in the ampoule under ionizing radiation conditions.

In March 2022, the reactor experiment has been stopped and the irradiation device has been moved to the spent fuel pool of the SM-3 reactor plant.

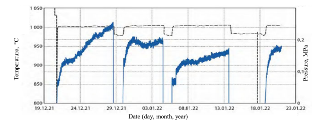

Figure 8 – Measurements of the temperature of fuel pellets (-) and pressure (---) in the ampoule during testing capsule

3. The Program of Experiments on the Fast and Intermediate-Neutron Critical Assemblies with 19.75% Enriched UZrCN Fuel at the Critical Facilities “Giacint” and “Kristal

At critical facilities “Giacint” and “Kristal” is planned to investigate fast-neutron critical assemblies with three types of fuel cassettes and different matrix materials (air, aluminum and lead). All of these critical assemblies will use Type 1 and Type 2 fuel rods.

The Type 1 (Type 2) fuel rod comprises a fuel core, cladding, and end pieces (Fig 8). The fuel rod cladding has the outer diameter 12 mm and the wall thickness 0.6 mm. The fuel rod comprises pellets, 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 total core height is 500 mm. The material of the fuel rod clad and end pieces (plugs) is stainless steel (Type 1 fuel rod) or alloy NbZr-1 (Type 2 fuel rod).

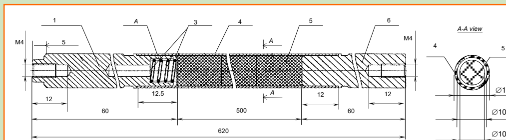

Fig 8. Type1 (Type 2) fuel rod : 1 – lower plug; 2 – spring; 3 – gaskets; 4 – cladding; 5 – fuel core; 6 – upper plug

3.1. Fast-Neutron Critical Assemblies

Fast critical assemblies, simulate physical features of the cores of the future fast-neutron reactors with gaseous (F-20-A) and liquid-metal (F-20-L, F-20-AL) coolants (Fig. 9). This fast critical assemblies represent a lattice of fuel cassettes with fuel rods with air or lead or aluminum as matrix materials, with a beryllium-steel reflector.

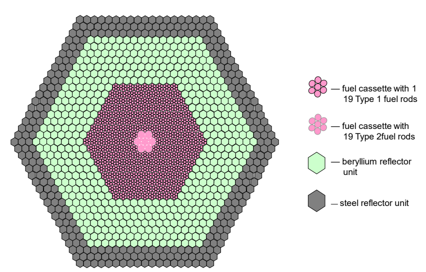

Fig 9. The loading diagram of the F-20-A (F-20-L, F-20-AL) fast critical assembly

3.2. Intermediate-Neutron Critical Assemblies

Intermediate-neutron critical assemblies H-20-1 and H-20-2 with a hydride zirconium moderator contain fuel cassettes, cassettes for the control rods and cassettes and blocks of reflectors in the form of a hexagonal lattice.

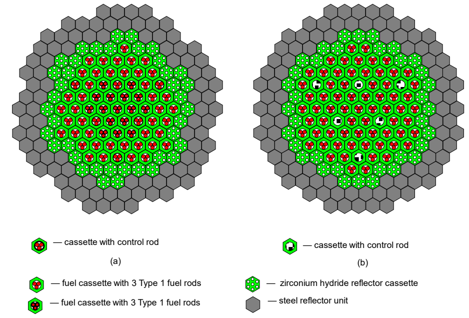

Figure 11 — Loading diagrams of H-20-1 (a) and H-20-2 (b) critical assemblies

4. Conclusions

As a result of the study of the properties and characteristics of fuel based on uranium-zirconium carbonitride under irradiation conditions, a methodical reactor experiment has been carried out in the SM-3 reactor. After conducting a methodical reactor experiment, a program of post-reactor studies of irradiated specimens has been implemented. A program of experimental work at these critical assemblies was prepared and safety justification was carried out during their implementation.

Design documentation has been developed for fast-neutron and intermediate-neutron critical assemblies containing fuel assemblies with nuclear fuel based on low-enriched UZrCN. The elements of intermediate-neutron critical assemblies have been made, and preparations are underway for experiments. A program of experimental work at these critical assemblies was prepared and safety justification was carried out during their implementation.