NUCET NUCLEAR ENERGY AND TECHNOLOGY

Nuclear Energy and Technology 10(2): 111–115
DOI 10.3897/nucet.10.130131

Research Article

# Modernization of uranium-zirconium fuel rod of IVG.1M research reactor*

Vladimir A. Solntsev[1], Dmitry M. Soldatkin[1], Vladimir N. Nuzhin[1]

1 *LUCH Research and Production Association, Research and Development Institute, Joint-Stock Company, 24 Zheleznodorozhnaya Str., 142103 Podolsk, Moscow Region, Russia*

Corresponding author: Vladimir A. Solntsev (solntsevva@sialuch.ru)

**Academic editor:** Georgy Tikhomirov ♦ **Received** 21 December 2023 ♦ **Accepted** 20 June 2024 ♦ **Published** 5 July 2024

**Citation:** Solntsev VA, Soldatkin DM, Nuzhin VN (2024) Modernization of uranium-zirconium fuel rod of IVG.1M research reactor. Nuclear Energy and Technology 10(2): 111–115. https://doi.org/10.3897/nucet.10.130131

## Abstract

This paper describes the development of a dispersion-type uranium-zirconium fuel rod. Uranium is distributed in the zirconium matrix material in the form of axis-oriented fibers. The fuel rod is designed for the conversion of the IVG.1M research reactor (Republic of Kazakhstan) from highly enriched uranium (HEU) to low enriched uranium (LEU). The need for the HEU-LEU conversion arose in connection with Kazakhstan joining the program to convert research and test reactors to fuel with reduced enrichment (RERTR 2023). The study solves the problem of deformation of a low-tech U-Zr alloy (located in a zone of low plasticity) by replacing them with a heterogeneous compound.

The manufacture of fuel rod is based on metal forming processes. Initially, a fuel rod wire with a core of fiber structure is formed by triple co-extrusion of cylindrical uranium and coaxial zirconium billets. At the next stage, the wire is processed to the required diameter by drawing, then the operation of flattening, twisting and cutting into specified lengths is carried out. Upon reaching a high total degree of deformation obtained during cold work, relaxation annealing is carried out at temperatures of 550 to 600 °C, which leads to the formation of a transboundary layer of the $UZr_2$ intermetallic compound with a thickness of 1 to 2 μm. The intermetallic layer, without having a significant effect on the strength and thermal conductivity of the compound, ensures high quality diffusion bonding of all fuel rod components. The final operations are melting of the ends of the fuel rods and sealing by electroplating with nickel. As a result, blade-profile fuel elements are obtained with a thickness of 1.5 mm, a diameter of the circumcircle of 2.8 mm and an average effective diameter of uranium fiber of 40 μm.

A set of 14040 fuel rods was manufactured and loaded into an operating IVG.1M reactor. The power start-up took place in 2023. Due to the unification and wide variability in the loading of the fuel component, in size and shape of the cross section, in structure and materials of the matrix compound, the fiber fuel element design can be used in the development of fuel rods for advanced reactors for various applications.

## Keywords

uranium-zirconium fuel rod, compound, technology, extrusion, drawing, cutting, heat treatment, intermetallic compound, temperature regime, fiber structure, research reactor

* Russian text published: Izvestiya vuzov. Yadernaya Energetika (ISSN 0204-3327), 2024, n. 1, pp. 96–106.

# Introduction

The need for low enriched fuel for research reactors has emerged in connection with the international program to convert research and test reactors to fuel with a U-235 isotope enrichment of below 20% (RERTR 2023).

As part of the program, a project was launched in 2010 in the Republic of Kazakhstan to convert the IVG.1M thermal-neutron pressurized-water research reactor (IVG.1M) based at the National Nuclear Center (NNC RK). The conceptual precondition for the reactor upgrading, as defined by Kazakhstan's Atomic Energy Agency, is that the thermal-hydraulic and neutronic performance of FAs and the reactor core be preserved with the same geometrical parameters of fuel elements. Computational studies undertaken by the US Argonne National Laboratory and the NNC RK have shown that the replacement of highly enriched (HEU) fuel with low enriched (LEU) fuel will not lead to major changes in the reactor performance with the mass fraction of uranium in the LEU fuel kernel to increase from 3–4.3% to 15.5–20.6%, i.e. by a factor of about 5.5. The HEU-LEU conversion was not however feasible with the extrusion technology used for fuel with a low U content to be automatically extended to fuel with a ~5 times higher U content due to the fact that the U-Zr alloy composition required for the LEU fuel rod fabrication is in the smallest plasticity region (Fig. 1) (Emelyanov et al. 1968), which makes it practically impossible to use extrusion.

The problem was solved by replacing the U-Zr alloy for a heterogeneous composition. The fact that the plastic characteristics of U and Zr are 4 to 6 times higher in pure form than those of the alloy with the required composition and differ slightly in terms of values, allows U and Zr to be jointly deformed. It was therefore decided that a fuel kernel be designed for LEU rods in the form of a dispersion-type bimetallic composition with a fiber structure based on U and Zr.

The concept of the fiber fuel kernel design was proposed in the 1990s by Dyakov et al. 1999.

The fuel element is a rod with a blade-shaped cross-section twisted about itself (see Fig. 2).

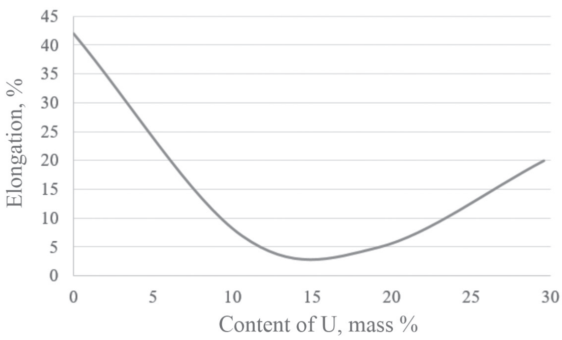


**Figure 1.** Elongation of U-Zr alloys at 370 °C.

Taking into account the condition of the FA design continuity with respect to the design of the LEU fuel components, the following requirements have been specified for the latter (see Table 1).

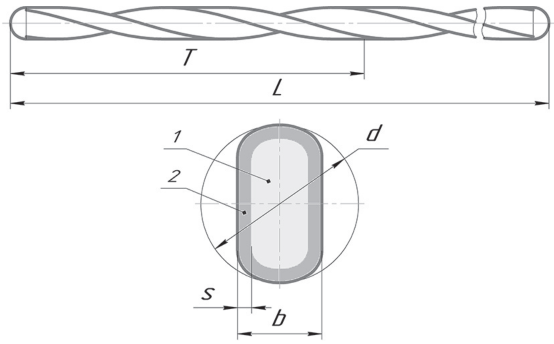


**Figure 2.** Design of the IVG.1M reactor fuel rod: 1 – kernel (U-Zr alloy); 2 – cladding (E110 alloy); $L$ – length; $T$ – twist pitch; $d$ – circumcircle diameter; $b$ – blade width; $s$ – cladding thickness.

**Table 1.** Fuel rod requirements specifications

| | |
|---|---|
| Circumcircle diameter, mm | $2.8_{-0.04}$ |
| Blade width, mm | 1.5 ± 0.03 |
| Length, mm | $800_{-1}$ and $600_{-1}$ |
| Blade twist pitch, mm | 38 ± 8 |
| Kernel U content, mass % | 15.5, 18.9, 20.6 |
| U-235 enrichment, % | $19.75^{+0.1}/_{-0.3}$ |
| U-235 content deviation from rated value, % | + 6 |
| Rod length non-uniform U distribution factor, % of mean value | ± 12 |
| Surface U-235 contamination density, g/cm$^2$ | $1 \cdot 10^{-8}$ |

Disclosed further are the key aspects of the technology to develop dispersion-type fiber LEU fuel components for the IVG.1M reactor.

# Flow diagram of the fiber fuel rod fabrication

Most efficient and least labor-intensive technologies were used for the fuel rod fabrication (pressing, drawing and section rolling of bimetallic U and Zr compositions). A schematic flow diagram of the fiber fuel rod fabrication is presented in Fig. 3 (Dyakov et al. 2017).

Fig. 4 illustrates the key stages in the kernel and fuel rod formation.

# Process stages

A bimetallic bar is produced by pressing at stage 1 the geometry of which defines the mass fraction of uranium in the kernel. The assembly consists of a cylindrical uranium rod and a sleeve with a zirconium (E100 alloy) end plug enclosed, in turn, in a copper sleeve with its own end plug. Prior to pressing, the zirconium sleeve was treated in a clarifying chemical solution to remove the oxide film from its surface. Grit paper is used to remove the oxide layer from the uranium surface. All pressing assembly components were washed in alcohol. The zirconium and copper sleeves were pressurized using electron-beam vacuum bonding to avoid uranium and zirconium oxidation.

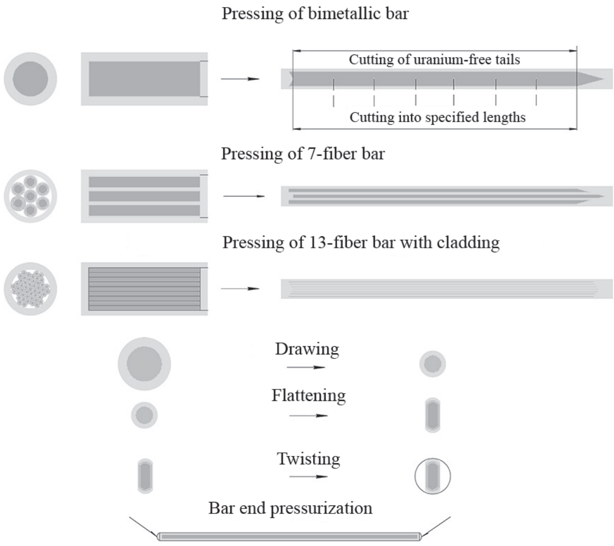


**Figure 3.** Flow diagram of the fiber fuel rod fabrication.

The heated billet was initially extruded through a heated matrix on a vertical hydraulic press. Normally, the ratio of the initial billet diameter to the billet height does not exceed 1:4. The resultant bar diameter is 3 to 5 times smaller than the initial diameter. The copper cladding is removed from the bar using a nitric acid solution.

An assembly of 7 bimetallic bars is produced by pressing at stage 2. After the copper cladding is removed, the bimetallic bar produced as the result of the initial pressing is cut into the specified number of segments which are composed further to form a symmetrical bundle of 7 bars and enclosed into the copper cladding.

An assembly of 19 seven-fiber bars is produced by pressing at stage 3. The bar produced in the second pressing cycle is cut into the preset number of relatively short bars, the uranium content in each rod is determined by the hydrostatic weighing method, and a bundle of 19 bars is composed with the required content of uranium which are enclosed further into the zirconium sleeve to form the fuel rod cladding billet. It is possible to adjust the ratio of uranium and zirconium in the fuel rod at stage 3 by selecting the uranium mass in the bar bundle or by changing the zirconium sleeve thickness. The assembly is enclosed into the copper sleeve and pressed.

Stage 4 is cold working (drawing, flattening, and twisting). For drawing, the bar is passed through a number of hard-alloy dies with the consecutively decreasing hole diameter.

Drawing is followed by the cylindrical fuel rod wire being flattened by passing it twice between the mill rolls in a rectangular ring groove on the lower roll. The ring grove dimensions were determined experimentally taking into account the wire elongation of 8% in the process of flattening.

After the cold working operation was finished, fuel rods of the specified length were cut from the long-length fuel rod wire with the final blade shape obtained.

Fuel rods were so produced with the blade thickness of 1.5 mm, the circumcircle diameter of 2.8 mm, and a length of 600 and 800 mm. The average effective diameter of the uranium fiber is 40 µm, and the mass content of uranium in the kernel is 15 to 21%.

The fuel rod ends were pressurized by a 30 to 100 µm thick layer of nickel with the end region having been fused in advance to apply it by electroplating.

The number of the pressing cycles depends on the required uranium fiber dimensions and by the allowable deformation ratio at each stage. The required number of the drawing passes was determined experimentally to achieve the required diametrical dimension and the preset roughness of the fuel rod wire surface.

Finished fuel rods are shown in Fig. 6.

# Selection of pressing modes

The three initial pressing stages include diffusion bonding of the kernel components. Producing the joint by diffusion bonding method requires the best possible combination of all process parameters, such as temperature, deformation ratio per pass and total deformation ratio, direction of deformation relative to the fiber laying direction, and ratio of the matrix and fiber plasticity (Zarapin et al. 1991). Pressing bimetallic billets is quite a rapid process (20 to 30 seconds long) required largely for the material to pass through the rolls, the die, etc., due to which the key factors that affect the diffusion bonding quality of the fuel rod structure components are temperature and deformation ratio. This requires taking into account the potentiality for phases to be formed with low performance characteristics (brittleness, endurance, high fusibility, etc.) as may result from the interactions of basic materials among each other and with process impurities.

It has been found by analyzing the compatibility of uranium and zirconium in accordance with the diagram of state that a process starts at a temperature of above 612 °C which forms a brittle intermetallic compound, $UZr_2$, the presence of which in the process of the fuel rod billet cold working, with the layer thickness being in excess of 4 to 5 µm, may lead to formation of defects and further ruptures.

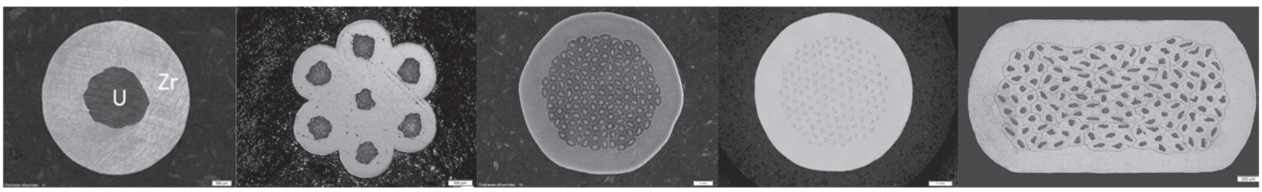


**Figure 4.** Micrographs of the fuel rod billet cross-section in the sequence of the process stages.

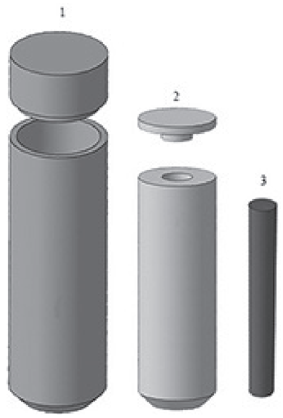


**Figure 5.** Pressing assembly components for pressing stage 1: 1 – copper sleeve with plug; 2 – zirconium sleeve with plug; 3 – uranium rod.

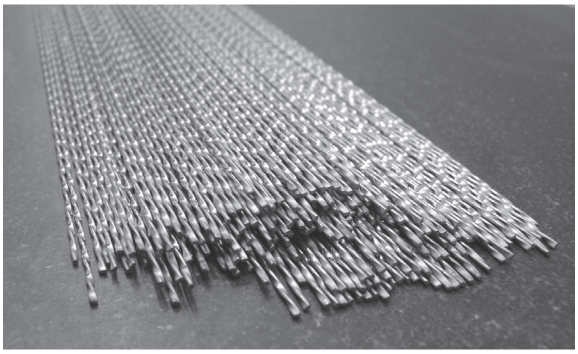

**Figure 6.** LEU fuel rods for the IVG.1M reactor.

With the above factors taken into account, the best possible thermomechanical pressing modes were deteremined experimentally: temperature at the upper uranium alpha phase existence boundary and deformation rate of about 94%, with which the intermetallic compound layer does not exceed 2 μm.

The effect of a local uranium fiber temperature increase due to the presence of a 2 μm thick $UZr_2$ intermetallic compound layer was evaluated separately. The overheating of uranium relative to the surrounding matrix is negligibly small, that is no more than one degree (the heat conductivity coefficient of the intermetallic compound is assumed to be 7.9 W/(m·K)) (Takahashi et al. 1988).

Due to zirconium and uranium being highly chemically active, the billets underwent heat treatment in an argon environment or in vacuum to prevent the formation of oxide and nitride films that hampered diffusion bonding.

With the above modes used for pressing, high quality of diffusion bonding is achieved for all semi-finished fuel rod kernel components and cladding.

# Selection of drawing modes

To obtain the required surface quality and the required size of the fuel rod wire (the final diameter is 2.3 mm), the best possible option is the process of drawing through hard-alloy dies. The existing experience in zirconium wire production was used to select the drawing mode with the semi-finished product diameter reduced by 0.2 mm per pass and the total deformation ratio being not exceeding 50%.

After the specified total deformation ratio was achieved, the fuel rod wire was annealed for one hour at a temperature of 550 to 600 °C to relieve stresses in the wire and to increase its plasticity. In addition, such a low zirconium annealing temperature limits the rate of diffusion processes at the uranium-zirconium interface.

The bimetallic product deformation process at the drawing stage has a number of features which manifest themselves in the presence of non-uniformities in the fuel rod's macro- and microstructure. The fiber structure of the kernel is formed in the process of repeated pressing, so the bonding strength is not uniform across the bundle volume.

Bonding theory says that there is a non-uniform distribution of tensile stresses in different concentric layers as the material passes through the deformation zone of the drawing die. Tensile stresses in the compression zone are greater in the central layers than in the peripheral layers, which can cause the metal drawn to lose integrity in the central layers. Besides, due to the shift of uranium along its interface with zirconium, internal breaks of fibers may occur, which is typical of the drawing process (Perlin, Ermanok 1971). This effect can be clearly seen when comparing the cross-sections of the same wire at different processing stages, that is after pressing and after drawing (Fig. 7). The dark central areas of the bimetallic components in the right-hand image correspond to the uranium fiber break points. After the flattening operation, the pores resulting from the uranium fiber breaks are closed normally, since the greatest plastic strains occur in the fuel rod's central part.

On the other hand, with relatively low tensile stresses in the peripheral layers and the same state of the metal drawn throughout the deformation region, transverse surface cracks are unlikely to form. Indeed, no defects, such as transverse cracks, are observed in the process of the fuel cladding integrity inspection.

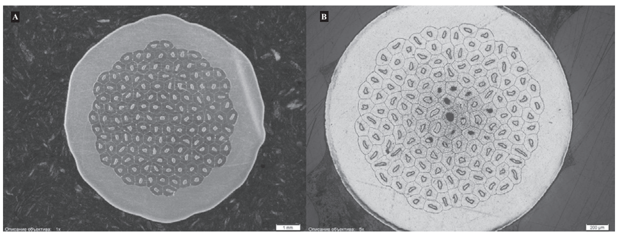


**Figure 7.** Cross-section of fuel rod wire: **A.** After pressing; **B.** After drawing.

# Fuel rod quality control

To assure quality of the LEU fuel rods, a system was developed and implemented for monitoring their key parameters (see Table 2). The system was provided with required equipment and verified measurement procedures, including those developed specifically for the IVG.1M LEU fuel rods (Yakovlev et al. 2016).

**Table 2.** System for monitoring key fuel rod parameters

| Monitored parameter | Determination method | Dedicated equipment |
|---|---|---|
| U-235 content in fuel rod and uniformity of its lengthwise distribution | Gamma spectrometry of uranium-emitted radiation | PNK SUT-2 instrument for measuring uranium content and its non-uniformity in IVG.1M fuel rods |
| Enrichment | Recording of uranium isotope ion currents using a mass spectrometer | |
| Chemical composition of fuel rod | Set of analytical methods for chemical analysis | |
| Fuel rod surface uranium contamination | Recording of uranium isotope natural alpha radiation | PNK RZA-7M instrument to monitor fuel cladding contamination |
| Side cladding integrity | Visual examination for intolerable defects | |
| Side cladding thickness | Selective inspections on longitudinal and transverse sections of witness samples | |
| End coating thickness | Eddy current induction | Eddy current flaw detector (Vektor) |
| End coating continuity | Recording of alpha radiation on fuel rod ends | DKS-96 dosimeter-radiometer with BDZA-96t detector |
| End coating adhesion strength | Testing of witness samples by separating coating from substrate | Special-purpose equipment |

# Conclusions

A uranium-zirconium microheterogeneous fuel rod with a fiber structure was developed for the HEU-LEU conversion of the IVG.1M reactor core. Fabrication of the fuel rods is based on metal forming operations: pressing, drawing, and flattening. Deformation modes have been defined for each operation, and the parameters of the heat treatments between operations were determined and confirmed by material testing.

Expert estimates and experimental studies, including full-life in-pile testing of 936 fuel rods as part of two experimental fuel assemblies, have shown the uranium-zirconium fiber fuel rods to meet all conditions for the IVG.1M reactor conversion.

A set of 14040 fuel rods was manufactured for the reactor core and loaded into the operating reactor. For the reactor first criticality, 1428 composite fuel rods were manufactured using 100, 200, 300 and 400 mm long segments. A batch of 270 fuel rods for destructive tests was manufactured separately. The power startup of the IVG.1M-LEU reactor took place on May 18, 2023.

Thanks to unification and extensive variability in terms of the fuel component loading, the cross-section dimensions and shape, and the structure and materials of the matrix composition with relatively simple readjustment of rolling technologies, the fiber technology can be used to develop fuel rods for advanced reactors for various applications.

# References


- Dyakov EK, Karetnikov IA, Nuzhin VN, Maskaev AS, Permyakov LN (1999) Fuel rods with increased environmental protection for research nuclear reactors. Izvestiya Akademii Promyshlennoy Tekhnologii 2: 19–23. [in Russian]
- Dyakov EK, Zhalilov RKh, Karetnikov IA, Kolganov AA, Martynenko AV, Nuzhin VN, Soldatkin DM, Solntsev VA (2017) Dispersion fuel rod and the method of its manufacture. Patent EAPO No. 027036, MPK G21C 3/02, 21/02, Application. Dec. 12, 2015, Published June 30, 2017. [in Russian]
- Emelyanov VS, Evstyukhin AI (1968) Metallurgy of Nuclear Fuel. Ed. 2 Moscow, Atomizdat Publ., 484 pp. [in Russian]
- Perlin IL, Ermanok MZ (1971) Drawing Theory. Ed. 2. Moscow, Metallurgiya Publ., 448 pp. [in Russian]
- RERTR (2023) Reduced Enrichment for Research and Test Reactors. https://www.rertr.anl.gov/ [accessed Nov. 20, 2023]
- Takahashi Y, Yamawaki M, Yamamoto K (1988) Thermophysical Properties of Uranium-Zirconium Alloys. Journal of Nuclear Materials 154: 141–144. https://doi.org/10.1016/0022-3115(88)90127-4
- Yakovlev EI, Zhavoronko AI, Elanchik AA, Babin AL, Gorbatenko AYu, Cheremukhin AI, Efremov YuP, Soldatkin DM, Nuzhin DM (2016) Arrangement calibration and errors of uranium mass measurements in fuel rods using natural gamma rays. VANT, Ser. Tekhnicheskaya fizika i avtomatizatsiya 72: 17–32. [in Russian]
- Zarapin YuL, Chichenev NA, Chernilevskaya NG (1991) Production of composite materials by pressure treatment. Metallurgiya Publ., Moscow, 351 pp. [in Russian]