Perspective Fuel Rod with Alloyed Lead in Contact Layer for Reactor BREST-OD-300

Daria Fedorovich[1] Elena Ignatieva[1,*] Irina Paramonova[1]

[1] *Institute of Energy, Peter the Great St. Petersburg Polytechnic University (SPbPU), St. Petersburg, Russia*
[*] *Corresponding author. Email: elig1359@yandex.ru*

ABSTRACT
The article examines the temperature field of the fuel cell of the BREST-OD-300 nuclear reactor if helium in the gap between the fuel pellet and the fuel element shell is replaced with a heat-conducting Pb-Mg-Zr liquid metal layer. The temperature field is calculated for the fuel rod with the highest power density in the mode of outage. In the future BREST-OD-300 would provide a closed nuclear fuel cycle and clean energy.

Keywords: *BREST-OD-300, Inherent safety, Closed nuclear fuel cycle, Fuel rod's contact layer, Lead, Eutectic, MNUP-fuel, Outage.*

1. INTRODUCTION

Due to the growth of energy consumption and climate change, State Atomic Energy Corporation Rosatom presents the energy concept Green square. This concept implies a transition to carbon-free energy sources and the need for accelerated development of nuclear and hydropower, as well as wind and solar energy. Nuclear power plants (NPP) do not emit carbon dioxide into the atmosphere and do not depend on external factors as much as renewable sources, also NPP are not limited by the resource of rivers like hydroelectric power plants. Moreover, nuclear energy is the most energy – intensive. Nevertheless, uranium and thorium resources are finite, therefore reprocessing of the spent fuel and closed nuclear fuel cycle is necessary.

The BREST-OD-300 nuclear reactor is under construction now in Seversk, Tomsk region, Russia. This reactor is part of Russia's Proryv (Breakthrough) project, which envisages the development of a new nuclear industry technology platform based on a closed nuclear fuel cycle with liquid metal cooled fast reactors. Using the mixed nitride uranium-plutonium fuel (MNUP-fuel) and pure lead coolant provides the inherent safety of the reactor. In future this Generation IV reactor would become a clean and energy-intensive energy source with its own fuel supply.

The Experimental and Demonstration Energy Center (EDEC) is established on the site of the Siberian Chemical Combine, Seversk. EDEC includes a Reprocessing Module for reprocessing of the spent MNUP-fuel and Fabrication and Re-fabrication Module which allows to work with both source materials and products of spent nuclear fuel reprocessing of the BREST-OD-300 reactor. The MNUP fuel itself is a ceramic mixture of waste uranium mononitride and plutonium mononitride. Thus, already extracted fuel mostly will be used in this "short fuel cycle" and depleted uranium will be returned into the fuel cycle. The disposal of radioactive waste is minimized and does not violate the radioactive balance of the Earth. Moreover, minor actinides will be burned in the BREST-OD-300, disposal and handling of which is very dangerous because of their radioactivity and high toxicity. One more advantage of the "short fuel cycle" is the minimization of the movement of nuclear materials, it reduces the radioactive trace from the nuclear power plant.

When speaking about the NPP as a clean source of energy, its safety is worth mentioning. Low moderating ability of lead makes it possible to expand the fuel cell grid and increase the flow section of the lead coolant. The joint use of lead coolant and MNUP-fuel provides a small and stable excess reactivity. Therefore, the inherent safety excludes accidents with loss of heat removal and accidents with an uncontrolled increase of power. The high melting point of lead (about 300°C) does not require high pressure in the circuit, moreover, lead is unreactive with water and air. MNUP-fuel is considered tolerant (ATF – Accident Tolerant Fuel), since its high thermal conductivity with the temperature rises grows too, in contrast to the uranium dioxide which has low thermal conductivity decreasing with the growing of the temperature. BREST-OD-300 is an integral layout reactor, this feature makes it possible to localize coolant leaks and draining of the reactor core will be impossible. Above-mentioned characteristics of the BREST-OD-300 makes it possible to avoid radioactive environmental pollution in case of an accident on the station.

Nevertheless, in fuel elements with a helium sublayer between the shell and the fuel, especially at high burnup, the workforce of the fuel elements is often determined by the contact of the shell and the fuel pellet. Eutectic lead contact layer could exclude the thermomechanical action between shell and fuel and reduce the yield of gaseous fission products and decrease their pressure on the shell. The energy accumulated in the fuel may be decreased by using lead eutectic due to its high versus helium thermal conductivity in the contact layer. Less accumulation of the energy leads to more favorable consequences in the case of a breakdown of forced circulation.

2. PROBLEM STATEMENT

For the BREST reactor, sodium and lead contact layers can be considered. In the case of sodium, there is a void effect of reactivity due to the possibility of boiling; pure lead is also unsuitable for this task since this material causes corrosion of the fuel element shell as it was found out during experimental irradiation in the BOR-60 reactor. Thus, searching material for this case remains an urgent problem. To reduce the intensity of corrosion processes, it was proposed to alloy lead with some components. For instance, the researchers of Leypunsky Institute for Physics and Power Engineering (IPPE JSC) offered to dope the lead with magnesium and zirconium. Scientists of Advanced Research Institute of Inorganic Materials named after Academician A. A. Bochvar (VNIINM) proposed to alloy lead with elements of the structural material of the fuel shell – EP823 steel; post-irradiation studies of these fuel elements irradiated in the BOR-60 reactor in the experimental assembly were also carried out. The results obtained confirm the efficiency of alloying the lead contact layer with EP823 components; however, additional studies of alternative alloying components are required [1]. For further calculations, the fuel element with a heat-conducting sublayer made of a lead-based alloy doped with 2.25 % Mg to and up to 0.2 % Zr is considered. Using this material leads to the formation zirconium carbonitride on the surface of the structural material in a lead-based eutectic containing magnesium and zirconium, which provides anti-corrosion protection, creating an additional barrier for steel nitriding and contributes to the “self-healing” of minor damage to the coating [2].

The lead-bismuth system is provided as a coolant in the SVBR-100 reactor project, where a fuel element shell material is also EP823. However, the thermal conductivity of this system differs slightly from the thermal conductivity of a lead-magnesium-zirconium alloy. Therefore, the consideration of the Pb-Bi eutectic is not so important for the thermal problem, especially because the experimental data about suitability of this material as a contact layer in the fuel element of the BREST-OD-300 reactor are absent now.

3. THE METHOD OF CALCULATING AND THE RESULTS

The temperature distribution along the height of the core for the fuel rod with the highest power density was obtained for the reactor under thermal capacity (700 MW) operation. A fuel rod envisaged by the project with helium gap and fuel rod with lead-based eutectic containing magnesium and zirconium were considered. The maximum temperature in the center of the fuel pellet with lead eutectic was 785°C. For the similar fuel element with helium in contact layer the maximum temperature was about 1150°C. The temperature difference between the center of fuel pellets and fuel rod shells has dropped about 50 times: from 423.58°C to 8.51°C, it's demonstrated the effectiveness of this replacement.

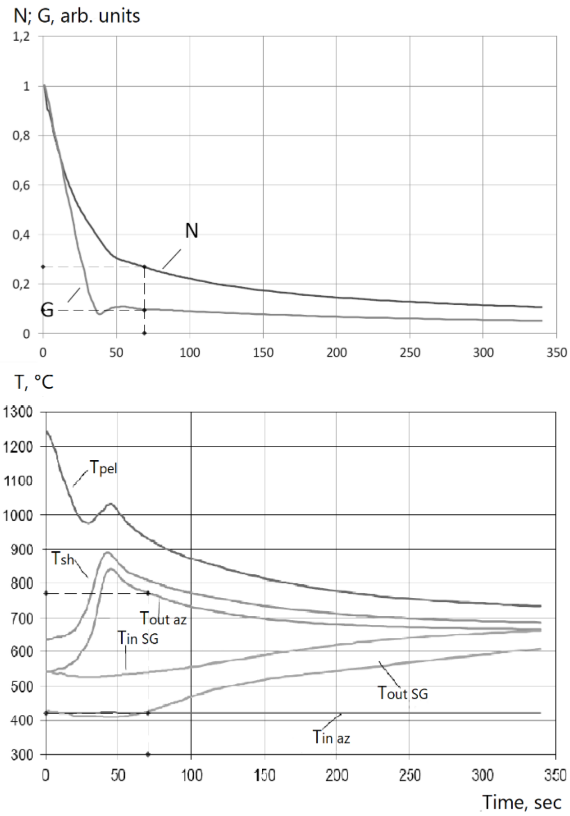

Figure 1. Outage of BREST-OD-300

Further the scenario with a complete outage was considered: while the reactor is operating under nominal thermal capacity, four main circulation pumps (MCP) turned off and the supply of feed water is stopped. The analysis of this scenario (Figure 1) is presented for the core provided for by the project, that means that the gap between the fuel element shell and the fuel pellet is filled with helium [3].

According to the graphs in Figure 1, the thermal power decreases from 700 to 189 MW, the flow rate drops to 4170 kg/s, the average inlet and outlet coolant temperatures of the reactor core are 420 °C and 775°C, respectively, at the moment when the natural circulation of the coolant is established (approximately 70 seconds after power outage). The coolant circulation velocity is 0.21 m/s.

Thus, the calculations were carried out for the fuel rod with the highest power density. The coolant temperature T_c, depending on the altitude, was determined as:

$$T_c(z) = T_{in} + \Delta T / 2 \times \left(1 + \sin(\pi z / H_{ef}) / \sin(\pi H_{az} / (2H_{ef}))\right),$$

where T_{in} – the average coolant temperature at the reactor core inlet, °C; ΔT – difference between inlet and outlet core temperatures of coolant, °C; H_{az} – height of the fuel in the reactor core, mm; $H_{ef}=H_{az}+2\delta_{ef}$ – reactor core height including reflector (δ_{ef}=200 mm), mm.

Convective heat transfer from the fuel rod shell to the lead coolant is determined according to Newton's law. It is necessary to determine the heat transfer coefficient for the calculation of the shell temperature on the outside.

The heat transfer coefficient was obtained from formula by A.V. Zhukov [4]. This formula based on the results of experimental studies of the thermohydraulic characteristics of the BREST reactor:

$$\alpha = Nu \times \lambda / d =$$
$$= \left(7{,}55 \times S/d - 14 \times \left(S/d\right)^{-5} + 0{,}007 Pe^{0{,}64+0{,}246 S/d}\right) \times \lambda / d,$$

where S/d – the gap ratio of the fuel cell grid; Pe=wd/a – the Peclet number; w – the coolant rate, m/s; λ – thermal conductivity coefficient of coolant, W/(m・K).

To determine the temperatures inside of the fuel rod shell, outside and in the center of the fuel pellet, an iterative calculation was implemented. The temperature in the center of the fuel pellet is defined as:

$$T_{pel}^{c}(z) = T_{sh}^{o}(z) + q_l(z) \times (R_{sh}(z) + R_g(z) + R_{pel}(z)),$$

where T_{sh}^{o} (z) – the temperature outside the shell, °C; q_l (z) – the linear heat flux, kW/m; R_{sh} (z) – thermal resistance of the shell, m·K/W; R_g (z) – thermal resistance of the gap, m·K/W; R_{pel} (z) – the thermal resistance of the fuel pellet, m·K/W.

Thermal resistances depend on geometry and thermal conductivity. Thermal conductivity is determined at average temperatures in the shell, gap and fuel. Therefore, it is necessary to carry out several iterations to obtain accurate temperature values at the center of the fuel pellet.

As a result, the temperature distribution along the height of the core was obtained for a fuel element with eutectic in the contact layer after the establishment of natural circulation under outage (Figure 2).

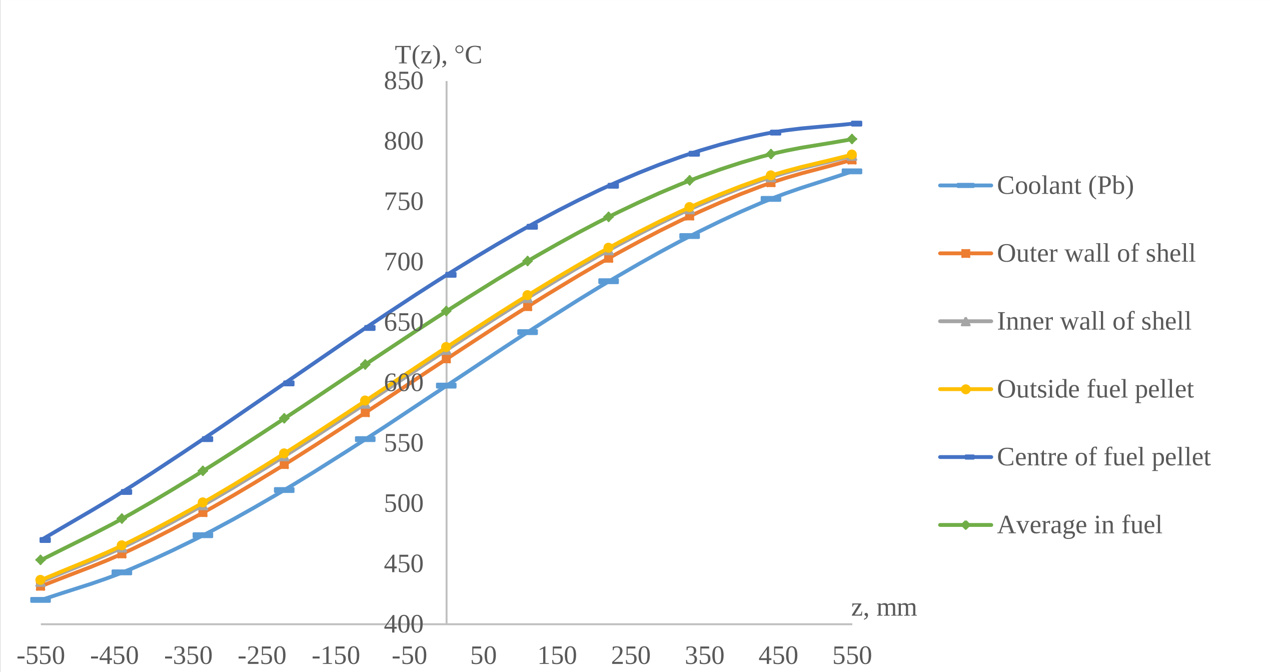

Figure 2. Temperature distribution depending on the altitude for the fuel rod with the highest power density

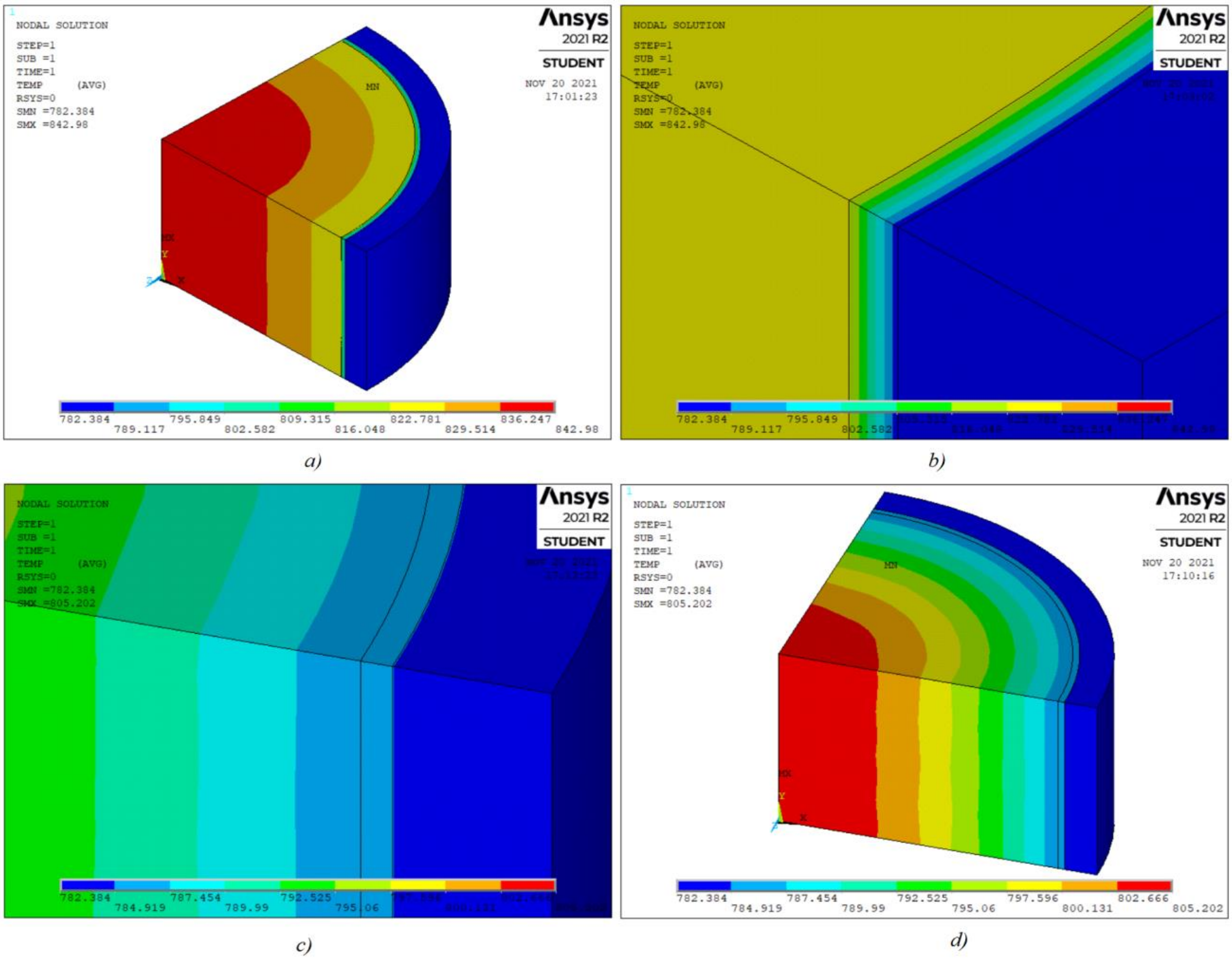

Figure 3. Numerical simulation of rod's temperature distribution (a. Rod with helium; b. Temperature gradient in helium gap; c. temperature gradient in eutectic gap; d. rod with eutectic Pb-Mg-Zr.)

It should be noted that this calculation of the fuel element temperature field is rather an estimate, as in the case of a fuel element with a lead-based alloy doped with 2.25 % Mg – up to 0.2 % Zr in the contact layer, the parameters will change, since the fuel will accumulate much less energy and heat removal will be more intensive, this means that the velocity and flow rate of the coolant will increase. Nevertheless, in the center of the fuel pellet the maximum temperature is about 815 °C, while for a helium fuel element, according to Figure 1, it is approximately 950°C. Since lines on the graph of the temperature distribution along the height of the core are equidistant, it can be judged that the temperature gradient in the fuel element with the eutectic decreases. Numerical simulations were performed to ensure this hypothesis.

The site, relevant to the exit from the core, where the fuel temperature is the highest, is considered. The result of numerical simulation from the software Ansys of a fuel element with helium in contact layer is shown on Figure 3(a) and Figure 3(b). The temperature in the center of the fuel is 842 °C, the temperature difference in the gap is 40 °C. The results of a similar simulation for a fuel element with a eutectic are presented on Figure 3(c) and Figure 3(d): the temperature in the center - 805°C, the difference in the contact layer is 2°C. Thus, the temperature gradient decreases, which leads to decrease of the thermal resistances, smoothes out the neutronic characteristics (the temperature and density reactivity effects damp) and reduces the amount of energy accumulated in the fuel.

4. CONCLUSION

The materials used for the core of the BREST-OD-300 have some distinctive features to ensure inherent safety. However, helium in the gap between the fuel pellets and the fuel rod shells lead to accumulation of heat in the fuel, although the gas carries out its function - compensates the fuel size change during swelling and ensures the release of gaseous fission products. According to post-irradiation examination, the contact layer with alloyed lead can carry out all these functions as well. Moreover, the eutectic contact layer not only ensures the release of gaseous fission products, but also decreases its intensity, the pressure on the shell and reduces the swelling rate of the fuel [5]. According to the developers of the fuel rod with a heat-conducting contact sublayer made of a lead-based alloy doped with 2.25% Mg and up to 0.2% Zr, these alloying components make it possible to achieve the lowest level of corrosion damage to the shell [2]. This paper shows that replacing helium with a more heat-conducting material reduces the maximum temperature from 1150°C to 785°C in the center of the fuel if the reactor is under nominal capacity operation. The temperature difference between the fuel rod shell and the center of fuel pellets decreases by 50 times: from 423.58 °C to 8.51 °C. According to the results of numerical modeling, the temperature gradient decreases if a heat-conducting liquid metal layer is replacing the helium in the gap between the fuel pellet and the shell. There is a heat-conducting liquid metal layer in the gap that leads to the decrease in temperature stresses and the increase of the workforce of the fuel element. The Figure 3(b) shows that in a thin helium gap the temperature difference is greater than in the fuel and especially in the shell. For the emergency case of power failure with shutdown of four main circulation pumps and the interruption of feed water supply while operating at initial power, temperature reduction is even more important. In this case, according to the performed calculations, the maximum temperature is about 815 °C in the center of the fuel pellet, while for a helium fuel element, as shown at Figure 1, temperature is approximately 950°C. Melting points of fuel and shell in this case are not reached, the boiling point of lead is more than 2000°C, therefore, the heat transfer crises are excluded at an average coolant temperature of 775°C at the exit from the core.

In summary, the operation of the fuel element with liquid metal sublayer for the core of the BREST-OD-300 reactor significantly increases its safety and reduces the probability of the release of the radioactive products and environmental pollution as a result of an accident. The possibility of closing the nuclear fuel cycle, the absence of greenhouse gas emissions and safety make nuclear energy one of the most promising and environmentally friendly.

5. FURTHER RESEARCH

To clarify this estimative calculation in the mode of outage, establishment of natural circulation in the core with fuel elements with eutectic in the contact layer should be explored in particular. It is also necessary to consider the effect of the lead sublayer on the neutronic processes in the reactor: on the non-uniformity coefficients and reactivity.

Certainly, the new experimental results are necessary for the further development of a fuel cell with a liquid metal sublayer. All of the considered lead-based materials provide an equally effective temperature reduction from the point of view of thermal physics, but many factors must be taken into account. The compatibility of materials with the fuel and the structural material of the fuel shell – EP823 steel is a very important matter, the effect of ionizing radiation on them shall also be taken into consideration in choosing a sublayer material. For example, the lead-bismuth layer has a potential drawback – conversion of bismuth into polonium, which has high radioactivity and lead complications in the reprocessing of fuel.

The calculation of the profitability of this layer is ambiguous and gives contradictory results from different experts. It is difficult to assess the economic benefit from this type of fuel rod because of problems in its manufacture: high viscosity of lead, the absence of methods for automating the manufacture of fuel rods and technologies for repairing fuel elements, which have different violations, as well as the lack of methods for monitoring the continuity of the inner fuel rod sublayer. It is also necessary to study the advantages of the eutectic contact layer, considering the optimization of the reactor core and a decrease in fuel consumption due to an increase in its burnup and capacity factor, on the one hand, and the complication of the technology for manufacturing fuel elements, on the other [1].

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