SIMULATION OF THE BEHAVIOR OF FUEL ELEMENTS IN A VVER-440 FOR HIGH BURNUP (USING REACTOR NO. 3 OF THE KOL'SKAYA NUCLEAR POWER PLANT AS AN EXAMPLE)

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In this paper, we present the results of simulating the behavior of fuel elements in two fuel assemblies which were in service within the 5th to 8th and 5th to 9th fuel loads of the core of Reactor 3 of the Kol'skaya Nuclear Power Plant and reached uranium burnup of -46 MW-days/kg in 4 years and -49 MW-days/kg in 5 years, respectively.

In this investigation, we used the PIN-mod2 thermophysical calculation program [1], designed for simulating the behavior of VVER fuel assemblies under quasisteady-state conditions. The program takes into account the change in the service conditions (linear thermal loading, temperature of the coolant, cooling methods, fast neutron flux), construction and technological parameters, and also the basic processes which occur during operation and affect the behavior of the fuel element. These include:

- radiation-induced contraction, thermal expansion, swelling, creep, and cracking of the fuel;
- creep, thermal expansion, deformation and irradiation growth of the cladding;
- mechanical interaction between the fuel and the cladding;
- release of gaseous fission products from the fuel and the resulting change in the amount and composition of the gaseous medium under the fuel element cladding, its thermal conductivity, and the length of the temperature jump at the gap or contact between fuel and cladding;
- mixing of the gaseous medium under the cladding;
- change in the energy generation density over the radius of the fuel and the resulting nonuniformity of burnup over the radius, including the surface so-called rim effect [2];
- change in the thermal conductivity of the fuel as a function of temperature and burnup;
- possible rearrangements of the structure and melting of the fuel, change in the size of the central hole;
- change in the free volume under the cladding, including as a result of its elongation, etc. In the PIN-mod2 program, in simulating the behavior of the fuel and the cladding we represent them as coaxial cylinders having constant initial geometric and structural parameters over the height.

Both fuel assemblies were fabricated in May 1986. The fuel was sintered pellets of uranium dioxide, enriched up to 4.4% in 235U. The pellets were unbeveled. The alloy Zr-1% Nb was used as the material for the cladding and the end caps of the fuel elements. The shroud tube was made of Zr-2.5% Nb alloy; the upper and lower head of the assembly and the spacer grids were made of Kh18N10T steel. The coolant temperature at the reactor inlet and the heating of the coolant in different operating periods were no greater than 265°C and 30°C respectively. The pressure in the first loop was 12.3 MPa.

The emergency shutdown AZ-1 button was pushed 15 times in the 5th to 8th fuel cycles but never in the 9th fuel cycle. According to the results of inspection of the hermetic sealing of the fuel element cladding, the fuel assemblies under consideration did not contain any fuel elements with broken seals.


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Fig. 1. Graph of the load for Reactor 3 of the Kol'skaya Nuclear Power Plant.

We especially note that the second fuel assembly in the fourth year of operation (the 8th cycle) was located next to the fuel assembly of the 6th group of control and safety rods (working), which led to significant deformation of the energy generation along the height.

In order to plot the power history of the fuel elements (the time dependences of the distribution of linear power, fast neutron flux, and coolant temperature along the height of the fuel element), we used the results of neutron and physical calculations (energy generation along the height for the fuel elements and the fuel assembly, heating of the coolant in the fuel assembly) in effective operating days and the operating history of the reactor facility (the power, the coolant temperature at the inlet). In Fig. 1, we present a simplified graph of the load for Reactor No. 3 of the Kol'skaya Nuclear Power Plant in the period from the beginning of the 5th to the end of the 9th fuel cycle; the periods between the planned yearly refuelings are excluded.

The power history and the extent of burnup of the fuel in the fuel elements and the fuel assemblies were calculated using data from the operation of Reactor No. 3 of the Kol'skaya Nuclear Power Plant at the Gidropress Experimental Design Institute and the Institute of Nuclear Reactors, Russian Science Center, Kurchatov Institute using the BIPR-7 and PERMAK programs [3]. In the calculation it was assumed that the mass of uranium dioxide in the fuel assembly was equal to the nominal value, 4.4% enrichment in $^{235}$U. The calculated average burnup over the fuel assembly was (over the years of operation) $9.4$ ($9.6$) -- $24.1$ ($24.3$) -- $34.1$ ($34.6$) -- $45.88$ ($46.2$) MW-days/kg for the first fuel assembly, $5.4$ ($5.6$) -- $18.4$ ($18.6$) -- $28.7$ ($29.2$) -- $41.3$ ($41.9$) -- $49.33$ ($50.4$) MW-days/kg for the second fuel assembly. In parentheses, we give the data for calculations for the Kol'skaya Nuclear Power Plant.

In connection with the spread in structural and geometric parameters of the fuel pellets, the clads, and the fuel element as a whole, we carried out the calculation for several variants according to the initial effective fuel-cladding gap (the gap taking into account the possible decrease in the pellet diameter due to radiation-induced contraction of the fuel), encompassing (with a large margin) the possible states of the actual fuel elements. We carried out calculations for the maximum, average, and minimum effective gap variants (Table 1). We note that the maximum or minimum effective gap may not be realized over the entire fuel element (in this case, in particular, the requirements of the normative documentation with respect to the mass and length of the fuel column will not be satisfied), but it is possible, for example, that pellets having the parameters of the maximum effective gap variant may be located where the cladding has an inner diameter corresponding to the same variant. In connection with the smallness of the heat leakage between pellets in the axial direction [4] in such a pellet, the maximum temperature at the beginning of operation will be close to the value calculated using the maximum effective gap variant. As the actually most unfavorable case for the fuel element (with respect to the release of gaseous fission products and the gas pressure under the cladding), we considered the highest actual effective gap variant.

Transferring the calculation results for different variants to real fuel elements, let us consider the following example. In some fuel element, let the effective fuel-cladding gaps be distributed equally over the maximum and minimum effective gap variants. Compared with the average gap variant, the increase in the release of gaseous fission products from part of the fuel of the first variant will be greater than the decrease in the release from part of the fuel of the second variant. Therefore as the cases closest to reality with respect to the calculated gas pressure and the release of gaseous fission products, we should take the case intermediate between the average and highest actual effective gap variants, aiming at calculations of the local characteristics of the fuel element (the maximum temperature at the beginning of operation, the mechanical interaction...