ESTIMATION OF FUEL NUCLIDE COMPOSITION INFLUENCE ON FUEL LIFETIME OF REACTOR UNIT KLT-40S

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ОЦЕНКА ВЛИЯНИЯ НУКЛИДНОГО СОСТАВА ТОПЛИВА НА ДЛИТЕЛЬНОСТЬ ТОПЛИВНОЙ КАМПАНИИ РЕАКТОРНОЙ УСТАНОВКИ КЛТ-40С

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Аннотация. *В работе диффузионно-возрастное приближение применено для оценки следующих параметров ядерного реактора: эффективный коэффициент размножения, коэффициент воспроизводства, кампания ядерного реактора. Данная методика применена для четырех топливных композиций:* $(U^{238}+U^{235})O_2$; $(U^{238}+Pu^{239})O_2$; $(Th^{232}+U^{235})O_2$; $(Th^{232}+U^{233})O_2$. *Содержание делящегося изотопа в каждой композиции составляло 18,6%. Согласно результатам, использование композиции* $(Th^{232}+U^{233})O_2$ *продлить топливную кампанию на 30%.*

Introduction. Fuel lifetime is usually a very crucial parameter for low power nuclear reactors used on floating power plants (FPP). These FPPs are potentially able to solve problems of energy supply in remote regions of the Federation. Large areas are situated in places with the lack of access to the centralized electric grid. Therefore, demand in independent and standalone power plants is very high. Nuclear power is one of the reasonable solutions to the problem. Several designs of nuclear power plants were invented to produce heat and electric energy in remote areas.

In order to justify the use of nuclear power plants, the frequency of core refueling must be estimated. This parameter is called fuel lifetime and it depends on two other parameters: breeding ratio and effective multiplication factor. Moreover, we need to keep an enrichment of fuel as low as possible to ensure safety, security, and the nonproliferation of nuclear materials.

The present study is devoted to the possibility of fuel lifetime increase for the KLT-40S reactor unit by changing the fuel composition and fuel kernel diameter. The diffusion-age neutron transport model was combined with the multi-group system of equations in order to estimate k_{eff}, BR, and fuel lifetime for 4 different fuel compositions: $(U^{238}+U^{235})O_2$; $(U^{238}+Pu^{239})O_2$; $(Th^{232}+U^{235})O_2$; $(Th^{232}+U^{233})O_2$. Fissile isotope concentration for each composition is equal to 18,6%.

Research methods. Two main parameters influencing fuel lifetime are effective multiplication factor (k_{eff}) and breeding ratio (BR). The higher these values, the higher would be fuel lifetime. K_{eff} value could be increased by raising the enrichment, however, it forces BR to decrease due to lower content of breeding material. The simultaneous increase of these two parameters could be achieved by change of fissile and breeding materials.

The calculation starts from the determination of nuclear concentrations of each material in reactor core:

$$N_i = \frac{\rho_i \cdot N_A}{\mu_i};$$

where N represents the nuclear concentration of i-th material (cm^{-3}); i refers to the material; ρ signifies the density of material (g·cm^{-3}); μ shows the molar mass (mol^{-1}·g).

When heterogeneous concentrations are calculated, we need to homogenize the reactor core. To do that, we calculate volumes of each part (fuel, cladding, burning absorbers, coolant, and the volume of a fuel assembly. After that concentrations are normalized on the volume of a fuel assembly. For example, the concentration of elements in fuel is multiplied by volume of fuel and divided by volume of fuel assembly. This step makes all elements mixed inside the reactor core to get rid of heterogeneous effects.

In order to describe neutron interactions with materials, we have applied neutron transport model based on the system of 26 multigroup equations of diffusion [1]:

$$-D^i \cdot B_i^2 \cdot \Phi^i - \Sigma_a^i \cdot \Phi^i - \sum_{k=i+1}^{I} \Sigma_R^{i\to k} \cdot \Phi^i - \sum_{k=1}^{i-1} \Sigma_R^{k\to i} \cdot \Phi^k + \varepsilon^i \cdot \sum_{k=1}^{I} \nu_f^k \cdot \Sigma_f^k \cdot \Phi^k = 0,$$

where i represents the number of the neutron group (the total number of groups: $I = 26$); k refers to the number of the neutron groups; Φ^i and Φ^k show the neutron flux in the k-th and i-th groups (cm^{-2}s^{-1}), respectively; $\Sigma_R^{i\to k}$ and $\Sigma_R^{k\to i}$ denote the macroscopic cross-sections of the transition of neutrons from the i-th group to the k-th group and from the k-th group to the i-th group (cm^{-1}), respectively; and ε^i signifies the probability that a neutron will be in the i-th group immediately after fission.

The present system made of 26 equations was solved by means of the iterative approach. During the zeroth iteration, the source of neutron was approximated as 1. 0-th iteration gives values of neutron fluxes, which could be used in the first iteration to calculate proper neutron source. Calculations were conducted until the difference between fluxes in the last two iterations becomes less than 1% [2].

We have made calculations of four fuel compositions: $(U^{238}+U^{235})O_2$; $(U^{238}+Pu^{239})O_2$; $(Th^{232}+U^{235})O_2$; $(Th^{232}+U^{233})O_2$. Each fuel composition was dispersed in silumine matrix (Al + 10% Si). Calculations were carried out for the reactor at working conditions with regards to temperature effects 26th neutron group (for each element), self-shielding effects (for each element) and Westcott factors (Th^{232}, U^{233}, U^{235}, U^{238}, Pu^{239}, Pu^{240}, Pu^{241} isotopes).

Fuel lifetime was calculated by solving a set of differential equations governing the production and loss of isotopes due to neutronics. We have taken into account the burn-up of primary fissile nuclides (U^{235} and U^{233}), and production and burn-up of secondary nuclides (Pu^{239}, Pu^{240}, Pu^{241}, fission products of U^{233}, U^{235}, and Pu^{239}). The time step for integration was chosen equal to 50 effective days. At each step, the value of $k_{\rm eff}$ was found in two cases: with neutron absorber compensation (e.g. shim rods of boron carbide or boric acid) and without it. Absorber concentration was chosen such as reactor would work in critical mode (k_{eff}≈1,00001) and all excessive reactivity would be compensated. Fuel lifetime ends at the moment when k_{eff} without absorber becomes lesser than 1.

Results. According to our findings, the highest values of BR and k_{eff} are achieved for $(Th^{232}+U^{233})O_2$ fuel composition. Table 1 shows values of several parameters for each fuel composition. $(Th^{232}+U^{233})O_2$ composition has the highest value of k_{eff} and lowest value of BR. However, the effect of BR on fuel lifetime is slightly lesser, than that of k_{eff}. Figure 1 represents four lines corresponding to the dependency of k_{eff} on time for each fuel composition.

Table 1

Multiplication and breeding properties of the KLT-40S reactor core for different fuel compositions with 18,6 % enrichment at the beginning of fuel lifetime

	k_{eff}	BR	$\Sigma_c^{breeding}$, cm^{-1}	$\Sigma_a^{fissile}$, cm^{-1}	$\Sigma_f^{fissile}$, cm^{-1}	Fuel lifetime, y
$(U^{238}+U^{235})O_2$	1,33614	0,17730	0,00300	0,01692	0,01327	1,781
$(U^{238}+Pu^{239})O_2$	1,28050	0,17269	0,00320	0,01851	0,01147	1,096
$(Th^{232}+U^{235})O_2$	1,27645	0,15663	0,00269	0,02079	0,01325	1,781
$(Th^{232}+U^{233})O_2$	1,50245	0,14329	0,00261	0,02170	0,01505	2,603

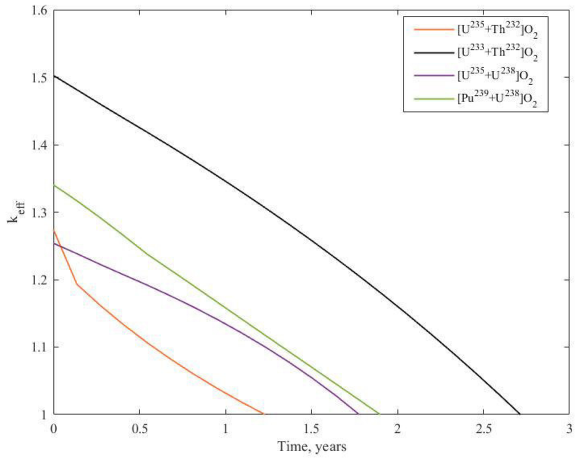

Fig. 1. Effective multiplication factor dependency on time for each fuel composition

Conclusion. A diffusion-age model was applied in order to estimate the fuel lifetime of four fuel compositions. Our findings state that thorium-uranium fuel provides longer fuel lifetimes (up to 30%). Plutonium-uranium composition provides the shortest fuel lifetime due to higher values of microscopic cross-sections of absorption for thermal and epithermal neutrons for Pu^{239} in comparison with U^{235} and U^{233}. Change of U^{238} to Th^{232} isotope leads in a decrease of effective multiplication factor because Th^{232} has lesser fission cross-sections for fast neutrons due to higher fission threshold.

Although thorium fuel composition provides long fuel lifetime, there are additional studies to carry out. For example, fuel lifetime dependency on fuel kernel diameter, the fracture strength of cladding during extended lifetime and simulation based on such precision software as Serpent, MCU, MCNP and other neutronic codes.

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