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The dependence of the nuclide composition of the fuel core loading on multiplying and breeding properties of the KLT-40S nuclear facility

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Abstract

The paper describes a method for determining the effective neutron multiplication factor and the breeding ratio for the KLT-40S at operating conditions. The main reactor design features are presented which are necessary for calculations. It is shown that the type of fertile nuclides has no virtual effect on the formation of the neutron flux density spectrum. The contribution of each neutron group to the fission reaction rate at the design fissile nuclide content of 18.6% is determined. The dependences of average values of macroscopic fission cross-sections, absorption of fissile nuclides and radiation capture of fertile nuclides on the fissile nuclide content in nuclear fuel are obtained. Averaging of cross-sections was carried out over neutron flux density spectra.

As a result, dependences were obtained of the effective neutron multiplication factor and breeding ratio on the fissile isotope content for various fuel compositions of uranium and thorium cycles at the KLT-40S campaign start-up. In terms of the effective multiplication factor, the best result is obtained in the ^{232}Th+^{233}U composition when the fissile isotope content is more than 5%, and in the ^{238}U+^{239}Pu composition when the the fissile isotope content is less than 5%. In terms of the breeding ratio, the best result is obtained in the ^{232}Th+^{235}U composition when the fissile isotope content is up to 10%; if it exceeds 10%, its values are relatively the same for the ^{232}Th+^{233}U and ^{232}Th+^{235}U and ^{238}U+^{235}U compositions.

Thus, the most efficient composition at the nuclear fuel campaign start-up is ^{232}Th+^{233}U with the ^{233}U nuclide content more than 5% due to very high values of the effective neutron multiplication factor.
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Keywords: KLT-40S reactor facility; Effective multiplication factor; Breeding ratio; Uranium fuel cycle; Thorium fuel cycle.

Problem state

Due to its huge size, diverse climatic conditions and, not infrequently, hard-to-reach areas, the Russian Federation is characterized by uneven population density and different economic levels in individual regions. Large territories are outside the zone of centralized power supply, and energy is supplied to these remote regions by autonomous sources of fossil fuel, the delivery of which is very costly and its exploitation seriously damages the environment [1–5].

It is estimated that for remote hard-to-reach regions, nuclear power is a reasonable alternative to traditional fuel-based systems. Today, there are several projects (in various implementation phases) of nuclear power plants designed to generate heat and electricity in underdeveloped regions [6–10].

One of the most important efficiency indicator for nuclear energy sources located in remote regions is the operating time before refueling which is characterized by the multiplication factor and breeding ratio, with that, for the purposes of security and non-proliferation, fuel enrichment should be as low as possible.

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Developed by OKBM Afrikantov, the KLT-40S reactor facility modular core is based on the icebreaker KLT-40 channel type core which has proven its reliability [11].

The KLT-40S reactor facility is used in the floating power unit (FPU) of the low capacity nuclear power plant (ATES-MM) intended to supply the country's remote regions with heat and electricity.

For now, the reactor campaign is within 2.5–3 years, which negatively affects the ATES MM economic performance. One of the possible ways to increase both the fuel burn-up depth and campaign life at the reactor design capacity is to use a fuel composition with a high effective multiplication factor (hereinafter "the multiplication factor", k_{eff}) and breeding ratio (BR) of nuclear fuel. Therefore, the goal of this work is to develop a method for determining the effective neutron multiplication factor and the breeding ratio for the KLT-40S at the operating parameters.

The k_{eff} and BR are estimated in cases where fuel compositions with various fissile and fertile nuclides are used without changing the outer diameter of fuel elements and structure of fuel assemblies (FA) as a whole.

Estimated multiplication factor and breeding ratio at different nuclide composition of KLT-40S fuel

To be estimated, at the fuel campaign start-up, the breeding ratio was defined as the ratio of fissionable nucleus formation rate to the burn-up rate [12]:

$$\text{BR} = \frac{\overline{\Sigma_c^{\text{breed}}}}{\overline{\Sigma_a^{\text{fiss}}}}, \tag{1}$$

wherein the numerator and denominator are the spectrum-averaged macro-cross-section values of radiation neutron capture by fertile nuclides ($\overline{\Sigma_{\text{c}}^{\text{breed}}}$) and neutron absorption by fissile nuclides ($\overline{\Sigma_{\text{a}}^{\text{fiss}}}$):

$$\overline{\Sigma_{\text{c}}^{\text{breed}}} = \sum_{i=1}^{I} \Sigma_{\text{c}}^{\text{breed}i} \cdot \delta^i, \tag{2}$$

$$\overline{\Sigma_{\text{a}}^{\text{fiss}}} = \sum_{i=1}^{I} \Sigma_{\text{a}}^{\text{fiss}i} \cdot \delta^i, \tag{3}$$

where $\Sigma_{\text{c}}^{\text{breed}i}$ and $\Sigma_{\text{a}}^{\text{fiss}i}$ are the macro-cross-sections for ith neutron group, radiation neutron capture by fertile nuclides, and absorption by fissile nuclides, respectively, cm^{-1}; $\iota\delta^i$ is the flux density fraction of ith neutron group:

$$\delta^i = \frac{\Phi_i}{\sum_{k=1}^{I} \Phi_k}. \tag{4}$$

The value of k_{eff} was estimated using the ratios:

$$k_{eff} = \frac{\overline{\nu_f \cdot \Sigma_f}}{\overline{\Sigma_a} + \overline{D \cdot B^2}}, \tag{5}$$

$$\overline{\nu_f \cdot \Sigma_f} = \sum_{i=1}^{I} \nu_f^i \cdot \Sigma_f^i \cdot \delta^i, \tag{6}$$

where $\nu^i{}_f$ is the average neutron number per fission event in the ith neutron group; $\Sigma^i{}_f$ is the fission macro-cross-section for neutrons in the ith neutron group, cm^{-1}.

$$\overline{\Sigma_a} = \sum_{i=1}^{I} \Sigma_{\text{a}}^i \cdot \delta^i, \tag{7}$$

where $\Sigma^i{}_a$ is the absorption macro-cross-section for neutrons in the ith neutron group, cm^{-1}.

$$\overline{D \cdot B^2} = \sum_{i=1}^{I} D^i \cdot B_i^2 \cdot \delta^i, \tag{8}$$

where D^i is the diffusion constant for neutrons in the ith neutron group, cm; $B_i{}^2$ is the geometric parameter for neutrons in the ith neutron group, cm^{-2}.

To determine δ^i, the multi-group method was used for solving a system of one-velocity kinetic neutron balance equations to an age-diffusion approximation (stationary problem) [13,14]:

$$-D^i \cdot B_i^2 \cdot \Phi^i - \Sigma_a^i \cdot \Phi^i - \sum_{k=i+1}^{I} \Sigma_R^{i\to k} \cdot \Phi^i + \sum_{k=1}^{I-1} \Sigma_R^{k\to i} \cdot \Phi^k + \varepsilon^i \cdot \sum_{k=1}^{I} \nu_f^k \cdot \Sigma_f^k \cdot \Phi^k = 0, \tag{9}$$

where i is the considered neutron group index, the total number of groups $I=26$; k is a neutron group index; Φ^i, Φ^k are the neutron flux densities in the ith and kth groups, cm^{-2}s^{-1}; $\Sigma_R^{i\to k}$, $\Sigma_R^{k\to i}$ are the macro-cross-sections of neutron escape from the upper ith group into the lower kth group and, correspondingly, from the lower kth group into the considered ith group, cm^{-1}; ε^i is the probability for a fission neutron to get into the i-group.

The system of equations was solved iteratively. For the ith group on the jth iteration:

$$\Phi_j^i = \frac{\varepsilon^i \cdot \sum_{\substack{k=1 \\ k\neq i}}^{I} \nu_f^k \cdot \Sigma_f^k \cdot \Phi_{j-1}^k + \sum_{k=1}^{I-1} \Sigma_R^{k\to i} \cdot \Phi_j^k}{D^i \cdot B_i^2 + \Sigma_a^i + \sum_{k=i+1}^{I} \Sigma_R^{i\to k} - \varepsilon^i \cdot \nu_f^i \cdot \Sigma_f^i}. \tag{10}$$

To start the iteration process at zero iteration, the number of neutrons produced in the second generation during the first generation neutron-induced fission was set equal to unity:

$$\sum_{k=1}^{I} \nu_f^k \cdot \Sigma_f^k \cdot \Phi^k = 1. \tag{11}$$

Then, the neutron flux densities at zero iteration are:

$$\Phi_0^i = \frac{\varepsilon^i + \sum_{k=1}^{I-1} \Sigma_R^{k\to i} \cdot \Phi_0^k}{D^i \cdot B_i^2 + \Sigma_a^i + \sum_{k=i+1}^{I} \Sigma_R^{i\to k}}. \tag{12}$$

The calculation was performed for the reactor facility at the operating temperature, taking into account the correction

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