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Research

FEATURES OF FUEL BURNUP CALCULATIONS FOR IRT-T REACTOR USING MCU-PTR CODE

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ABSTRACT

The article considers computation methods of the kinetics of changes of fuel nuclide composition vs. fuel burnup. Calculations were carried out at different intervals of operating time at a constant level of power of the IRT-T research reactor for different types of fuel assemblies. The analysis of the reaction rate of absorption for the isotopes included in the fuel composition is given. It is shown that the use of piecewise linear interpolation is the most preferred in order to calculate the long fuel campaigns as there is a high convergence of calculation data.

INTRODUCTION

Currently, the issue of determining the nuclide composition of irradiated fuel in fuel assemblies of nuclear reactors is an important aspect of ensuring operation security of facilities. During operation of the nuclear reactor, huge number of nuclides affecting the neutron parameters of the reactor core are produced in fuel elements. Concerning the research facility, one of the most important operating parameters is to maintain the set characteristics at the constant level for the entire period of continuous operation, then simulation of the changes in the nuclide composition in the fuel assembly is necessary to study neutron-physical characteristics of the operation of research nuclear reactor (Aniki, *et al.*, 2015; Bakhvalov, 2004).

Approximation methods for evaluative calculation can be applied in different areas such as simulation of changes in the nuclide composition of the fuel assembly, or, for example, in terms of activity for the FA. This is also suitable for a quick assessment of the nuclide composition of the fuel in contact with water in case of depressurization of the assembly as well as in the simulation of the interaction of iodine, tellurium and cesium with shell materials during long shutdowns. During the simulation of the nuclide composition change, the interaction cross-sections of nuclides and the density of neutron flux are approximated by using different interpolation methods.

Interpolation is a way to approximate or exact finding of any quantity of known individual values of the same or other variables associated with it.

The simplest interpolation methods are piecewise constant (each interval between the points is a constant), and piecewise linear interpolation, in which all points of the table function connected line segments as shown in Fig. 1 (Naymushin, *et al*., 2015).

The disadvantage of a piecewise constant interpolation is that the obtained interpolation function is not a continuous function, while the linear interpolation function is continuous. Accuracy of piecewise linear interpolation will be more than in the case of a piecewise constant interpolation.

RESULTS AND DISCUSSION

Research nuclear reactor IRT-T – pool reactor with water as coolant, moderator and top biological protection (Varlachev, *et al*., 2011). 8-tube and 6-tube fuel assemblies of IRT-3M type constitutes the reactor core (Naymushin, *et al*., 2015).

To calculate the concentrations of nuclides produced in nuclear fuel while burning up, the MCU-PTR program with estimated nuclear data library ENDF/B-VII.0 was used. In the calculations piecewise constant interpolation of the cross-sections interaction of nuclides and neutron flux density was used. In the calculations were derived concentration

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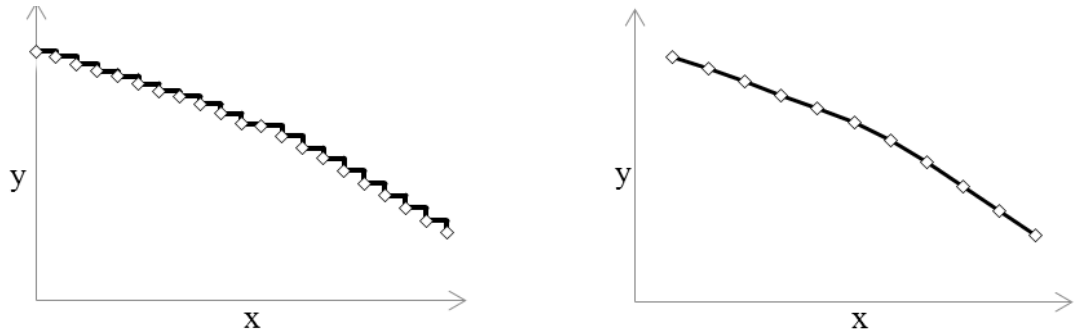

Fig. 1 Interpolation: left – piecewise constant; right – piecewise linear.

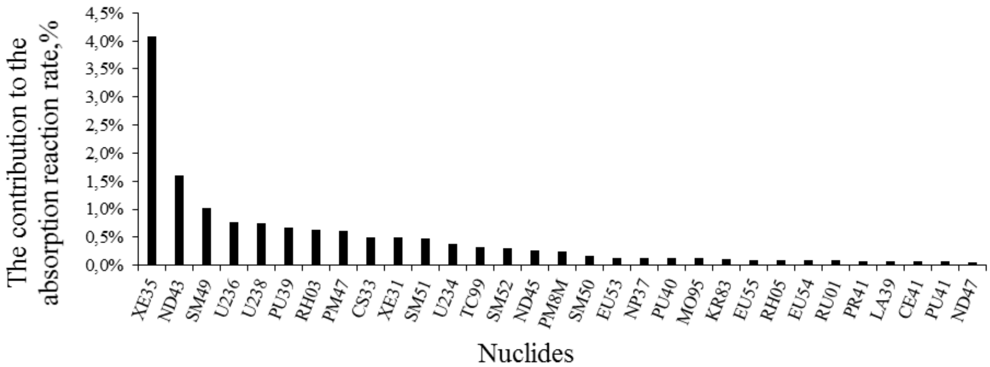

Fig. 2 Isotopes and their contribution to the absorption reaction rate (The figure does not show U-235 which contribution is 79.47% and the Al-27 with contribution of 6.36%).

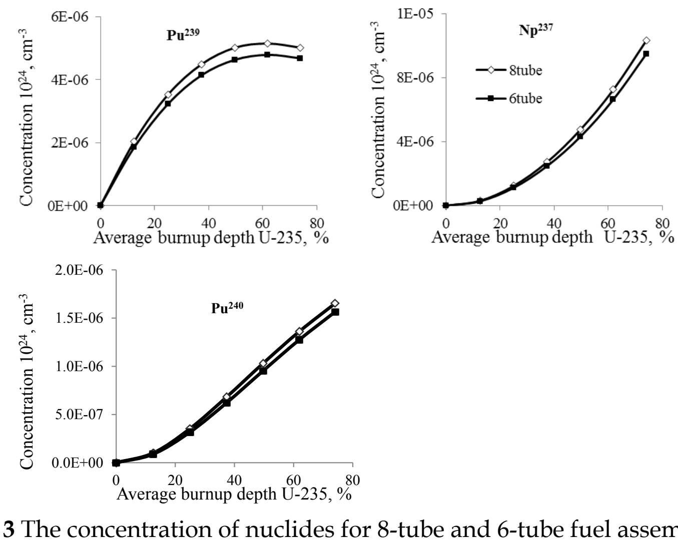

Fig. 3 The concentration of nuclides for 8-tube and 6-tube fuel assemblies.

of nuclides contained in the nuclear fuel in the time of its burnup, after 20, 50, 100, 200 and 600 effective days of operation of the IRT-T reactor for 8-tube and 6-tube IRT-3M fuel assemblies. Out of all nuclides were selected those whose contribution to the absorption reaction was more than 0.1% (Fig. 2). Thus, the total contribution of all nuclides was accounted for 99.82%.

For each nuclide an analytical dependence of the concentration change on the burnup depth of U-235 was obtained, as the only reliably controlled neutron-physical parameter is burnup depth, measured as a percentage of initial charge of U-235. Based on obtained dependences, were built functions for the concentration change of each nuclide in the fuel assembly. Fig. 3 shows these dependences, which characteristic form corresponds to the considered nuclides.

Analytical dependence for burnup calculating with the step of 20-day effective has the smallest error, but the calculation takes long, nearly 230 hours (hereinafter the performance of calculation station

– 134 GFlops). By increasing the step of burnup calculating the measurement error increases (Fig. 4). As seen from the Fig. 4, a step of calculating of burnup up to 100 effective days is the most preferred for further research, since there is an optimal ratio between the duration of the calculation (47 hours) and error (not more 7%). At step of calculation up to 50 effective days error reaches 4%, at the calculation time of 88 hours 200 effective days – 15 % and 27 hours 600 effective days – 32% and 12.5 hrs.

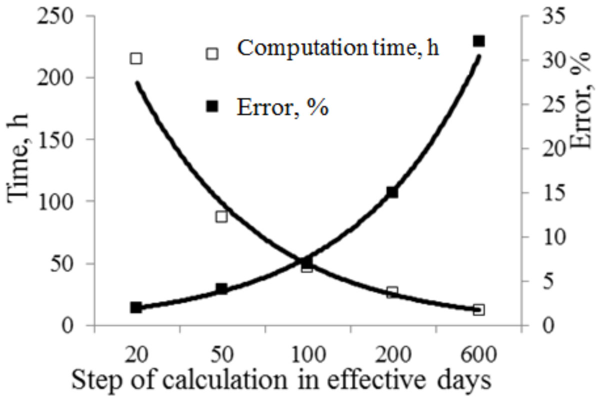

Fig. 4 The dependence of the error and the time of calculation on the calculation step.

Comparing the dependencies with burnup steps up to 20 and 100 effective days for most nuclides calculation error was no more than 2%, for the rest of nuclides – 4% to 7%, as their concentrations are relatively low (Table 1).

Table 1. Comparison of concentrations of nuclides in the step of calculating burnup 20 eff. day and 100 eff. days

Burn up of U-235, %	10	20	30	40	50	60	70	80
The relative error in the U-238 concentration, %	0.02	0.05	0.08	0.13	0.19	0.27	0.37	0.49
The relative error in the Sm-149 concentration, %	1.65	2.02	2.43	2.89	3.36	3.82	4.21	4.41
The relative error in the Pu-239 concentration, %	5.33	3.66	3.61	3.99	4.58	5.32	6.15	7.07

Analysis of calculated data for six-tube and eight-tube fuel assemblies showed that for the majority of radionuclides the difference between the concentrations was not more than 3%. For some nuclides difference increased by more than 5%, for example, for Pu-239 the difference is 6% for Np-237 – 8% for Pu-240 – 10% (Fig. 3). This error occurs because of the fact that the concentrations of these nuclides are relatively small.

Calculation by means of mcu-ptr program using the method of piecewise linear interpolation

In this method was used piecewise linear interpolation for nuclides cross-section interaction and neutron flux density (Alekseev and Gomin, 2011). In the calculations were derived concentration of nuclides contained in the nuclear fuel in the time of its burnup, after 20, 50, 100, 200 and 600 effective days of operation of the IRT-T reactor for 8-tube and 6-tube IRT-3M fuel assemblies. By the obtained data were recovered analytical dependences of the nuclides concentration on the fuel burnup.

For the analysis, the obtained dependences were compared with the dependence of calculation step of burnup up to 20 effective days, because it is the most accurate. The analysis showed that by the step of calculation up to 20, 50, 100 and 200 days the difference between the effective concentration was not more than 1.5%. By the step of calculation up to 600 effective day the error for most of the nuclides was no more than 3%, which corresponds to an error in the step of calculation step up to 50 effective days without an approximation of nuclides sections and neutron flux density. For some nuclides difference is increased by more than 5%, for example, for the Pu-239 is the difference 6.5% for Np-237 – 7.4% of Pu-240 – 7.5% for Pm-147 – 4.2% for Pm-148m – 5.8%. This error occurs because of the small values of these nuclides. Using the method of piecewise linear interpolation, we can assume that the step of 600 effective days has the optimal ratio of accuracy and computation time for further research.

CONCLUSION

Analytical dependences of changes in the nuclides concentrations produced in the fuel assembly in the process of burning up were obtained. When the neutronic calculations are carried out for different layouts of IRT-T reactor, it is usually known only the depth of fuel burnup at the initial point of time for those layouts. These relationships can be used to set detailed nuclide composition of the fuel assemblies for any burnup in them.

It is shown that the optimum ratio between the accuracy of calculation by using the method of piecewise constant interpolation and its time is reached at calculation step of up to 100 effective days. Calculations time in this step is ~47 hours. For faster and more accurate calculation the method of piecewise linear interpolation can be used, since even with the step of 600 effective days for most nuclides the difference of concentrations is up to 3%. The computation time during this step is 21 hours, which is 2 times faster than during the step of 100 effective days using the method of piecewise constant interpolation.

The difference between the concentrations for eight-tube and six-tube fuel assemblies for most nuclides reaches 3%. For some nuclides, the difference reaches 10%, which suggests that it is impossible to use a universal scheme to calculate the value of burnup. It is necessary to consider differences in the spectra of neutrons in 8-tube and 6-tube fuel assemblies of IRT-T reactor when modelling various reactor's life times.

The using approximation of cross-sections interaction of nuclides and neutron flux density allows using nuclear concentration of fuel areas for all depths of burnup.

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