Investigation of radiation levels in the region of fuel-transfer cask for transportation of spent fuel of light-water reactors

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Abstract. The paper formulates the main problems associated with research on transmutation, which should be paid attention to by today's young researchers. The processes of production of hazardous nuclides during transmutation in reactor facilities are considered. The goals of transmutation and the choice of nuclides to be transmuted are discussed. The concept of radiotoxicity is explained as a measure of the radiological hazard of radioactive nuclides, based on the maximum permissible concentration of nuclides according to the IAEA standards. The problem of the formation of secondary radioactive nuclides in nuclear fuel during generation of neutrons for transmutation is discussed. The advantages and disadvantages of various methods of transmutation in nuclear installations are considered: inclusion of transmutable nuclides in nuclear fuel in fast reactors, transmutation in specialized thermal and fast transmutation reactor installations and ADS systems. The problem of accumulation of highly radioactive actinides in a transmutation facility during long-term transmutation and the problem of a potential hazard of the transmutation facility itself are discussed. The unacceptability of application of common-type power reactors for the transmutation of long-lived fission products is demonstrated.

1. Introduction
Thermal neutron power reactors of VVER-1000 type are currently the most popular in the nuclear power industry of Russia. Spent fuel in the form of fuel assemblies is transported to the place of long-term storage or radiochemical processing. Fuel transportation casks are used for transportation. Fuel-transfer casks should provide adequate protection of the environment against all types of radiation from spent fuel located inside the container.

In this paper, calculation study of the properties of the most typical cask used for transportation of spent fuel assemblies in Russia is performed. For this purpose calculations of isotopic composition of uranium nuclear fuel at typical levels of burnup were carried out, data on radiation levels inside the cask with the spent fuel assemblies were obtained, radiation levels at different distances from an external surface of the cask were calculated. These data are important for justification of safe transportation of spent fuel from the point of view of the safety requirements [1].
2. Calculation model of the elementary cell of VVER-1000 reactor

To calculate the burnup of uranium isotopes and accumulation of plutonium isotopes in uranium nuclear fuel, a model of a hexagonal elementary cell representing a fuel rod surrounded by water was used (Figure 1). The geometric dimensions of this cell are: the outer diameter of the fuel rod is 0.772 cm, the outer diameter of the zirconium shell is 0.916 cm, and the lattice pitch is 1.275 cm. In real fuel assemblies fuel pellets have a central hole with diameter of 0.15 cm. However, for ease of simulation, the central hole in the fuel pellet was not taken into account in calculations. Thus, the calculation was performed for infinite lattice consisting of identical cells. Nuclear fuel based on low-enriched uranium with enrichment of 4.95% in the form of uranium dioxide UO\textsubscript{2} was considered in calculations (Table 1).

![Figure 1. Elementary cell for calculations: 1 – fuel, 2 – cladding, 3 – coolant.](image)

| Nuclide | Concentrations, $10^{22}$ cm$^{-3}$ | T, K |
|---------|-----------------------------------|------|
| **Fuel** |                                   |      |
| $^{238}$U | 1.8636                            | 1027 |
| $^{235}$U | 9.8291·$10^{-2}$                  | 1027 |
| $^{16}$O | 3.9235                            | 1027 |
| **Cladding** |                                 |      |
| Zr      | 4.23                              | 579  |
| **Coolant** |                                 |      |
| H       | 4.783                             | 579  |
| $^{16}$O | 2.391                             | 579  |

To calculate the radiation characteristics of fuel assemblies, the model of fuel assembly containing 312 fuel cells was used (Figure 1). The calculations were performed for two values of fuel burnup: 33 and 60 MWd/kg. The axial calculation was based on the fuel assembly height of 275.6 cm.

3. Programs for calculation of isotopic composition and radiation characteristics of irradiated fuel

The code SCALE together with SASH2 was used to calculate the isotope composition of the fuel during the burnup process, as well as to calculate the radiation characteristics of the irradiated fuel.

SCALE [4] is a well-known software package, which includes reliable codes and data libraries for the analysis of nuclear fuel, as well as for solving problems of radiation safety. The codes are made on a modular basis.

SAS2 is the control module for the SCALE code to provide automated calculation of fuel burnup and analysis of neutron and gamma radiation generated by spent nuclear fuel. The parameters of numerical code schemes were chosen from the experience of solving the tasks of fuel burnup in light-water reactors [5-7]. These results were used to analyze the protective properties of the transportation cask.
4. Transport package and calculation model

Transportation of spent fuel of VVER-1000 reactors at nuclear power plants is performed using transportation package of the TUK-13B type [8]. The cask is a thick-walled cylindrical vessel closed by a massive cover. Inside the cask there is the set of hexagonal pipes, designed to accommodate spent fuel assemblies. Neutron protection layer is located outside the cask. It is made of material that reduces neutron radiation from the cask. Vessel and cover are designed to remove heat from the internal contents of the cask and to reduce gamma radiation. In addition, vessel contains structural elements for safe fixation of placed fuel assemblies.

The main dependences of the dose rate of neutron and gamma radiation on the distance from the outer wall of the cask, on the fuel burnup, and on the time of cooling of spent fuel after unloading from the reactor were studied in this paper. Therefore, a simplified one-dimensional model of a cylindrical cask (figure 2) was used to calculate gamma and neutron radiation. The cask contained six irradiated fuel assemblies with a burnup of 33 or 60 MWd/kg.

![Figure 2. One-dimensional model of the cask for calculations in SCALE and SAS2.](image)

Calculation of equivalent dose rate at various points outside of the cask was performed in the program SASH2 after calculation of the isotopic composition of spent fuel and sources of neutron and gamma radiation. This calculation consisted of two stages. At the first stage, calculation of radiation transfer through the layers of the walls of the cask in one-dimensional cylindrical geometry was made. In this calculation all placed fuel assemblies inside the cask were homogenized. At the second stage, doses were calculated with account of axial dimensions.

5. Results of calculation of dose rate outside the cask

Calculations of radiation characteristics outside the cask were performed for the cask walls made of steel containing 19% chromium, 9.5% nickel, 69.5% iron, and 2% manganese [4]. The layer of water and boron was taken as neutron protection layer. Spent nuclear fuel was kept in storage for 4, 6, 8, or 15 years. After this time of storage, six fuel assemblies were placed in a cask. The dose rate of neutron and gamma radiation at different distances from the surface of the cask is shown in figures 3, 4.

Additionally, radiation dose rate was calculated for PWR and VVER reactors with enrichment $x = 4.4\%$ and 4.95%. Dose rate at different distances from the cask surface for these reactors are presented in figures 5-8.

The data presented show the following.

- When fuel burnup increases, the dose rate of neutron and gamma radiation increases (figures 7 and 8).
- When the time of storage increases, the dose rate of gamma radiation decreases, but neutron radiation decreases slightly (figures 3 and 4).

For different thicknesses of the walls of transportation casks, the contribution of different components into dose rate can vary significantly. For "thick" casks, gamma radiation due to capture reactions can play an important role. When it needs to simulate correctly transfer of radiation through the walls of the cask, it could be desirable to use 3-d programs.
Figure 3. Dependence of the dose rate of neutron radiation on the distance from the surface of the cask for burnup of 60 MWd/kg and different time of storage.

Figure 4. Dependence of the dose rate of gamma radiation on the distance from the surface of the cask for burnup of 60 MWd/kg and different time of storage.
**Figure 5.** Dependence of the dose rate of neutron radiation on the distance from the surface of the cask for spent fuel of PWR and VVER reactors with burnup of 33 MWd/kg and time of storage of 4 years.

**Figure 6.** Dependence of the dose rate of gamma radiation on the distance from the surface of the cask for spent fuel of PWR and VVER reactors with burnup of 33 MWd/kg and time of storage of 4 years.
Figure 7. Dependence of the dose rate of neutron radiation on the distance from the surface of the cask for spent fuel of VVER reactor with different burnup and time of storage of 5 years.

Figure 8. Dependence of the dose rate of gamma radiation on the distance from the surface of the cask for spent fuel of VVER reactor with different burnup and time of storage of 5 years.

6. Conclusions
Dose rates of neutron and gamma radiation at different distances from the transportation cask containing spent fuel assemblies of the VVER reactor were calculated. For calculation of fuel burnup, isotopic composition of spent fuel assemblies, and radiation doses outside the cask, the software package SCALE and SAS2 was used.

The analysis of the results demonstrated the main dependences of the dose rates of neutron and gamma radiation on the distance from the outer surface of the cask, on the burnup of spent nuclear fuel, and on the time of storage of spent fuel after unloading from the reactor.

The results obtained were used to simulate the composition of the corium of light-water reactors [9, 12].
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