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Analysis of uranium oxide fuel transmutation in Vver-1000 reactor using VISTA and Wims-D4 codes

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ABSTRACT

In this study, the front end and back end of Uranium Oxide (UO_X) fuel cycle were first calculated using nuclear fuel cycle simulation system (VISTA) code. The front end calculated values are 19595.3 ton/year for ore-U, 196 ton/year for natural-U, 196 ton/year for UF6, 111.4 metric ton separative work unit (MTSWU) for enrichment requirements, 172.211 ton/year depleted-U and 23.8 ton/year for Fresh Fuel (FF) requirements. The back end calculated values were 23.8 ton/year for Spent Fuel (SF), 0.024 ton/year for Actinide Inventory (AI) and 0.981 ton/year for Fission Product (FP) isotopes. After that the transmutation of UO_X fuel results calculated by VISTA benchmarked with WIMS-D4 code. The calculations carried out in 30,000, 35,000, 40,000 and 45,000 MWd/tHM discharge burnup stages. The comparison of results showed that variance was less than 15%.

1. Introduction

Generation of energy by nuclear power reactors leaded to production of actinide elements and radioactive nuclear waste. Assessment of the worldwide inventories of Actinide Inventory (AI) in spent fuel (SF) is important due to non-proliferation issues (for fissile isotopes of U and Pu) and radiotoxicity of long lived Actinides. The investigations of AI in nuclear reactors were reported by (Coates et al., 2011; Coates and Parks, 2010; Ahmad, 2006; Rose et al., 2011; Zheng et al., 2015; Liu et al., 2014; Şahin et al., 1999).

Also, fuel transmutations in various types of nuclear reactors such as Fixed Bed Nuclear Reactor (FBNR) reactor (Şahin et al., 2009, 2010a,b), Canada Deuterium Uranium (CANDU) reactor (Hyland and Gihm, 2011; Şahin et al., 2006, 2010a,b) and fusion hybrids reactor (Ridikas et al., 2006; Şahin et al., 2006) were studied.

VVER-1000 is a Russian type of a pressurized water reactor (PWR). A typical 1000 MW(e) VVER-1000 reactor generates about 20–25 tons of spent fuel per year (Hu, et al., 2015). Although the yield of fission product (FP) in the nuclear reactors is very high, most of it has very short half-life and decays after days to weeks of their production. Generally, the FP isotopes in the SF are ^{137}Cs, ^{90}Sr and ^{131}I. The final composition of the SF depends on different parameters such as the fuel type, chemical composition, and level of initial enrichment in ^{235}U, neutron energy spectrum of the reactor, the fuel burnup, and cooling time (Ewing, 2015; Ceyhan, 2007).

The main difference between the PWR and VVER is related to the design of the fuel assembly and the core geometry. The VVER-1000 Reactor produces 3000 MWth in maximum power which is generated from 163 hexagonal fuel assemblies. The VVER-1000 fuel materials consist of three types of uranium oxide enrichments including: 3.62%, 2.4% and 1.6%. Each one of the fuel assemblies consists of 311 fuel rods. The VVER-1000 refueling interval period is about 7000 h/year in maximum power and its burnup is 12,000 MWth/TU (Karahroudi et al., 2013).

The views of the VVER-1000 core plan and fuel assembly configuration are presented in Fig. 1. The main characteristics of the core and fuel used in VVER-1000 reactor are shown in Table 1.

First of all, we analyzed the front and back end of the nuclear fuel. The AI and FP in the VVER-1000 reactor with UO_X fuel material were calculated for some of the burnup stages whereas the IAEA's nuclear fuel cycle simulation system (VISTA) (IAEA-TECDOC-1535, 2007) and WIMS codes were used to analysis the AI and FP.

2. Methodology and materials

2.1. VISTA simulation

To calculate the AI and FP in the VVER-1000 reactor with UO_X fuel material with 30,000, 35,000, 40,000 and 45,000 MWd/tHM burnup stages, we used VISTA calculations. The general characterizations of VVER-1000 reactor with UO_X fuel which we used as input parameters in VISTA simulation are shown in Table 2

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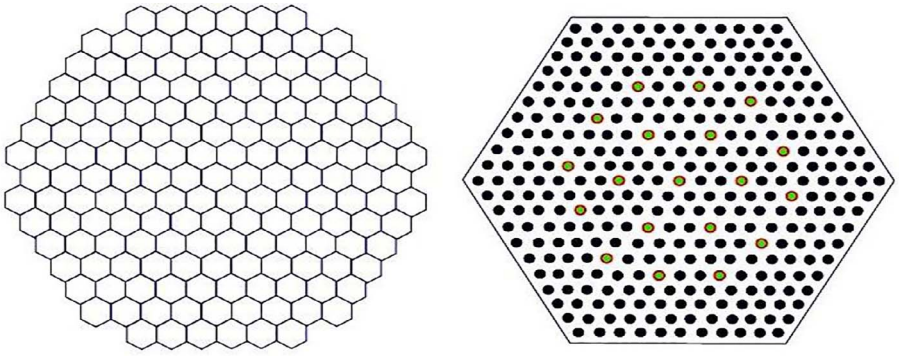

Fig. 1. The horizontal cross-section view of (a) core plan (b) fuel assembly configuration in VVER-1000.

Table 1
The VVER-1000 core specifications.

Parameter	Value
Core major characteristics	
Thermal power (MW)	3000
No. of fuel assemblies in the core	163
No. of fuel assemblies with control rods	61
Height of heating part (m)	3.53
Fuel assembly pitch (m)	0.236
Pass section of the core in the heating part (m^2)	4.17
Coolant's flow rate through the core (kg/s)	17,650
Fuel assembly major characteristics	
No. of fuel rods	312
Fuel rod pitch (mm)	12.75
No. of tubes for absorber elements	18
Length of FA active part (mm)	3530
No. of distant grids (support plates)	14
Fuel pin major characteristics	
Fuel pin diameter (mm)	9.1
Diameter of central aperture in fuel pellet (mm)	2.3
Fuel density (g/cm^3)	10.4
Enrichment of feeding fuel (%)	3.3, 4.4, 3.0, 4.0
Cladding thickness (mm)	0.69
Cladding material	Zr110
Fuel pellet diameter (mm)	7.53
Fuel material	UO_2

Table 2
The characteristic of VVER-1000 reactor with UO_X fuel.

Parameter	Value
Scenario Parameters	
Nuclear Power (MWe)	1000
Load Factor (%)	85
Thermal Efficiency (%)	32.6
Tails assay from enrichment (%)	0.3
Fuel Parameters	
Fuel Type	UO_X
Enrichment (%)	3.685
Burnup (GWd/tHM)	45

In Calculating Actinide INventory (CAIN) section of VISTA code, a general formula to calculate disappearance of a nuclide by nuclear transmutation and radioactive decay is shown as in the following:

$$\frac{dY_i}{dt} = \sum_{j=1}^{N} q_{ij}\lambda_j Y_j + \xi \sum_{k=1}^{N} f_{ik}\sigma_k Y_k - (\lambda_i + \xi\sigma_i + r_i)Y_i + F_i, i = 1,\ldots,N \quad (1)$$

where Y_i is the atom density of nuclide$i$, $N$is the number of nuclides, q_{ij} is the fraction of radioactive disintegration by nuclide j which leads to nuclide i formation, λ_j is the radioactive decay constant, ξ is the space and energy averaged neutron flux, f_{ik} is the fraction of neutron absorption by nuclide k which leads to formation of nuclide i, σ_k is the spectrum averaged neutron absorption cross section of nuclide k, r_i is the continuous removal rate of nuclide i from the system, and F_i is the continuous feed rate of nuclide i (Rezaeian and Kamali, 2016).

The deterministic calculation method of VISTA code is able to link with CAIN to evaluate the Fresh Fuel (FF), AI and FP inventory. In the current scenario, actinide radioelements were calculated in different burnup stages by using VISTA simulator. The calculated actinide group includes elements from Np (Z = 93) to Cm (Z = 96) and major components of fresh and spent nuclear fuel (U and Pu isotopes). This code was used to determine potential pressurized water reactor (PWR) spent fuel compositions (Alajo and Tsvetkov, 2011).

The main reaction chains used in CAIN to calculate the changes in isotopic calculations during the irradiation period with UO_X fuel are ^{238}U and ^{235}U. The reaction of ^{234}U is ignored because its transmutation into ^{235}U is small. Also, the actinide transmutations to each chine are calculated by:

$$\frac{dN_i}{dt} = -\sum_{i\neq j} [\lambda_{ji}^d + \sigma_{ji}^{tr}\varphi]N_i + \sum_{j\neq i} [\lambda_{ij}^d + \sigma_{ij}^{tr}\varphi]N_j \quad (2)$$

where N_i is the atomic content of ith-isotope; λ_{ji}^d is decay constant, (1/s); σ_{ji}^{tr} transmutation cross section from isotope i to isotope j, (barn) and φ is the average neutron flux, ($n/s{\cdot}cm^2$).

In overall the calculation results are divided into front end and back end section. In the front end of the cycle, FF requirements, enrichment value requirements, depleted uranium amount, conversion requirements and natural uranium requirements are calculated per year. In the back end of the cycle, SF discharge amount, AI and FP per year were calculated.

For validation of AI calculation results, we used the WIMS-D4 code in 30,000, 35,000, 40,000 and 45,000 MWd/tHM discharge burnup stages.

2.2. WIMS simulation

To evaluate the atom densities and masses of all fuel at different time steps, we used WIMS-D4 code (Winfrith Improved Multigroup Scheme Code System). This code is a general lattice cell programme which uses transport theory via deterministic method. Input data to WIMS code include the structure core, and its shapes, enrichments, and as well as dimensions. The cylindrical geometry was defined in WIMS input cells while the cell shape in VVER-1000 is a hexagonal. So, we used the following formula to approximate hexagonal geometry with cylindrical one:

$$b = \frac{a}{2}\sqrt{\frac{2\sqrt{3}}{\pi}} \quad (3)$$

where b and a are the cylinder radius and hexagonal VVER cell pitch, respectively (Amin et al., 2017).

The 69 energy group WIMS-D4 library is used to obtain cross section data. So, fuel burnup analysis for VVER-1000 was carried out at the 30,000, 35,000, 40,000 and 45,000 MWd/tHM.