KnE Engineering

AtomFuture-2017
XIII International Youth Scientific and Practical Conference
"FUTURE OF ATOMIC ENERGY - AtomFuture 2017"
Volume 2017

Conference Paper

The Comparative Analysis of Neutrons Properties of the Nuclear Fuel Produced by the Westinghouse and the TVEL for the Reactors VVER-1000 by Code SERPENT

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Abstract

In this paper, we analyze the impact on isotopic composition of spent nuclear fuel VVER-1000, due to various operational conditions, such as concentration of boric acid dissolved in water, the temperature of the fuel, and others. In addition, the impact that is caused by the technological allowances applied while manufacturing fuel assemblies that were analyzed by the mass fuel and its enrichment. The calculations were performed on models of the fuel assemblies of reactor VVER-1000. The basis was taken of a typical fuel Assembly of the Russian TVEL suppliers and the new fuel assemblies of the American company Westinghouse.

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Received: 23 December 2017
Accepted: 15 January 2018
Published: 21 February 2018

Publishing services provided by Knowledge E

Selection and Peer-review under the responsibility of the AtomFuture Conference Committee.

Keywords: SERPENT, FA-A, FA-WR, VVER-1000, SNF, Operational Conditions.

1. Introduction

In the power reactors, uranium is mainly used as a nuclear fuel, from which other isotopes are produced. Uranium-238 produces plutonium-239 in the reactor through irradiation and subsequent radiation capture. Other isotopes are created by capturing neutrons, which are usually very toxic. About 70 tons of plutonium are accumulated annually in spent fuel [1].

At present, the operator of Ukrainian nuclear power plants "Energoatom" plans to give six power units in Ukraine for the loading of American Westinghouse fuel: two at the South Ukrainian NPP and four at the Zaporizhzhya NPP. In June 2016, began the loading of Westinghouse FA-WR nuclear fuel in the core of the fifth power plant unit of Zaporizhzhya NPP, the largest energy facility in Ukraine. The first batch of

Westinghouse fuel was delivered there in February. Previously Westinghouse fuel was loaded into two units of the South-Ukrainian NPP.

In the present work are compared to the typical assembly of the Russian TVEL suppliers of fuel element [2-5] and new fuel assemblies Westinghouse [6] from the point of view of the management and storage of spent fuel. Characteristics of spent fuel, important for safety, mainly determined by the isotopic composition, generated during fuel burnup. Determination of isotopic composition of spent fuel is required to solve problems associated with:

1) Accounting and controlling of the quantities of the nuclear hazardous material;

2) Determination of the initial conditions in the analysis of thermal and radiation safety;

3) Using burnup as a nuclear safety parameter while substantiating nuclear safety of spent fuel management systems.

The isotopic composition of spent fuel is determined not only by level of burnup, but also those conditions, or, more precisely, the neutron spectrum under which this burnup occurred [7, 8]. Spent nuclear fuel with the same burnup value can have different isotope composition depending on neutron spectrum in which this occurred a burnup. The more harder was neutron spectrum, the more U-238 participates in the process of burnup (mainly because of the formation of Pu-239), and the more U-235 remains in the spent fuel with the same level of burnup [9].

In this work, the concentrations of the isotopic composition of spent fuel VVER-1000, due to various operating conditions such as concentration of boric acid dissolved in the moderator water density and fuel temperature will be calculated. In addition, the impact caused acceptable performance in the production of fuel assemblies that were analyzed on the mass of fuel and its enrichment, these conditions presented in table 1.

2. Materials and Methods

In figures 1 and 2 present an Assembly of hexagonal fuel assemblies FA-A and FA-WR designed for VVER-1000 reactor manufactured by the Russian company TVEL and Us Westinghouse, respectively. Parameters of the fuel assemblies FA-A and FA-WR are shown in table 2. In figures 3 and 4 shows images of the FA-A and FA-WR, the received code Serpent. These parameters are based on [2-5]. Each fuel Assembly contains 312

Table 1: Operational parameters that were used when performing isotope composition calculation.

Parameters	Av.	Max.	Min
Enrichment (wt%)	FA -A: 306*4.4%+ 6*3.6% (BA) FA -WR:240*4.2% + 60*3.9% + 6*3.6% + 6*3.0%(BA)	+0.5 +0.5	-0.5 -0.5
Weight fuel (kg / FA)	FA -A: 497.9 FA -WR: 552.8	+ (4.8) +(5.3)	-(4.8) -(5.3)
Concentration Boric acid (ppm)	525	1050	0
Water density (g/ cm^3)	0.72	0.74	0.7
Water temp.(Grad K)	600	600	600
Fuel temp. (Grad K)	1050	1100	900

fuel rods, 18 guide tubes and one Central tube into hexagon grids with a pitch of 12.75 mm.

Table 2: The main differences in the geometric and material parameters of FA-A and FA-WR.

Parameters	FA-A	FA-WR
Fuel stack length, mm	3530	3530
Fuel mass (UO_2), kg	497.9±4.8	552.8±5.3
Fuel pin (312 pieces/FA)		
Enrichment (wt%)	306*4.4%+ 6*3.6% (BA)	240*4.2% + 60*3.9% + 6*3.6% + 6*3.0%(BA)
Pellet ID/OD (mm)	1.4/7.57	-/7.84
Cladding ID/OD (mm)	7.73/9.1	8.0/9.14
Cladding material/ density (g/cm3)	alloy Э110 / 6.45	alloy ZIRLOTM / 6.55
Central tube		
ID/OD, mm	11.0/13.0	11.0/12.6
Material / density (g/cm3)	alloy Э635 / 6.45	alloy ZIRLOTM / 6.55
Guide tube (18 pieces)		
ID/OD (mm)	10.9/12.6	11.0/12.6
Material	alloy Э635	alloy ZIRLOTM

For calculating the isotopic composition of spent fuel assemblies of VVER-1000 were chosen code SERPENT (Version 2.1.28). SERPENT is three-dimensional continuous-energy Monte Carlo particle transport code. The burnup calculation capability in Serpent was established early on, and is entirely based on built-in calculation routines, without coupling to any external solvers. The number of depletion zones is not

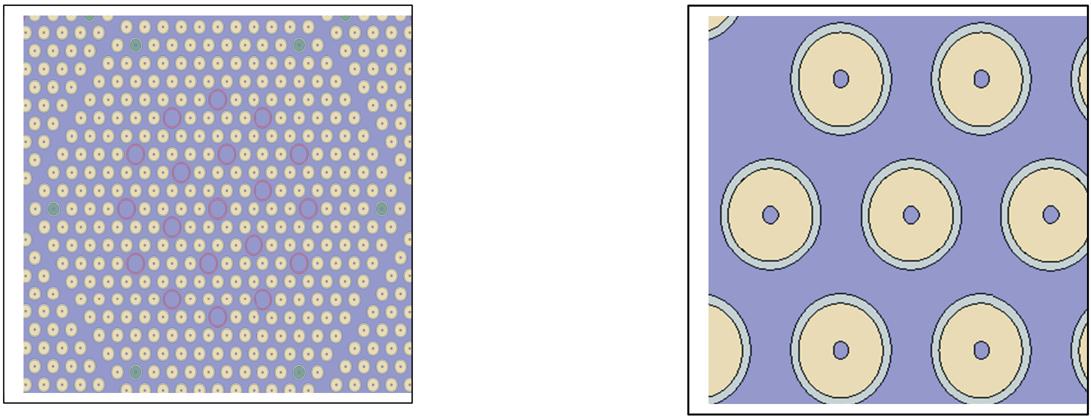


Figure 1: Model of the FA-A, at the left, FA, and at the right, the pins fuel.

Figure 2: Model of the FA-WR, at the left, FA, and at the right, the pins fuel.

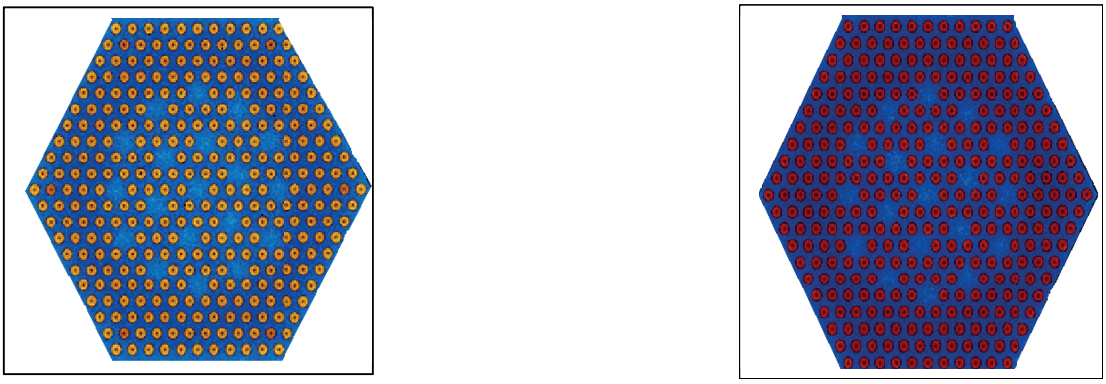
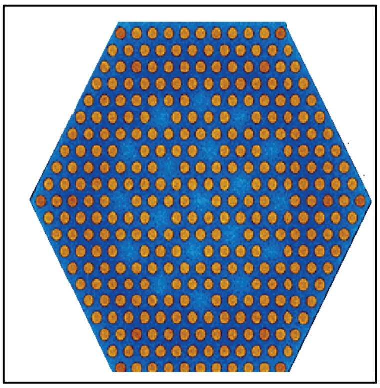


Figure 3: Mesh model of the FA-A, at the left, 0 burnup, and at the right, 50 MW.day/Kg HM.

restricted, although memory usage may require reducing the optimization when the number of burnable materials is large.

Fission and activation products and actinide daughter nuclides are selected for the calculation without additional user effort, and burnable materials can be sub-divided into depletion zones automatically. The irradiation history is defined in units of time or burnup [10].

DOI 10.18502/keg.v3i3.1603



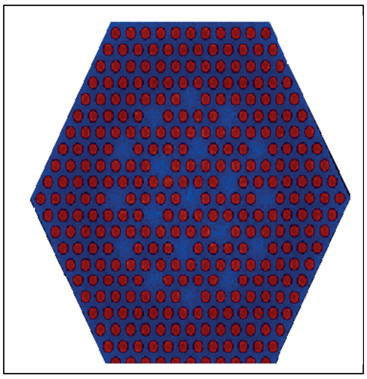


Figure 4: Mesh model of the FA-WR, at the left, 0 burnup, and at the right, 50 MW.day/Kg HM..

Each calculation Monte Carlo used 5000 neutrons per cycle, 20 inactive cycles to converge the source and 480 active cycles. Nuclear data (cross sections, data on the fission and decay of nuclei) were obtained from the library of nuclear constants ENDFB7 [11].

The calculations were carried out on supercomputers MI U (n110.basov.hpc.mephi .ru), type CPU "Intel (R) Xeon(R) CPU E5-2680 0 @ 2.70 GHz".

The calculations were performed on models of the fuel assemblies of VVER-1000 under the burnup level 50 (MW.Day/kgTM) in four-year fuel cycle. These models are based on a typical modern fuel Assembly FA-A of Russian TVEL suppliers and new fuel assemblies FA-WR of Westinghouse company.

3. The results of the calculations

To compare the two models, used for all variants of the linear power in the fuel Assembly 166 W/see through 1361,84 day, up to 50 MW.day/kgtm burnup.

The results of the calculations of the neutron multiplication factor K_{eff} depending on burnup for average operating condition (column Av., Table 1) is shown in Figure 5 and presented in table 3.

Despite that the larger volume of fuel (552.8 vs 497.9 kg), Westinghouse FA-WR has lower value of the neutron multiplication factor K_{eff} compared to FA-A company TVEL. Obviously, this is due to the lower average enrichment of fuel FA-WR. It is also noticeable that the difference between the two values in the first two and a half years (until burnup 31 MW.day/KgHM) was constant (approximately 0.02), and on the third year it became (0.015) and in the fourth year, it quickly decreased to 0.005.

For analyzing the safety characteristics of the SNF, selected parameters that play an important role in the assessment of nuclear and radiation safety in the management and storage of SNF, such as activity, the residual heat and concentration of several

Table 3: Dependence Of K_{eff} from the burnup..

Глубина	K_{eff} (FA-A) (Error=0.0004 %)	K_{eff} (FA-WR) (Error=0.0004 %)	ΔK_{eff} (TBC(A- WR))
0.0	1.33	1.32	0.013
10	1.18	1.16	0.016
20	1.09	1.07	0.016
30	1.02	1.00	0.015
40	0.95	0.94	0.010
50	0.887	0.882	0.005

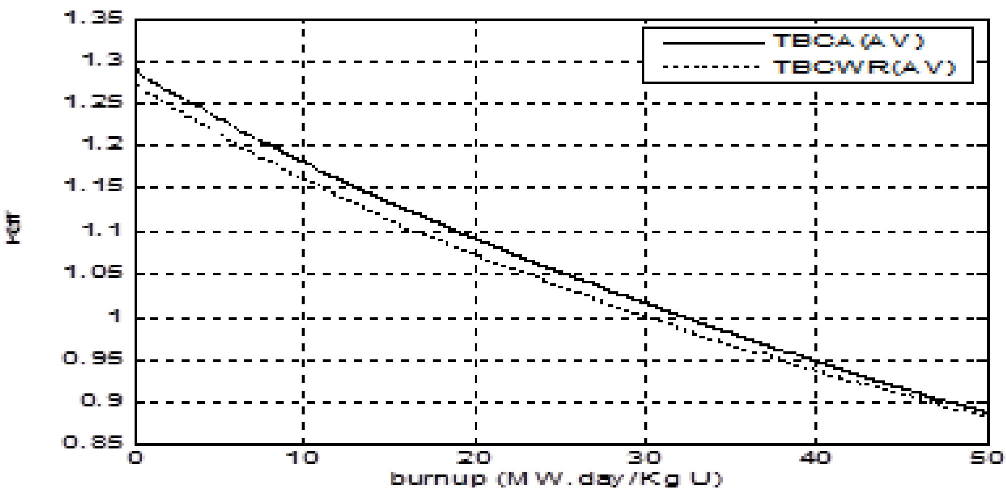


Figure 5: Dependence Of K_{eff} from the burnup.

isotopes U, Pu, Cs and Eu. The results obtained are analogous to the results published in the paper [12].

Sets of initial data were formed for each of the selected parameters according to the results of the preliminary sensitivity analysis. They include fuel assembly characteristics (fuel enrichment and fuel weight) and operational data (fuel temperature, boric acid concentration in water coolant, water density and temperature), which allow obtaining the maximum (column Max., Table 1) and the minimum (column Min., Table 1) value of the considered parameter regarding the average value (column Av., Table 1).

This was followed by the calculation for each set of the input data (Av., Max. and Min. seen in Table 1). The results of these calculations are presented in Fig. 4-12 and in table 4.

DOI 10.18502/keg.v3i3.1603

Table 4: Parameters variations for FA-A and FA-WR.

Parameters	FA-A		FA-WR		FA-WR/FA-A
	Max./Av.	Min./Av.	Max./Av.	Min./Av.	
Concentration U235	1.38	2.03	1.38	2.04	1.02
Concentration U236	1.10	1.23	1.12	1.27	1.05
Concentration Pu239	1.09	1.21	1.08	1.19	1.20
Concentration Pu	1.04	1.09	1.04	1.09	1.19
Concentration Eu154	1.02	1.06	1.02	1.06	1.21
Concentration Cs134	0.97	0.95	0.98	0.95	1.13
Concentration Cs	1.02	1.05	1.03	1.06	1.12

The concentration of the isotopes which is presented in table 4 was calculated at the moment of the discharge fuel from the core of reactor directly (at burnup 50 MW.day/KgHM). But the residual heat and the activity in table 5, were calculated after 3 years cooling in the cooling pool.

Table 5: Parameters variations for FA-A and FA-WR.

Parameters	FA-A		FA-WR		FA-WR/FA-A
	Max./Av.	Min./Av.	Max./Av.	Min./Av.	
Residual heat	0.99	0.97	0.99	0.97	1.12
Активность	1.01	1.02	0.99	1.00	1.09

In figures 7-10 presents the concentration of the main fuel isotopes such as U-235, U-236 and Pu-239, which are important for the assessment of nuclear safety in the management and storage of spent fuel.

4. Discussion

First of all, it should be noticed that, for the same burnup levels, the FA-WR (Westinghouse) has lower values of the neutron multiplication factor K_{eff} relative to FA-A. Which means that in the normal and emergency operating conditions there are additional safety margins and the possibility of additional load management systems of spent fuel.

On the other hand, it should be noticed that the value of K_{eff} in the two models comes in the fourth year the burnup, which means that the continuity of the burn up

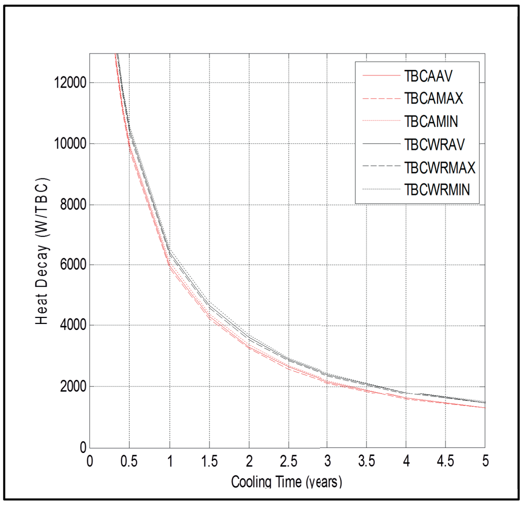


Figure 6: The Release of residual heat in spent fuel, depending on the cooling time.

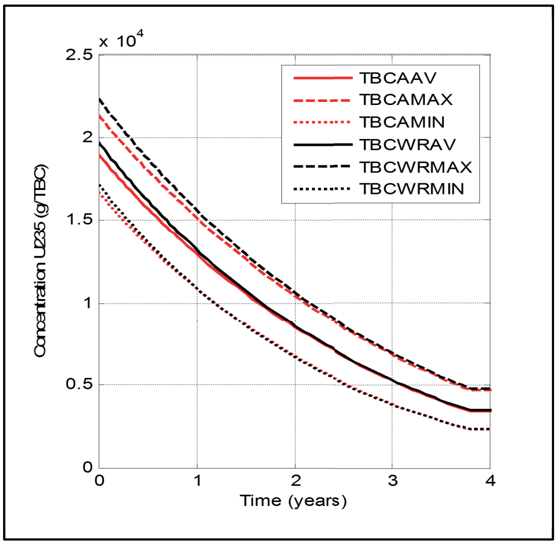


Figure 7: The mass of U235 during operation.

are equal in the two models, and the preference models of the FA-A, although the fuel weight is lower.

By comparing column 2 and 3 with 4 and 5 in table 4, we can notice that, for all isotopes, the changing of concentration due to the variation of the condition operator in the two models is the same, which means that FA-WR has the same characteristic of FA-A in these conditions. In column 5, seen the compare between the standard condition of FA-WR and FA-A, obviously, the changing of the concentration actinides (U236, Pu239 and Pu) between them is equal or less than the changing due to the changing of conditions operations. But for the fission product concentration, we see that in the FA-WR more than FA-A, and that because of the difference of the fuel mass.

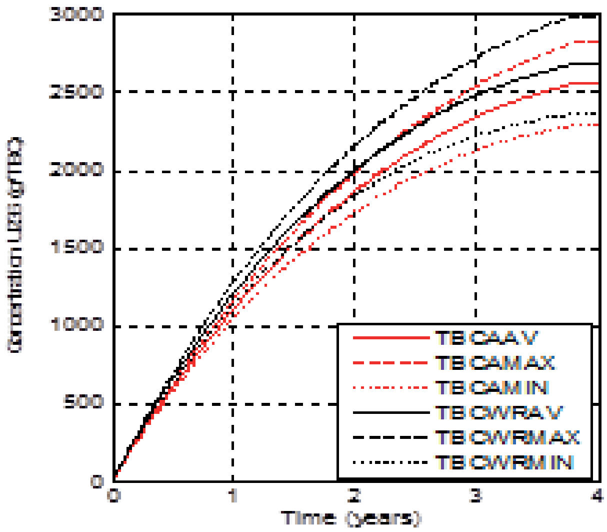


Figure 8: The mass of U235 during operation.

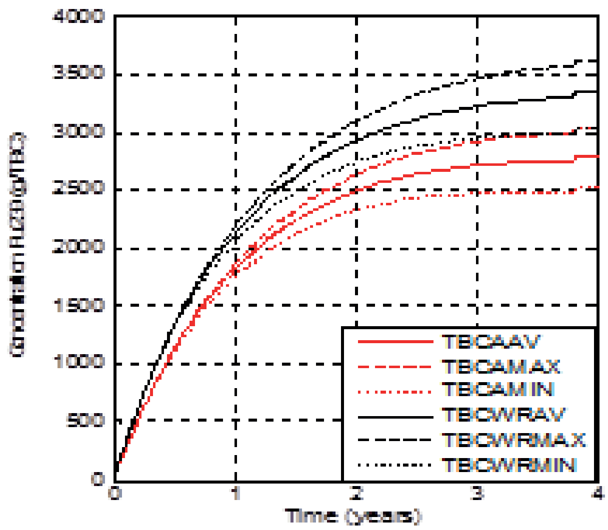


Figure 9: The mass of Pu239 during operation.

From table 5, it is necessary to notice a slight higher residual heat and activity in FA-WR (Westinghouse) when it is compared with FA-A (TVEL) after 3 years of cooling, and this is due to the difference of the concentration of the fission product. This may need a little more prolonged of cooling for the spent fuel in the cooling pool of reactor.

5. Conclusion

In General, the obtained results allow to conclude that from the point of view of the safe management and storage of spent fuel, implementing a new alternative fuel

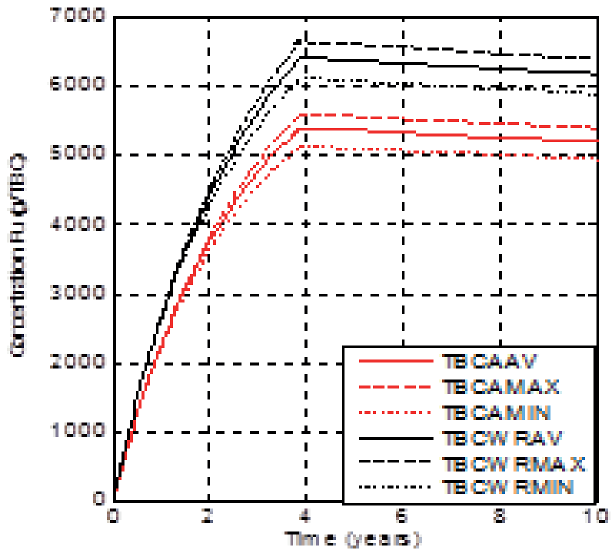


Figure 10: The mass of Pu during operation.

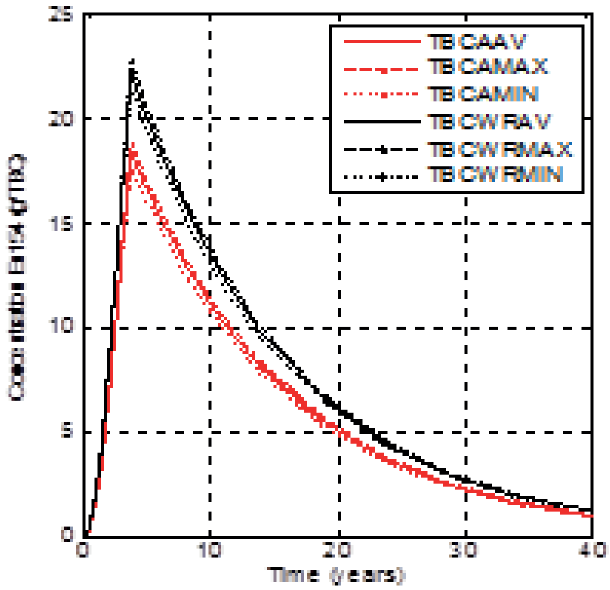


Figure 11: The mass of Eu154 during operation and cooling.

company Westinghouse on the VVER-1000 does not require modification of the current conditions and procedures.

For most of the considered spent fuel performance differences between FA-A (TVEL) and FA-WR (Westinghouse) are less than the total change in these characteristics depending on the tolerances and operating conditions for some characteristic.

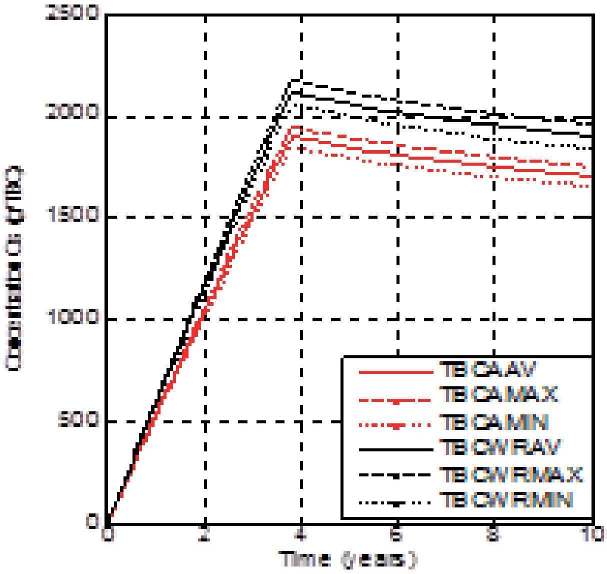


Figure 12: The mass of Eu154 during operation and cooling.

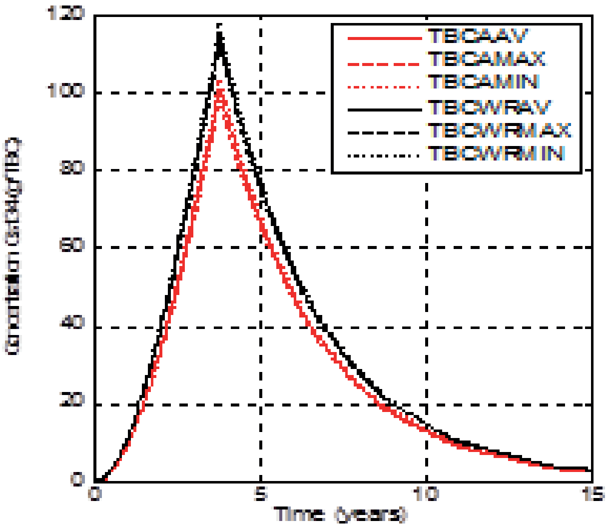


Figure 13: The mass of Eu154 during operation and cooling.

Acknowledgments

The authors wish to acknowledge the valuable opportunity given by the National Research Nuclear University MEPHI to participate easier in the conference 'atom future', and providing his students with the necessary tools for the researches like the programs and super computer etc.

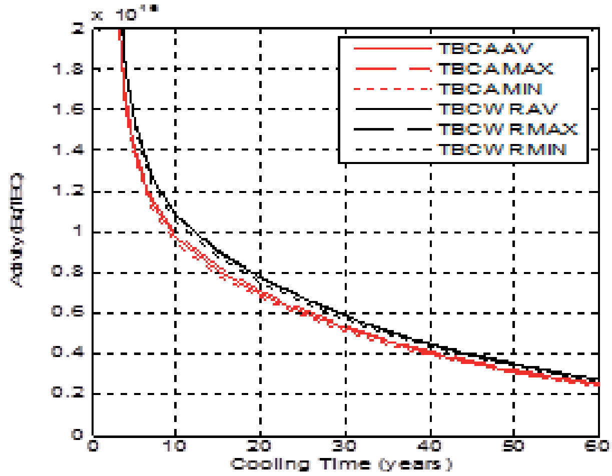


Figure 14: Release of activity in spent fuel depending on the exposure time.

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