The Monte Carlo Method: Versatility Unbounded In A Dynamic Computing World
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A PACKAGE FOR COMPUTING ISOTOPIC BURNUP AND ACTINIDE AND FISSION PRODUCT RECOVERY FOR CODE TDMCC

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ABSTRACT

This paper describes the TDMCC package for computations of initial isotopic burnup and some results of test computations. Monte Carlo methods are used as a basis of the package. The burnup program is a constituent of the code TDMCC designed for computations of VVER and PWR type nuclear power plant reactor core dynamics. The burnup package can be used both for computation of the initial state of reactor fuel in TDMCC dynamic computations and for independent computations. The burnup package employs the TDMCC interface. The work was carried out under ISTC grant (ISTC Project #1086) by Russian Federal Nuclear Center – VNIIEF.

Key words: Nuclear fuels; Burnup; Actinides; Fission products; Monte Carlo method; Testing; Fuel assembly; TDMCC.

INTRODUCTION

A detailed description of a change in isotopic composition of fuels in different reactor areas requires the partitioning of the core into the relevant number of computational zones. In the general case, in each of them, the change in the isotopic composition of the reactor is described by a set of coupled integro-differential equations, which should be solved simultaneously. This is not a simple problem. In accordance with the general approach, the time variable is separated from space-energy coordinates, and the original problem is divided into two subproblems:

- the eigenvalue problem is solved with fixed irradiation time values, with neutron flux density and essentially any other neutron functionals being determined depending on space and energy coordinates of neutrons;
- the temporal problem of the reactor fuel transmutation is solved using neutron reaction rates obtained from the eigenvalue problem solution for fixed times.

It is the simultaneous solution to these two problems that ensures the solution to the overall problem.

The direct employment of the Monte Carlo method in the neutron-physical calculations of the eigenvalue problem for multizone, heterogeneous systems (nuclear reactors) seems promising. The straightforward description of neutron path in space and in energy naturally reflects the neutron generation, moderation, absorption, etc., i.e. the neutron kinetics in wide energy and space regions. It is therefore possible to directly evaluate any physical functionals and assess their dependence on macroscopic parameters of systems under study. In particular, rates of reactions proceeding in fixed spatial (computational) fuel zones and determined by macroscopic parameters of the entire system can be estimated. This allows us to solve the temporal problem of the fuel transmutation description for each computational zone, that is, allows the parallelization (over the computational zones) of integration of isotopic kinetics temporal differential equation system for the reactor.

The content of this paper is the following:

- formulation of the computational model reliably describing the neutron kinetics and transmutation in reactor systems;
- description of the isotopic kinetics code for computations given a large number of the computational zones, which the whole system is partitioned into.;
- testing of the computational technique through description of results of integral critical assembly experiments and numerical test experiments;

computations and presentation of results for a model fuel assembly partitioned into 590 computational zones.
The burnup package has been developed within the TDMCC code /11,12/.

1. BASIC PHYSICS AND COMPUTATIONAL MODEL OF FUEL TRANSMUTATION

The description of isotopic kinetics of nuclear fuels and structural elements of the facility requires the following physical and mathematical items:

- computer program;
- neutron-nuclear interaction constants;
- independent yields of fission fragments;
- decay characteristics of fission products and actinides.

The nuclear reactor transmutation is described by the well-known system of linking differential equations. Any functionals can be determined from the computations, including:

$N_i(t)$, concentrations of separate isotopes at any time;

$\widetilde{A}_i(t)$, activity of i isotopes;

$\widetilde{P}_i(t)$, heat emission of i isotopes;

$\widetilde{R}_{f,i}(t)$, fission rate of the i-th isotope;

$\widetilde{R}_{\gamma,i}(t)$, rate of neutron radiative capture by i isotopes;

$\widetilde{R}_{r,i}(\tau)=\int_0^{\tau}\widetilde{R}_{r,i}(t)dt$, numbers of reactions r on i isotopes for time τ of the plant operation.

The technique [1] for the numerical solution to the equation system should necessarily provide for the solution to the following problems:

- eigenvalue problems with estimation of reaction rates for a number of static reference core states;
- computed data processing to supply input data to the isotopic kinetics code;
- computation of isotopic kinetics in fuel during burnup.

The flow chart of the technique to compute the neutron physical parameters and fuel isotopic composition of the thermal reactor core appears in Fig.1. The initial core parameters (concentration of i cell isotopes over k zones N_k^{0i}, geometric parameters, fuel temperature T_f and moderator temperature T_m) and the library of elementary neutron constants in the format ENDF are used for the initial data. These initial data are input into the TDMCC code. The reaction rates $R_{j,i,k}$ (where j is the reaction, i is the isotope and k is the zone) calculated with the static

package are transferred through the processing package to the isotopic kinetics code as input data. The isotope concentrations calculated by the isotope kinetics code for the reference power production points are transferred to the code TDMCC. This loop closes and ensures calculation of the initial and recovered isotope content in fuel for any power production.

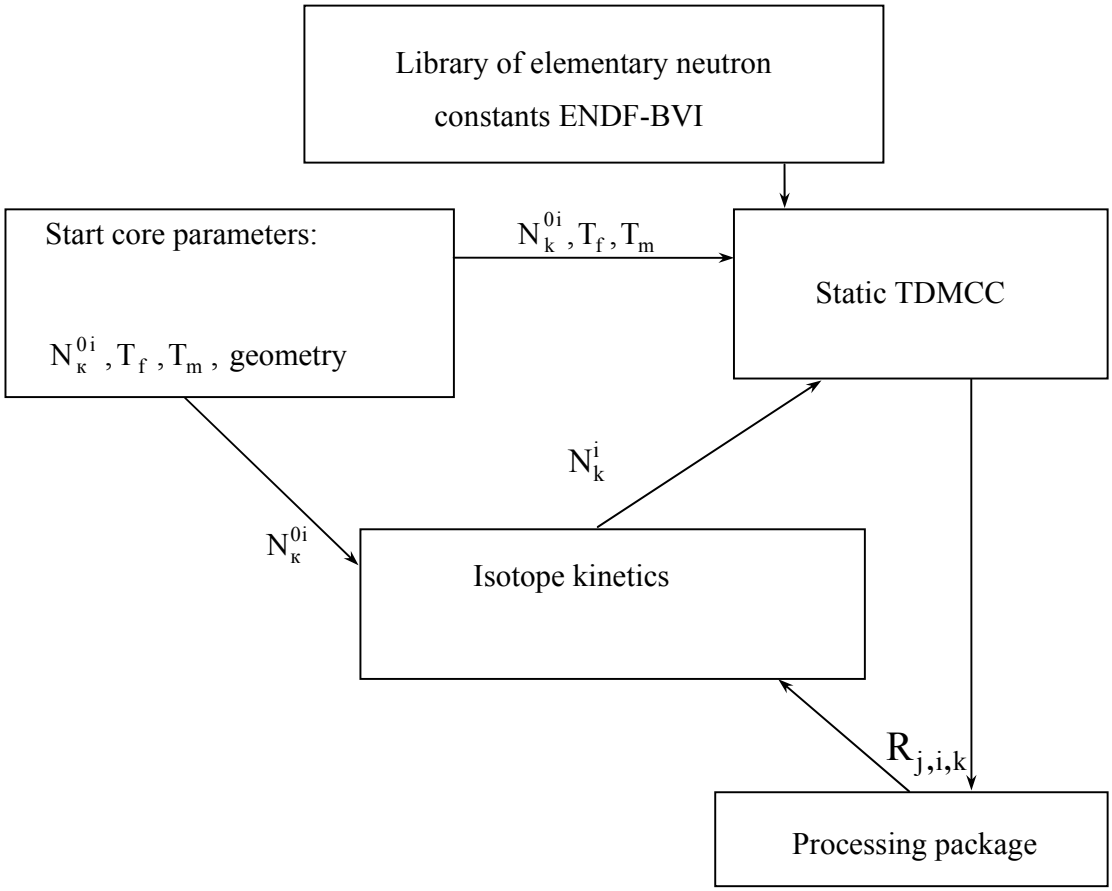

Fig. 1 – Flow chart of the technique for computing neutron physical parameters and cell isotopic composition of the thermal reactor core

The following facts should be taken into consideration when deciding on a method for solving the differential equation system of isotope kinetics:

- the actinides and fission products half-lives vary over a wide range, $0.001 \le T_{1/2} \le 10^{17}$ sec, the decay rates vary accordingly;
- in the general case, rates of neutron reactions, including atomic nuclear fission, can significantly vary in time;
- the relation among the equations is characterized with a relatively small number $(3 \div 5)$ of links.

In view of the wide range of the decay and reaction rates the isotope kinetics equation system is considered to be a rigid system. Rigid differential equation systems are readily solved with implicit methods. They allow the equation system to be integrated at a large step when rapidly varying components have tapered off to stationary values.

The Rosenbrock's method that has shown itself to be of advantage in solution of this kind of problems has been used for integration of the isotope kinetics equation system.

2. INITIAL DATA AND LIMITATIONS

The fuel isotope transmutation computations require the constant support in the following for the initial data:

- interaction of neutrons with initial and recovered isotope nuclei;
- independent fission product yields;
- nuclei decay characteristics.

In the general case this is a very large data amount and its limitation without sacrifice of the computation results is natural and, in some cases, not straightforward.

The direct inclusion of the interaction of neutrons with all fission products is hard to implement technically because of a large number of them and insufficient information about the interaction of neutrons with them. At the same time it is clear that the interaction of neutrons with short-lived fission products may not be included, since the neutrons really do not have time to interact with the fission products because of their decay and small cross section of the radiative capture of the neutrons by them.

The number of fission products T was reduced using the following criteria:

- $\sigma_{n\gamma} \geq 10$ barns;
- relatively high FP cumulative yields $(Y_{FP} \geq 1\%)$;
- radioactive nucleus recovery only in the (n,γ) process.

It should be noted that this is a conditional limitation in a certain sense, and the list of the explicitly included FP nuclei can be extended as needed.

The neutron constants for these FP isotopes are taken from libraries ENDF-V5 and ENDF-V6. The half-lives for all FP, whose independent yields are known, are taken from ref. /3/. The values of the independent yields of thermal neutron nuclear fission fragments are taken from the estimation presented in ref. /4/.

The transmutation of the initial fuel isotopes ^{235}U and ^{238}U is provided as follows. The transmutation chain of ^{235}U ends with isotope ^{238}Pu and that of ^{238}U with isotope ^{244}Cm. All the initial isotopes $(^{235}\mathrm{U}, ^{238}\mathrm{U})$ and the isotopes recovered in neutron irradiation fit into the well-known four radioactive series. All these processes have been included in the linking equation set, and the results of the computations allow us to follow the recovery, burnup, decays, specific content of actinides at any time.

The neutron constants for actinides are taken from libraries ENDF–B/V and ENDF–B/VI. It is important to note that they are fully entered in the electronic form into the Monte Carlo program complex and the Monte Carlo method naturally accounts for the self-blocking of reaction rates in the resonance cross section region and removes a number of other difficulties.

The half-lives in the β^- and α processes are taken from reference book /3/. The decay processes play an important role during fuel irradiation by neutrons and are responsible for the isotope activity taken separately and entire fuel as a whole during the fuel cooling and further handling.

3. COMPUTATIONAL TECHNIQUE TESTING

Accuracy and reliability of the results of the computations by the code TDMCC depend on those of its principal codes: calculating module TDMCC and burnup package.

The static package is much similar to and based on the C-95 code. The code TDMCC was tested in the critical assembly computations. The assembly parameter description was taken from the reference critical system compilation performed by International Cross Section Evaluation Working Group CSEWG 1974 /5/. The assemblies cover a relatively wide range of uranium and plutonium dilution with hydrogen and are characterized with various geometry and heterogeneity. The represented assembly set is full enough from the standpoint of the testing of the possibility to describe the neutron kinetics laws and features in water-cooled and water-moderated reactors.

Table 1 summarizes K_{eff} of these critical systems that have been computed by TDMCC code with using the neutron constants from the library ENDF-B/VI. The figures in brackets stand for the relative error (1σ) of the computation.

For comparison, Table 1 also presents the data computed with the widely known program MCNP elsewhere /7- 9/ using the constants from the libraries ENDF-BV and ENDF-BVI.

For most critical assemblies the computed effective neutron multiplication factors agree within a reasonable range ($< 2\sigma$) with the experimental data and the data calculated elsewhere. This suggests closeness between the results of the static TDMCC and MCNP computations, a good level of accuracy of the ENDF-B/V library's neutron constants, and, hence, reliability of the results of the neutron physical computations for reactor systems.

There is an IAEA coordinating research program providing for the feasibility study of using the thorium fuel cycle to restrict plutonium recovery and reduce toxicity of nuclear power engineering long-lived wastes. On the initiative of its authors, it was suggested that the worldwide leading laboratories should perform neutron-physical computations for a strictly definite PWR reactor cell with plutonium-thorium fuel. The computed data is summarized in report /6/. The comparison between our computed data for the same cell and those of the worldwide leading laboratories is of an undoubtful methodological interest.

The cell geometry is characterized with the following parameters: outer fuel radius R_F = 0.47cm; outer fuel cladding radius R_{clad} = 0.54cm; outer water radius R_W = 0.85cm. The fuel is a mixture of thorium dioxide and plutonium dioxide. The average fuel temperature is $T_f = 1023^oK$, the average water temperature is $T_m = 583^o$ K. The specific power in the cell is P = 211W/cm.

Tables 2 through 4 present the following cell characteristics depending on power production that have been calculated by us and by foreign laboratories:

- neutron multiplication factor in infinite medium;
- total neutron flux density;
- average nuclei fission energy;
- ratio of the plutonium content to the start content;
- ratio of the content of plutonium isotopes ^{239}Pu, ^{241}Pu to the total plutonium content;
- ratio of the recovered minor actinides to the initial plutonium;
- ratio of the recovered ^{233}U and ^{233}Pa nuclei to the initial quantity of plutonium isotopes ^{239}Pu and ^{241}Pu.

Table 1 – Critical system K_{eff}'s. computed by TDMCC code in comparison to those computed by code MCNP

No.	Assembly /5/	TDMCC ENDF-BVI	MCNP ENDF-BV	MCNP ENDF-BVI
1	ORNL1	0.9959 (0.0031)	1.0007 /7/	0.9965 /7/
2	ORNL2	0.994 (0.0024)	1.0005 /7/	0.9964 /7/
3	ORNL3	0.999 (0.003)	0.9975 /7/	0.9935 /7/
4	ORNL4	0.9963 (0.003)	0.9989 /7/	0.9950 /7/
5	ORNL10	1.0005 (0.0036)	0.9993 /7/	0.9961 /7/
6	TRX-1∗	0.9993 (0.002)	1.0003 (0.0013) /8/	
7	TRX-2∗	0.9981 (0.003)	0.997 (0.0013) /8/	
8	TRX-1	0.991 (0.0025)		
9	TRX-2	0.997 (0.003)		
10	PNL-1	1.0065 (0.003)	1.0157 (0.0015) /8/	1.0089 /7/
11	PNL-2	1.00967 (0.0024)	1.0115 (0.0017) /8/	1.0037 /9/
12	PNL-3	0.9927 (0.0021)	0.9978 /9/	0.9904 /9/
13	PNL-4	0.997 (0.002)	1.0049 /9/	0.9971 /9/
14	PNL-5	1.0004 (0.00175)		
15	PNL-6A	1.0066 (0.00139)		
16	PNL-6B	1.0075 (0.00138)		1.0025 /7/
17	PNL-7A	1.0032 (0.0032)		1.0052 /7/
18	PNL-7B	1.0032 (0.0031)		
19	PNL-8A	1.0084 (0.0021)		1.0066 /7/
20	PNL-8B	1.0071 (0.0021)		
21	PNL-9	0.9998 (0.0022)		
22	PNL-10	0.9966 (0.0027)		
23	PNL-11	1.010 (0.003)		
24	PNL-12A	1.0087 (0.0032)		1.0066 /7/
25	PNL-12B	1.00867 (0.0032)		

∗ infinite lattices

The statistical computational errors (1σ) of the values in Tables 2 through 4 are:

- 1% in the neutron multiplication factor;
- 1% in the average energy release per fission event;
- 2% in the ratio of the plutonium content to the start content;
- 4% in the ratio of the content of isotopes ^{239}Pu, ^{241}Pu to the total plutonium content;
- 4% in the ratio of recovered minor actinides to initial plutonium;
- 2% in the ratio of recovered nuclei ^{233}U and ^{233}Pa to the initial quantity of plutonium isotopes ^{239}Pu and ^{241}Pu;
- 2% in the one-group cross sections.

Table 2- K_∞ versus power production B [MW·days/kg]

B	K_∞	
	This paper	**Results of other laboratories**
0.0	1.122	1.110-1.133
30.0	0.942	0.889-0.975
40.0	0.894	0.796-0.941
60.0	0.861	0.724-0.91

Table 3 - Average energy release per fission event Q [MeV/fission] versus power production

B	Q	
	This paper	**Results of other laboratories**
0.0	210.4	205 - 211
30.0	206.95	200 - 207.7
40.0	204.94	199 - 205.6
60.0	202.63	198 - 202.8

Table 4 - Dependency of power production on the ratios of:

- **the current content of all plutonium isotopes N(Pu, B) to their start content N(Pu);**
- **the current content of fissile plutonium isotopes N (^{239}Pu, ^{241}Pu, B) to the current content of all plutonium isotopes N(Pu, B);**
- **the current content of Am and Cm isotopes N(Am, Cm, B) to the start plutonium content N(Pu);**
- **the current content of isotopes ^{233}U and ^{233}Pa recovered on thorium N(^{233}U, ^{233}Pa, B) to the start content of fissile plutonium isotopes N(^{239}Pu, ^{241}Pu).**

B	N(Pu, B)/N(Pu)		N(^{239}Pu, ^{241}Pu, B)/ N(Pu, B)		N(Am, Cm, B)/N(Pu)		N(^{233}U, ^{233}Pa, B)/ N(^{239}Pu, ^{241}Pu)	
	This paper	**Results of other laboratories**	**This paper**	**Results of other laboratories**	**This paper**	**Results of other laboratories**	**This paper**	**Results of other laboratories**
0.0	1.0	1.0	0.7	0.7	0.	0.	0.	0.
30.0	0.422	0.40÷0.43	0.399	0.39÷0.42	0.040	0.0315÷0.046	0.394	0.32÷0.4
40.0	0.296	0.28÷0.31	0.315	0.29÷0.34	0.053	0.0428÷0.0612	0.461	0.37÷0.48
60.0	0.145	0.12÷0.16	0.172	0.12÷0.23	0.074	0.06÷0.087	0.511	0.42÷0.54

The comparison of our results to those obtained by foreign laboratories indicates that:
- the obtained values are within the foreign laboratories' range;
- all numerical data agree with pressurized water cooled and water moderated reactors.

Thus, it can be concluded that the developed method for computing neutron-physical parameters of a cell and transmutation in it ensures reliable and quite accurate results. Note, however, that the semi-empirical relation /10/ used by us to calculate the energy release in isotope fission may somewhat overestimate this quantity. This naturally leads to the relevant underestimation of the neutron flux density and inadequate estimation of other calculated characteristics.

4. TRANSMUTATION IN FUEL OF THE MODEL FUEL ASSEMBLY OPERATED IN THE NOMINAL MODE

The model fuel assembly is designed as a set of zirconium pipes 10.92 mm in outer diameter and 0.66 mm in wall thickness and one continuous metal rod of the same diameter. Inside each zirconium pipe there is UO_2 nuclear fuel of 3.6% enrichment in ^{235}U and 10.2 g/cm^3 density. The assembly height is H =3.7 m. There are 204 rods. The distance between the fuel rod centers is 14.4 mm. In the fuel assembly there are also 4 longitudinal channels to accommodate control rods and one central guide channel for installation of the fuel assembly in the reactor with a 10.92-mm-diameter steel rod positioned in it. The 14-mm-diameter control rod is located inside a 1-mm-thick steel cylindrical shell. The rod is inserted into the fuel assembly as deep as one fourth of the assembly height.

The fuel assembly is cooled with water flowing along the rods. The longitudinal velocity of coolant at the inlet is 4.55 m/s. The absolute value of flow pressure at the inlet is 15.5 MPa and that of temperature 285°C. The irradiation proceeded at heat emission power of $P = 70\ kW/rod$. By virtue of the fuel assembly symmetry ¼ of the construction is considered with the boundary conditions introduced on the planes of symmetry. The overall dimensions of the computational domain are 108x108x3700 mm.

The TDMCC code described a fuel element, which was then placed 59 times into the core filled with borated water. A control rod inserted into the fuel assembly by one fourth of the assembly height was positioned at the center. A 10.92-mm-diameter steel guide rod was in the left bottom corner of the fuel assembly. From above and from below the fuel element zone was covered with steel lids with water interlayer. Specular reflection was given by the control geometry at the boundaries of the fuel element region. In the neutron kinetics simulation, the thermal motion of medium nuclei was included with account for chemical bonds, a so-called model *S(α,β)*. The simulation with account for chemical bonds used the interaction cross sections and energy-angle distribution corresponding to a given temperature. The computations employed the constants formed using library ENDF/B-V. The neutron-water interaction was described using the model of hydrogen with chemical bonds [model *S(α,β)*] in water at 600 K temperature.

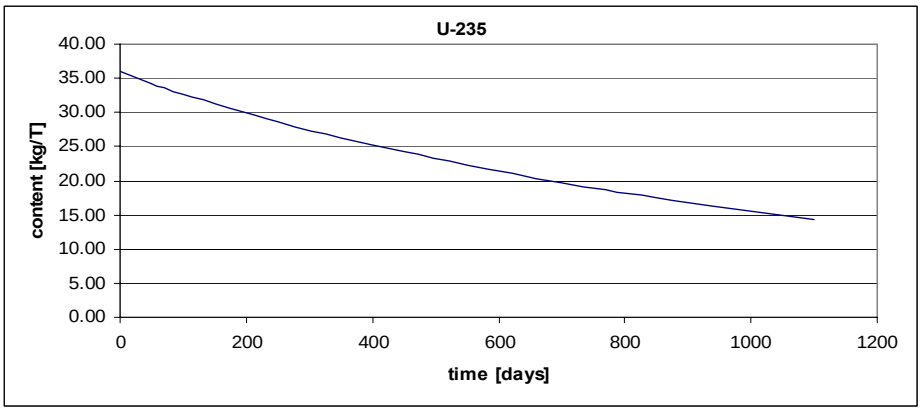

Fig. 2 – Specific content of isotope ^{235}U in the fuel assembly versus irradiation time.

Specific contents [kg/t] of actinides in fuels as functions of irradiation time $0 \le T_k \le 1100$ days have been calculated for each of the 59 fuel elements, with the fuel elements being partitioned throughout the height $(0 \div 370 cm)$ into 10 equal intervals. This is a very large amount of numerical information, and here we will present but a part of the data, which, on the one hand, characterize the irradiated fuel and, on the other hand, illustrate local features of the fuel transmutation.

Fig. 2 shows depletion of initial isotope ^{235}U from the beginning to the end of the computational irradiation, $T_k = 1100$ days. In so doing the total content of isotope ^{235}U decreased from 36 kg to ~ 15 kg per heavy metal ton. As ^{235}U burns up the contribution of ^{235}U to the total energy release naturally decreases with irradiation time.

Fig. 3 plots ^{239}Pu recovery from fuel isotope ^{238}U. The slowing-down of growth in specific content of ^{239}Pu with irradiation time is due to its depletion by fission and neutron radiative capture reactions. For irradiation time $T_k \approx 1100$ days specific content of ^{239}Pu tends to the equilibrium content.

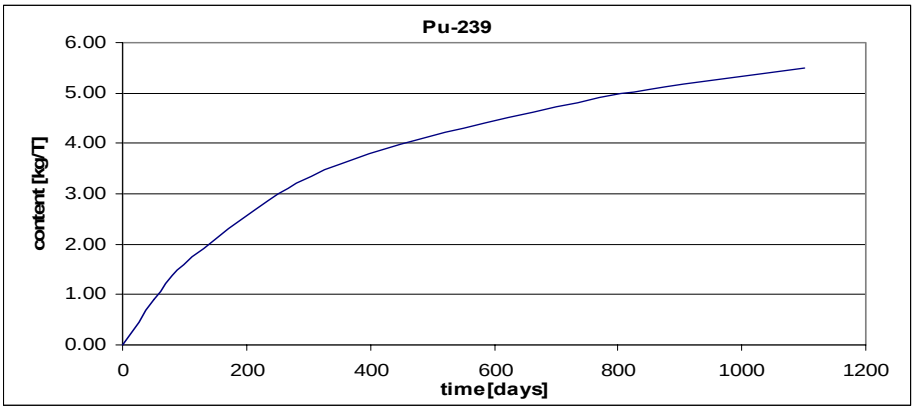

Fig. 3 – Specific content of isotope ^{239}Pu in the fuel assembly versus irradiation time.

Similar numerical data are available for a wide range of actinides from ^{235}U to ^{245}Cm. Figs. 4 and 5 present specific contents of those isotopes versus the height in the fuel assembly for 10 irradiation times, $0 \le T_k \le 1100$ days.

Fig. 4 presents the specific content of burning-up isotope ^{235}U. At the assembly ends, $H \le 50$ cm and $H \ge 250$ cm, ^{235}U is consumed less than at the central part, $50 \le H \le 250$ cm, because of neutron release. At the same time, at the upper part of the fuel assembly, for $H \ge 277.5$ cm, neutrons are absorbed intensively by the control system rod located in the fuel assembly at a depth of 92.5 cm for the entire irradiation time. The neutron absorption is so intensive, that for $H \ge 320$ cm ^{235}U does not essentially burn and its specific content remains initial $(\sim 35 kg/t)$ up to the irradiation time of $T_k = 1100$ days. From Fig. 4 one can see the ^{235}U specific content plateaus at the level of 6-8 kg/t for irradiation time of $T_k \ge 900$ days. Moreover, for $T_k \ge 1100$ days two minimums begin to show in the ^{235}U specific content that are shifted up and down from the center.

Fig. 5 presents the change in specific content of isotope ^{239}Pu produced from neutron capture in isotope ^{238}U. Qualitatively, Fig. 5 is similar to Fig. 4. The quantitative difference in the specific content values of these isotopes is due to to:

- different specific contents of initial isotopes ^{235}U and ^{238}U;
- recovery intensity in accordance with neutron radiative capture cross sections;
- depletion intensity in neutron flux.

This information can be output for all isotopes recovered.

Thus, the code ensures the computations of initial fuel isotopic burnup, recovery and burnup of isotopes up to ^{245}Cm as functions of irradiation time. In so doing there is a possibility

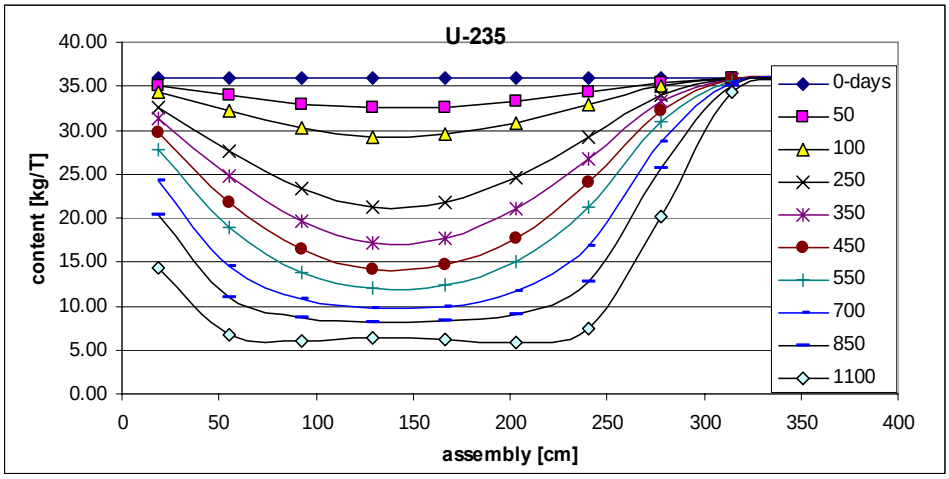

Fig. 4 – Specific content of isotope ^{235}U versus height in the fuel assembly for 10 irradiation times.

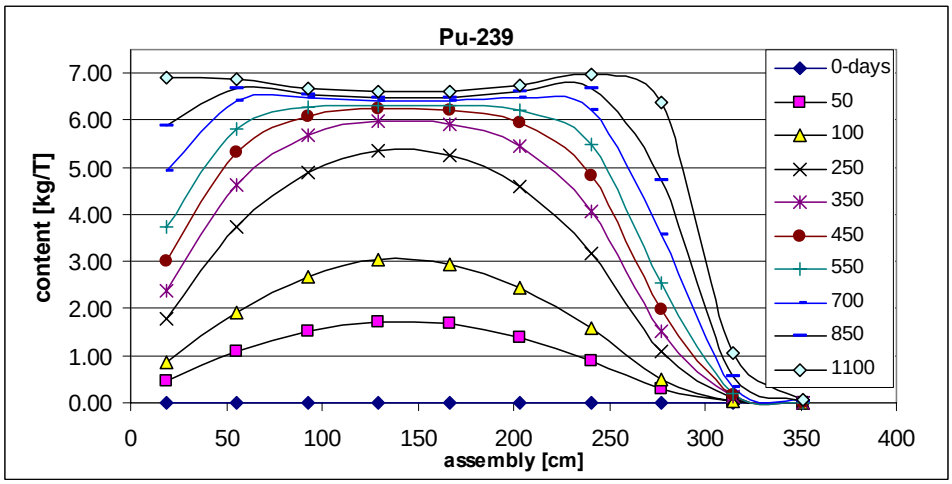

Fig. 5 – Specific content of isotope ^{239}Pu versus height in the fuel assembly for 10 irradiation times.

of accurate inclusion of all structural features of the core and study of their impact on specific contents of separate isotopes in separate parts of separate fuel elements and, hence, working on creation of conditions for optimal actinide depletion and recovery.

Figs. 6 and 7 present the fission product specific content buildup versus time and the specific content distribution over fuel assembly height at a number of irradiation times.

The time-linear fission product recovery curve in Fig.6 reflects the constancy of reactor power (fission rate). The curves in Fig.7 for distribution of recovered fission products over height have the nature of the curves for recovered actinides. A feature of the fission product recovery is that combined, they are not depleted, so there is linear growth in the fission products content until the end of irradiation in both the figures 6 and 7.

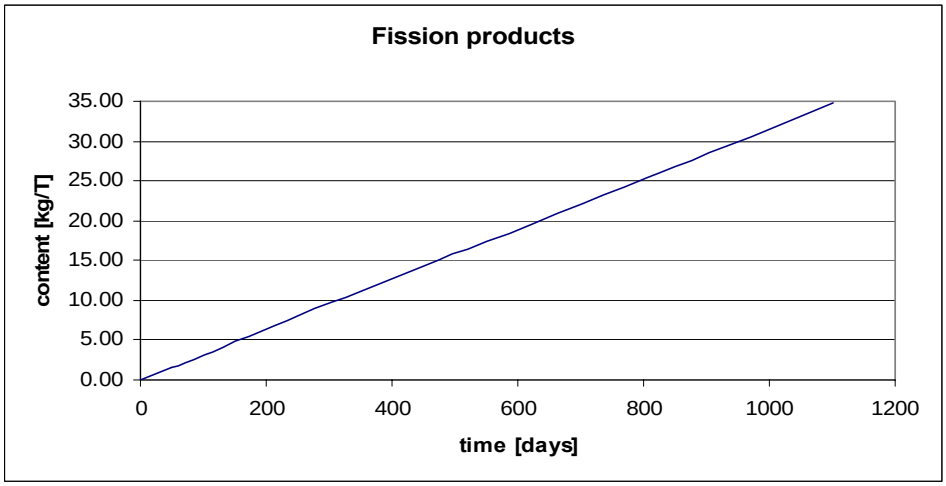

Fig. 6 – Specific content of fission products in the fuel assembly versus irradiation time.

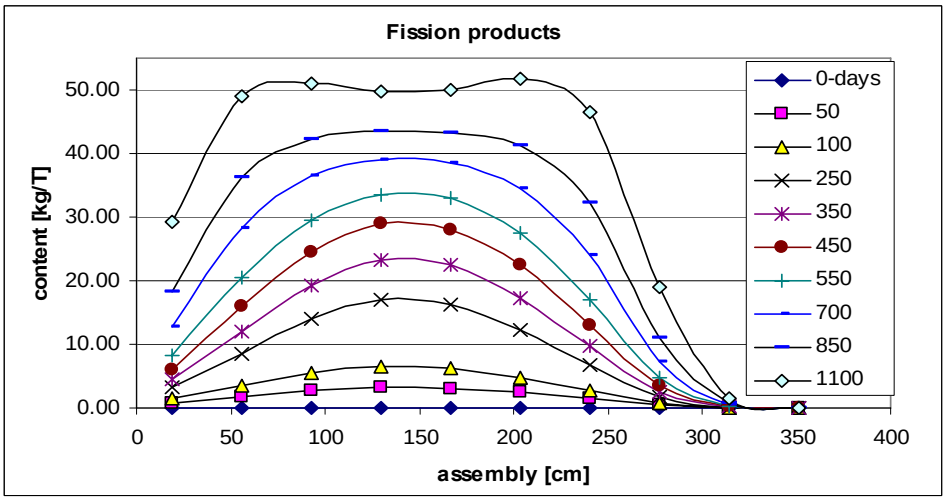

Fig. 7 – Specific content of fission products versus height in the fuel assembly for 10 irradiation times.

Thus, the code is seen to ensure the description of the fission product recovery features.

CONCLUSION

In conclusion we mention the principal results of this effort.

1. A technique for computing neutron-physical parameters and isotope kinetics in the reactor cell has been developed in the framework of code TDMCC for the description of initial fuel isotopic burnup, secondary isotope nuclei recovery, burnup and buildup.
2. The computations have been tested by description of K_{eff} of reference critical assemblies, a wide range of functionals of the numerical experiment proposed by IAEA, which allowed us to estimate the errors of the computed characteristics.
3. The computations have been performed for one fourth (block of symmetry) of the model fuel assembly containing 59 fuel rods, control and guide rod, and the computed data has been output for 590 computational zones describing local heterogeneities in the local content of burning-up and recovered actinides, as well as for a number of fission products.
4. The analysis of the computed data suggests the possibility of research into local features in the neutron field and isotopic composition in reactor cores and their manifestation both in nominal and dynamic modes of operation of the reactor.
5. The code can be efficient when run on powerful computers.

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REFERENCES

1. Marshalkin V.E., Povyshev V.M., Zhitnik A.K., Ronzhin A.B. A technique for computing neutron-physical parameters and isotopic kinetics of a thermal reactor cell and its testing. VANT. Ser. Tereticheskaya i Prikladnaya Fizika. 2001, No.3., pp.11-21.
2. E.N. Donskoy, V.A. Yeltsov, A.K. Zhitnik et al., "Monte Carlo Method at VNIIEF", *VANT, Ser.:Math.Model.Phys.Proces.*, Is.2, pp.61-64, 1993.
3. Belanova T.S., Ignatyuk A.V., Pashchenko A.B., Plyaskin V.I. Radiative capture of neutrons. Reference book. Moscow: Energoatomizdat Publishers, 1986.
4. Meek M.E., Rider B.E. Compilation of Fission Products Yields. Vallecitos Nuclear Center. NEDO-12154-2, 1977.
5. Sitaraman S., MCNP: Light Water Reactor Critical Benchmarks, General Electric (GE) Nuclear Energy, NEDO-32028, March 1992.
6. Calculation of the Isotopic Composition, Cross-Sections and Fluxes for a Typical PWR-cell Loaded with (Pu-Th)O_2 - fuel, as a Function of the Fuel Burnup. Report of IAEA, 1996.
7. Wright R.Q., White J.E., Ingersoll D.T., Fast and Thermal Reactor DATA Testing of ENDF-B-VI, Nuclear DATA for Science and Technology. Proceedings of the International Conference. Gatlinburg, Tennessee, May 9-13, 1994, Vol.2, p 815-818.

8. Sitaraman S., MCNP: Light Water Reactor Critical Benchmarks, General Electric (GE) Nuclear Energy, NEDO-32028, March 1992.
9. Robert E. MacFarlane. DATA Testing of ENDF-BVI, Nuclear DATA for Science and Technology. Proceedings of the International Conference. Gatlinburg, Tennessee, May 9-13, 1994, Vol.2, p 786-788.
10. Croff A.G., Nuclear Technology, 62, 1983, p. 335.
11. A.K. Zhitnik, N.V. Ivanov, V.E. Marshalkin, S.P. Ognev, V.M. Povyshev, I.E. Ponomarev, V.I. Roslov, T.I. Semenova, V.A. Tarasov, V.P. Fomin, T. A. Taiwo and W. S. Yang, "The TDMCC Monte Carlo Capability for Spatial Kinetics Calculations of Reactor Cores", *Trans. Am. Nucl. Soc*., **91**, 248 (2004).
12. A.K. Zhitnik, N.V. Ivanov, V.E. Marshalkin, S.P. Ognev, A.V. Pevnitsky, V.M. Povyshev, I.E. Ponomarev, V.I. Roslov, T.I. Semenova, V.A. Tarasov, V.P. Fomin, T. A. Taiwo and W. S. Yang. “Code TDMCC for Monte Carlo computations of spatial reactor core kinetics”. Chattanooga, Tennessee, Monte Carlo-2005.