Study on Nuclear Characteristics of Sodium Cooled Accelerator-Driven Transmutation System

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Series of calculations of OECD/NEA benchmark problem [1] are performed using ABBN-93 [2] cross-section data library. For an accelerator-driven system (ADS) the core neutronic characteristics and transmutation rates of minor actinides (MA) are obtained and analyzed. Discrepancies with the results of benchmark participants are found due to the differences in cross-section data and due to different methods used in calculations.

1. Introduction

Transmutation calculations are not as well established as the conventional uranium and plutonium burn-up reactor systems, because nuclear data for minor actinides are less reliable and the burn-up chains for decay and generation are not completely modeled in the computation.

We suggest to compare calculational results of neutronic characteristics for the ADS benchmark concerning MA transmutation using the ABBN-93 cross-section data library. This data library has been verified and validated by a wide set of various fast and thermal critical experiments and benchmarks.

The main idea of this work is to evaluate the status of MA transmutation calculations and also to validate codes and data for the benchmark.

The ABBN-93 cross-section data library includes group constants for nearly all-chemical elements, all long-lived actinides, all important fission product nuclides and for many isotopes. The range of considered neutron energies is from 0 to 20 MeV. Number of neutron groups is 28 or 299 as a standard and variants can be assumed between 28 and 299. The ABBN-93 is prepared from the evaluated data files library FOND-2 (File Of evaluated Nuclear Data). This library contains data for nuclides selected from BROND-2, JENDL-3, ENDF/B-V, -VI, and JEF-2 libraries. The selection is based on the detailed study of the relative performances of the above libraries in predicting the results of several micro and macro experimental results. The ABBN-93 cross-section data library can be applied in different scientific and technical fields for calculations.

2. Evaluation Method

The system subject to analysis consists of a proton accelerator and sodium cooled subcritical region driven by a proton beam of 1 GeV energy and current of 10 mA in a subcritical state of $k_{eff} \approx 0.9$. Specifications of this ADS are given in Table 1. Solid tungsten is used as a target material. Fuel is a mixture of plutonium (10%) and minor actinides (90%). Two types of fuel compositions are considered, which are denoted as MOX11 and MOX12. These compositions are given in Table 2.

Table 1. Specifications of target / core transmutation system

Specification	Value
Proton beam	1.0 GeV, 10 mA
Beam radius	15 cm
Beam profile	Uniform
Target	Tungsten (disk layers type)
Height / Radius	80 cm / 15 cm
Upper region height	26 cm
Lower region height	54 cm
Core radius	40 cm
Fuel	(10Pu-90MA)N
Pin diameter	7.3 mm
Pin pitch	9.9 mm
Pin height	80 cm
Fuel pellet diameter	6 cm
Na bond thickness	0.35 mm
Cladding thickness	0.3 mm
Reflector	Stainless steel
Inner / outer radius	40 cm / 90 cm
Top thickness	30 cm
Bottom thickness	40 cm
Coolant	Sodium

Table 2. Fuel composition (weight percent)

Fuel	MOX11	MOX12
Pu-238	1.5	2.6
Pu-239	59.3	44.5
Pu-240	23.7	31.0
Pu-241	8.7	10.7
Pu-242	5.5	9.5
Am-241	1.3	1.7
Total:	100	100
Np-237	44.6	4.5
Am-241	43.6	62.5
Am-243	9.7	24.3
Cm-244	2.1	8.7
Total:	100	100

Two-dimensional model of ADS system is used in calculations. It consists of two-region tungsten target, MA fueled core, and stainless steel reflector, each of which is cooled by sodium. The atom densities are homogenized for each region. Fig. 1 shows this caculational model.

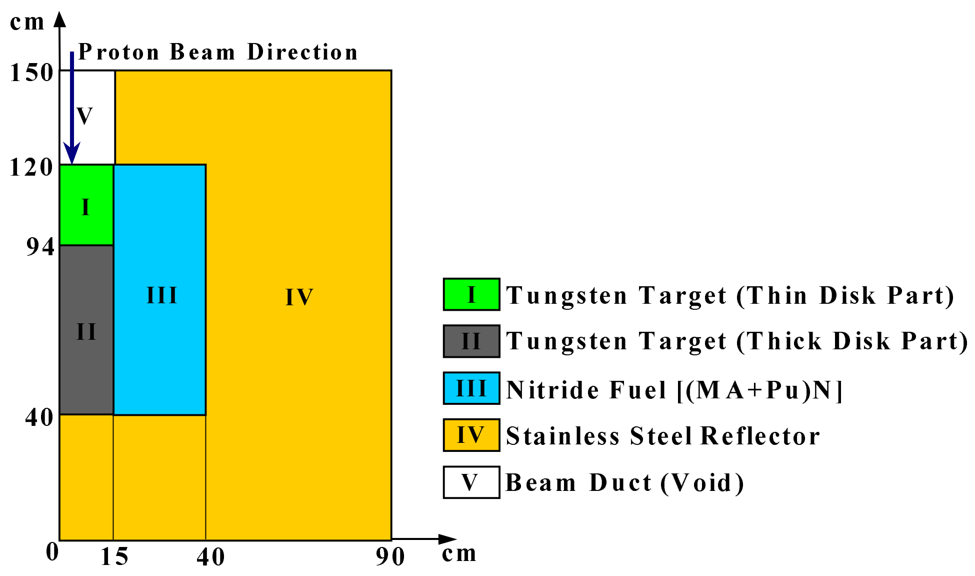

Fig.1. Two-dimensional calculational model

Calculation codes and nuclear data libraries which are used by participants of this benchmark are given in Table 3. Our calculational method is denoted as TIT. This calculational scheme is shown in more detail in Fig. 2.

Table 3. Codes and nuclear data libraries

Calculation	TIT*	JAERI**	PSI***
> 20 MeV	NMTC/JAERI	NMTC/JAERI	HETC-PSI
< 20 MeV	TWODANT	TWODANT	TWODANT
Burn-up	ORIGEN	BURNER	2DTB
Cross-section Formation	CONSYST	SCALE-4	NJOY89.62
Nuclear Data Library (Groups Number)	ABBN-93 (299 Groups)	JENDL-3.2 (73 Groups)	JEF-2.2 (33 Groups)

* TIT – Tokyo Institute of Technology
** JAERI – Japan Atomic Energy Research Institute
*** PSI – Paul Scherrer Institute

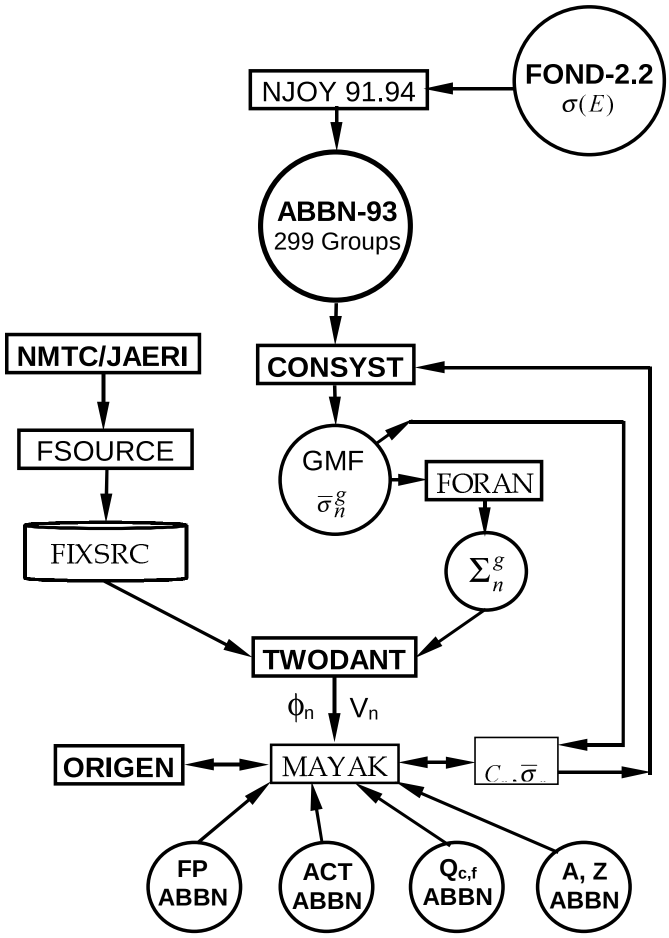

Fig. 2. Scheme of calculational method

The neutron transport process in the target/core is calculated as the fixed source problem. The nucleon-meson transport code NMTC/JAERI [3] is used for the calculations of external spallation neutron source distribution. Neutrons slowed down below 20 MeV are stored to the history file. Fixed neutron source FIXSRC file for the transport calculation is built from the history file by FSOURCE Code [3]. Only the neutron analysis below 20 MeV is considered for calculations of multiplication factor and burn-up. The transport of neutrons is calculated by TWODANT Code [4] using ABBN-93 (299 Groups) constant set. CONSYST Code [5] is used for cross-section formation. Burn-up calculations are done by the ORIGEN-S Code [6] in which cross-sections of original libraries are updated during calculations by the cross-sections of additional isotopes from the external group neutron constant set ABBN-93 libraries, which are necessary for burn-up chains. These libraries are for fission products (FP ABBN, about 170 isotopes) and actinides (ACT ABBN, about 60 isotopes). MAYAK Code [7] is used not only for intercommunication between CONSYST, TWODANT, and ORIGEN, but also for calculations of neutron reaction rates, neutron balance components, power distribution etc. MAYAK receives information about averaged fluxes and volumes of the zones from transport code. CONSYST provides MAYAK with isotopic composition and microscopic cross-sections. Then MAYAK prepares the information for burn-up code and provides the interaction of different parts of the program scheme. All complex of codes is driven by a system of batch files which give the possibility of multi-step burn-up calculations.

3. Results and Discussions

Using the calculational method and model mentioned above we have obtained main neutronic characteristics of ADS system. These results are given below in tables and figures.

Good agreement is observed for time evolution of MA, MA transmutation rates, less agreement for capture and fission reaction rates. However there are discrepancies between values of multiplication factor and core neutron spectrum. The major reasons are due to differences between fission neutron spectrum, capture and fission cross-sections used for MA nuclides in different nuclear data.

We use ABBN-93 cross-section in our standard calculations. There are only 28-group data for MA (for such as ^{237}Np, ^{241}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm). New version of ABBN cross-section data is preparing now for release. In comparison we give results which are obtained by using ABBN with new 299-group data for MA in order to take into account more correctly the resonance part of cross-sections and threshold reactions. These results are denoted as TIT-MA.

Table 4. Neutronic characteristics analysis results

	MOX11				MOX12			
	JAERI	PSI	TIT	TIT-MA	JAERI	PSI	TIT	TIT-MA
k-eff	0.8993	0.8966	0.9359	0.9184	0.9180	0.9205	0.9609	0.9406
Na Void Reactivity (% Δk/k/k')	+7.16	+7.31	+5.17	+5.23	+7.30	+7.44	+5.31	+5.38
Av. Neutron Flux (10^{15} n/cm^2/s)	2.39	1.58	3.09	3.02	2.92	1.99	3.68	3.64

Neutron spectrum in the core is shown in Fig. 3. Harder TIT spectrum is observed. Here we give results for MOX11 fuel case only, because results of MOX12 case are similar.

Core fission and capture reaction rates normalized to unit are shown for MOX11 case in Fig. 4.

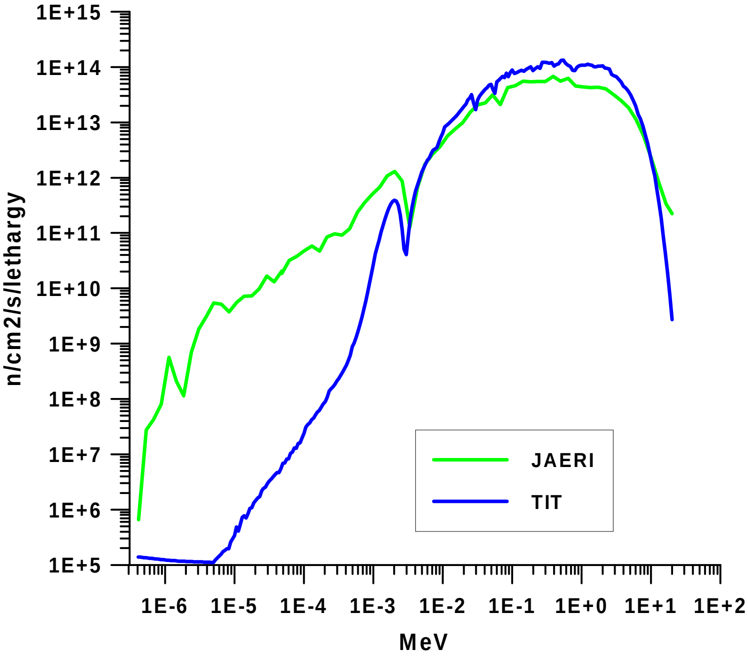

Fig. 3. Core neutron spectrum

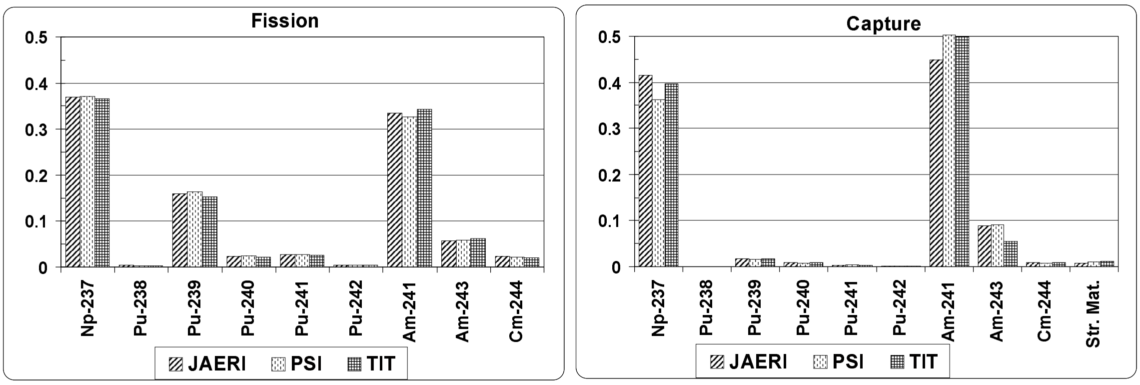

Fig. 4. Core fission and capture reaction rates for MOX11 fuel case

Some nuclides in the fuel composition of the large fractions give the contribution in reaction rates. These are ^{237}Np, ^{239}Pu, ^{241}Am and ^{243}Am. Differences between results of reaction rates for MOX11 case are reported in Table 5.

Table 5. Comparison of reaction rates results for MOX11 case, %

Nuclide	Fission		Capture	
	TIT-JAERI	TIT-PSI	TIT-JAERI	TIT-PSI
^{237}Np	-0.9	-1.1	-4.6	8.9
^{239}Pu	-4.7	-7.3	4.5	9.1
^{241}Am	2.6	5.2	10.1	-0.9
^{243}Am	6.2	5.0	-63.7	-65.1

It is necessary to note that calculations with 299-group cross-section for MA (TIT-MA results) reduce the discrepancy of the reaction rates on the average in comparison with the results of participants. Especially it is appreciable in capture rate for ^{243}Am. This discrepancy decreases from 65 to 3 percent. A similar tendency of the core reaction rates is found for MOX12 case. These results are shown in Fig. 5.

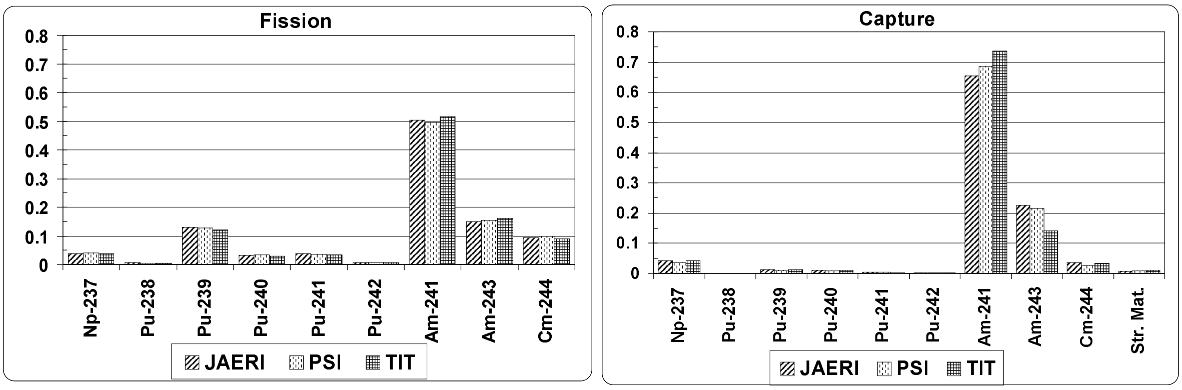

Fig. 5. Core fission and capture reaction rates for MOX12 fuel case

Time evolution of ^{237}Np number density for MOX11 and MOX12 cases is shown in Fig. 6. No big discrepancy is found between TIT and PSI results. Differences from the JAERI results are about 10 percent at the end of burn-up. It is explained by the different burn-up codes are used in calculations which consider different burn-up chains and nuclide yields.

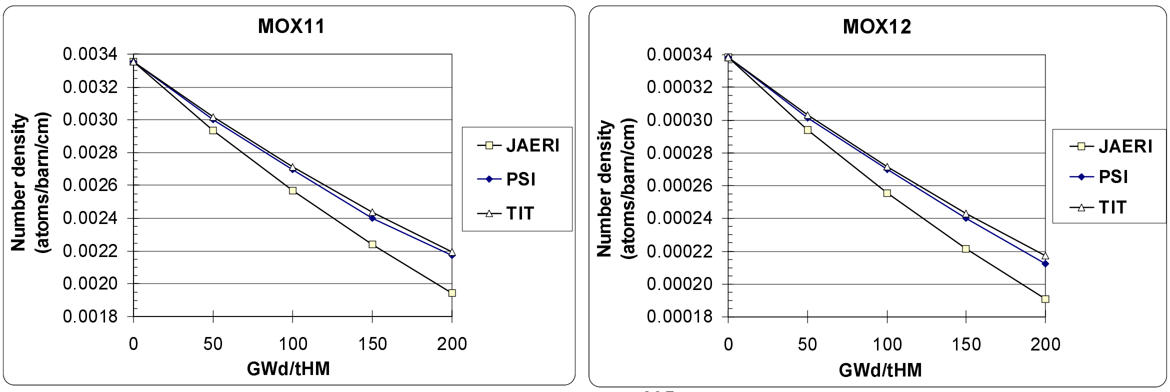

Fig. 6. Time evolution of ^{237}Np number density

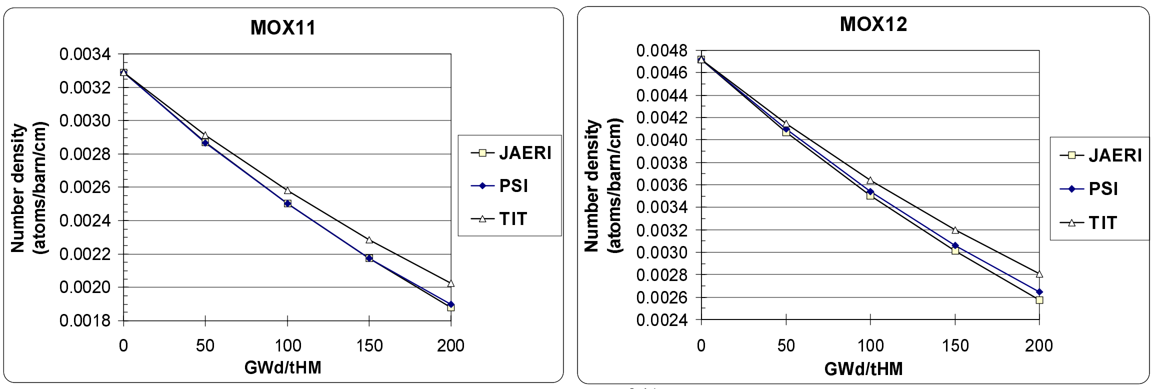

Fig. 7. Time evolution of ^{241}Am number density

Time evolution of ^{241}Am number density for MOX11 and MOX12 cases are shown in Fig. 7. The ^{241}Am number density (TIT results) for MOX11 case is 7 percent higher at the end of burn-up in comparison with results of JAERI and PSI. For MOX12 case TIT calculational results differ from JAERI results by 8 percent and from PSI results by 6 percent.

The effective multiplication factor is one of the most important integral data. Calculational results of k_{eff} as a function of burn-up are shown in Fig. 8. Remarkable discrepancies between k_{eff} are observed. Major reasons for the discrepancies are considered to be due to fission neutron spectrum, capture and fission cross-sections used for MA nuclides in different nuclear data libraries.

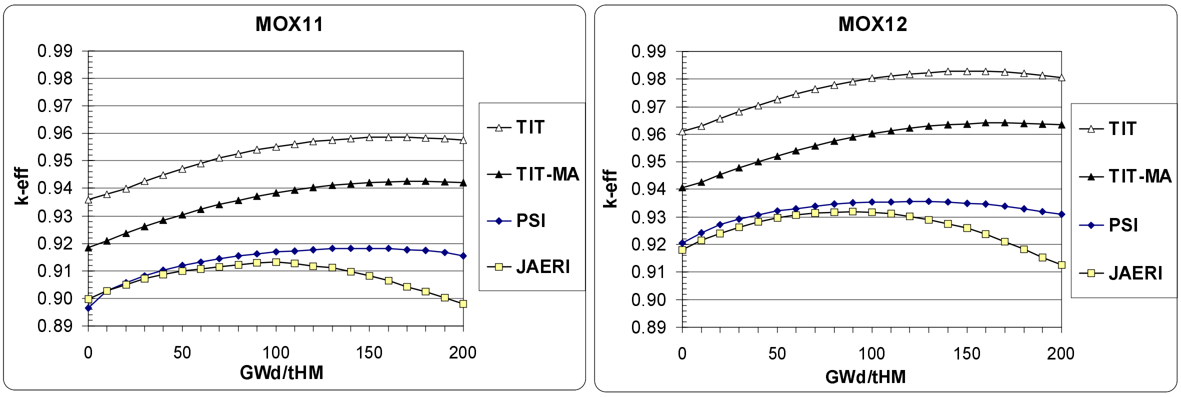

Fig. 8. Comparison of k_{eff} values as a function of burn-up

Discrepancies between k_{eff} increase at the end of burn-up. The main reason is due to the burn-up calculations by the standard ORIGEN working library for Liquid Metal Fast Breeder Reactor. It means that the fission product yields are limited by the isotopes which are included in this library for fissile nuclides. These fissile isotopes are ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu. The subcritical system under consideration has only 10 percent of plutonium and 90 percent of MAs in the fuel. Therefore the fission product accumulation is not predicted correctly (Fig.9).

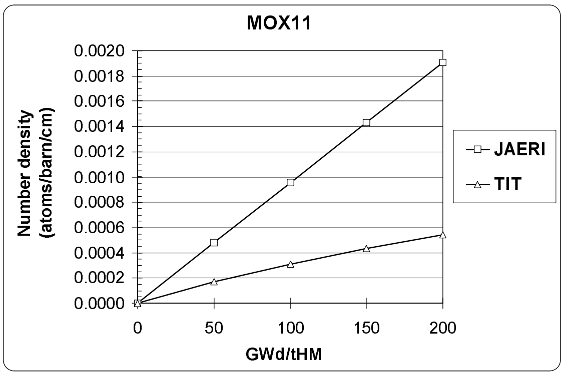

Fig. 9. Time evolution of fission products number density

Table 6. Transmutation rates of Pu and MA (only by fission), %

Nuclides	MOX11				MOX12			
	JAERI	PSI	TIT	TIT-MA	JAERI	PSI	TIT	TIT-MA
^{237}Np	36.99	37.07	36.67	36.92	3.95	3.98	3.86	3.92
^{238}Pu	0.37	0.32	0.30	0.30	0.66	0.59	0.55	0.54
^{239}Pu	15.99	16.39	15.27	14.93	13.01	12.86	12.12	11.98
^{240}Pu	2.23	2.38	2.17	2.20	3.13	3.31	2.97	3.02
^{241}Pu	2.79	2.78	2.63	2.54	3.79	3.55	3.44	3.36
^{242}Pu	0.37	0.43	0.39	0.40	0.82	0.79	0.71	0.72
^{241}Am	33.46	32.56	34.36	35.08	50.41	49.62	51.57	52.76
^{243}Am	5.76	5.83	6.14	5.56	14.99	15.54	16.11	14.61
^{244}Cm	2.23	2.21	2.06	2.09	9.39	9.76	8.90	9.09

Pu and MA transmutation rates are given in Table 6. Most discrepancies are observed for ^{238}Pu, ^{241}Pu and ^{242}Pu. Other differences do not exceed 10 percent.

4. Concluding Remarks

For an accelerator-driven system the core neutronic characteristics and transmutation rates of minor actinides are obtained and analyzed.

Good agreement is observed for time evolution of MA, minor actinides transmutation rates, less agreement for capture and fission reaction rates. However there are notable discrepancies between results for a multiplication factor k_{eff}. They are caused by differences between fission neutron spectrum, capture and fission cross-sections used for MA nuclides in different nuclear data. Increasing time evolution discrepancy between k_{eff} values is explained by underestimation of the fission products accumulation during burn-up. To consider correctly fission product yields, it is necessary to add MA nuclides to the ORIGEN working fissile isotopes library. A revision of the fission products library (FP ABBN) is required in order to add nuclides which appears due to MA fission.

Obtained results indicate that the ABBN-93 cross-section data library with appropriate modifications could be used as a basis for calculations of nuclear characteristics of subcritical MA fueled core.

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