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PHOTON SOURCE AND SHIELDING STUDIES RELATED TO THE SPENT FUEL STORAGE OF THE RA REACTOR

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ABSTRACT

The paper presents the methods developed and calculations carried out to analyse the photon source term and equivalent dose rates of the Vinča RA reactor spent fuel load inside the aluminium barrels. An analysis methodology for the radiological characteristics evolution, based on the SAS2H, MCNP-4C/ORIGEN2.1 and KENO-V.a/ORIGEN2.1 utility codes is presented along with information representing the validation of the methods and geometrical models. The sequence SAS4 with the MORSE-SGC code, and the MCNP-4C, a Monte Carlo code has been used for solving the shielding problem. Gamma dose rates estimated at 1 cm and 100 cm in radial direction from the water reflector surface of the aluminium are presented and discussed.

INTRODUCTION

The planning and activities related to the safe transport of spent fuel elements from the RA reactor[1] at Vinča Institute to the reprocessing plant, performed during last few years, have resulted in development of valid methods for radiological characterisation of spent fuel and shielding analysis of storage containers and transport casks with spent fuel elements of the RA reactor. This development has been based on the Vinča Institute report presenting the historical review of fuel depletion and fuel burnup data for the RA reactor spent fuel elements.[2]

Determination of the isotopic composition of the materials present in spent fuel of the RA reactor and subsequent derivation of the heat generation and radiation source terms has been the first study performed.[3] The complexity of this study arises from the end region effect between fuel elements placed inside the fuel channel. This paper describes three procedures, being prepared for the analysis of spent fuel characteristics (radiation sources, decay heat, and spent fuel isotopic composition) of the RA reactor. The first procedure is based on the application of the SAS2H control module[4] from the SCALE-4.4a code system[5] and an approximate geometrical model. The second computational tool employed consists the MCNP-4C utility code[6] and ORIGEN2.1 code[7] interfaced by the MOCUP driver.[8] Recently developed,[9] the third procedure KWO2 is based on the applications of the KENO-V.a code[10] and ORIGEN2.1 code.

Results representing the validation of the methods and geometrical models are included and radiation characteristic evaluations (radiation sources) of spent fuel slugs of the RA reactor are presented. Analysis of those results has allowed for further discussion in choosing the appropriate values of the radiation source terms needed for the shielding investigations of the stainless steel and aluminium containers with spent fuel slugs of the RA reactor.

The design oriented shielding analysis was performed with the standard calculation model available in the SAS4 sequence[11] of the SCALE-4.4a code system, which is based on the MORSE-SGC,[12] a Monte Carlo code with the (27 neutron – 18 gamma) coupled energy groups cross-section data library. The calculation model used assumes that the materials of fuel assemblies and their sources have been smeared over the horizontal cross section of the fuel assembly. For reference shielding calculation the MCNP-4C, a Monte Carlo code with proton cross section data library MCPLIB (designed by identifiers ending in .02p) and detailed geometry of spent fuel containers were used.

Example application is given for a case with 30x6 spent fuel slugs in the aluminium barrel surrounded with 100 cm of light was analysed. The results obtained under assumption that all fuel slugs were irradiated at 6000 MWd/t (average fuel burnup of the RA reactor spent fuel with 2% ^{235}U enriched metal uranium) and cooled for 40 years.

PROBLEM DESCRIPTION

The 6.5/10 MW heavy water moderated and cooled research reactor RA started operation in 1959. It was first operated with 2% enriched uranium metal fuel. The fuel element is aluminium clad hollow cylinder with diameter of 3.7 cm, and height of 11.3 cm, containing metal uranium in tubular form. New fuel with 80% enriched uranium dioxide dispersed in aluminium, having the same geometry shape and amount of ^{235}U per fuel element, was purchased in 1976. From 1976 to 1979, the reactor operated with mixed 2% and 80% enriched fuel core.

The complexity of fuel depletion and decay analysis for the reactor RA arises from the effect of the end region between the fuel tubes of adjacent fuel elements in the reactor RA core. This effect can be accurately account only with detail three-dimensional (3D) model. One 3D model of reactor RA unit cell, used in the MCNP-4C and KENO-V.a Monte Carlo calculations is shown in Figure 1.

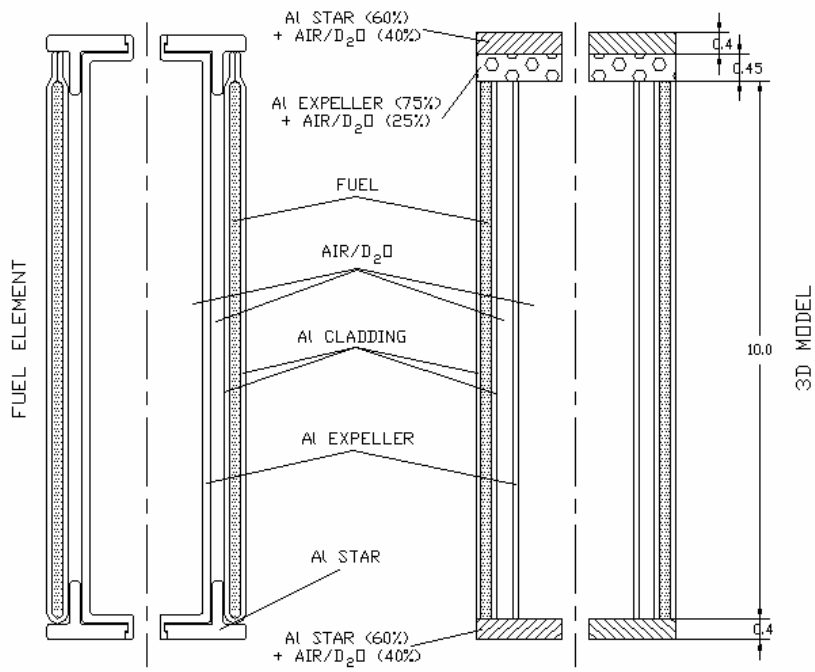

Fig.1. Vertical cross section of the fuel element (slug) and the 3D model

In this model, detailed design specifications for the reactor RA fuel element are modelled as closely as possible. In order to overcome the end region problem, a multi-region cylindrical Wigner-Seitz cell with white boundary conditions at the cell boundary (i.e., with isotropic reflection at the outer cylindrical boundary) is introduced.[3] It corresponds to an infinite array of infinitely long cylindrized cells. In this one-dimensional (1D) model, the aluminium from the end region is added to the inner and outer aluminium cladding and outer cylindrical moderator is enlarged for the volume of heavy water in the end region. The geometry of the proposed unit cell is given in Table I.

Table I. Geometrical Dimensions of the Wigner-Seitz Cell of Reactor RA

Zone	Material	Radius (cm)	
		3D-Model	1D-Model
1	Heavy Water	1.05000	1.00677
2	Clad (Al-Sav)	1.17500	1.17500
3	Heavy Water	1.45000	1.42021
4	Clad (Al-Sav)	1.55000	1.55000
5	Metal U (2% ^{235}U)	1.75000	1.75000
6	Clad (Al-Sav)	1.86000	1.89093
7	Heavy Water	2.05000	2.05000
8	Fuel Channel (Al-Sav)	2.15000	2.16266
9	Heavy Water (Outer Moderator)	Square Cell (Pitch = 13 cm)	7.79664

As an example of validation work, the case with 30x6 spent fuel slugs in the aluminium barrel (figure 2) surrounded with 100 cm of light was analysed. The results obtained under assumption that all fuel slugs were irradiated at 6000 MWd/t (average fuel burnup of the RA reactor spent fuel with 2% ^{235}U enriched metal uranium) and cooled for 40 years.

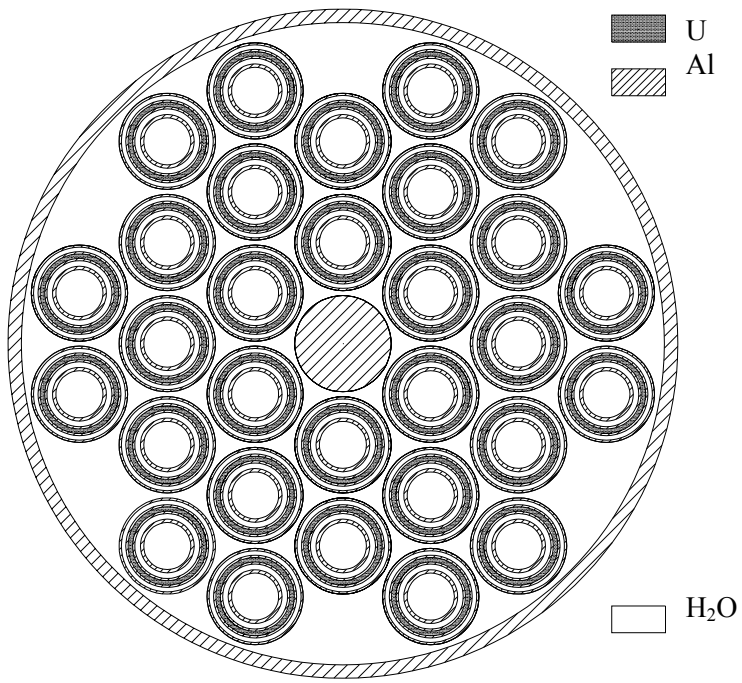

Fig. 2. Horizontal cross section of aluminium barrel with spent fuel slugs of the RA reactor

MODELLING AND CALCULATIONS

The complexity of isotopic composition study arises from the end region effect between fuel elements placed inside the fuel channel. This paper describes three procedures, being prepared for the analysis of spent fuel characteristics (radiation sources, decay heat, and spent fuel isotopic composition) of the RA reactor.

The first, design-oriented procedure is based on the application of the SAS2H control module from the SCALE-4.4a code system and an structured approximate geometrical model, which uses a simplified unit fuel within an infinite lattice for the fuel burnup analysis (*asquarepitch* option for annular cylindrical rods in a square pitch).

The second computational tool employed consists of the MCNP-4C and ORIGEN2.1 codes interfaced by the MOCUP driver. In addition to calculating, flux and power distribution, MCNP-4C calculates effective one-group cross-sections for the fuel constituents. These cross sections are used by ORIGEN2.1 code for burnup analysis. The cross-sections used for MCNP-4C calculations are taken from the Vinča VMCCS library;[13] based primarily on ENDF-B/VI data but, if not available, on JENDL3.2 and BROND2 data. For isotopes not included in the MCNP-4C analysis, ORIGEN2.1 uses cross sections from its pre-processed standard libraries. The new ORIGEN2.1 one-group cross sections library (Hwrleu.lib) was prepared by using a detailed (3D) model of reactor RA unit cell and the MCNP-4C Monte Carlo code. This library replacement is important in regard to the radiological characterisation of reactor RA spent fuel elements.

Recently developed, the third procedure KWO2 is based on the applications of the KENO-V.a and ORIGEN2.1 utility codes with the ENDF/B-V based 238-group cross section library.[5] In this procedure the standard predictor/corrector steps are included (instead of predictor steps that are used in the MOCUP procedure). The major advantage of using KENO-V.a/ORIGEN2.1 procedure is the decrease in computation time by about a factor of 20 compared with MOCUP. In the both: MOCUP (MCNP-4C/ORIGEN2.1) and KWO2 (KENO-V.a/ORIGEN2.1) reference procedures, the fuel burnup model with 90 main fission products was used.

The design oriented shielding analysis was performed with the standard calculation model available in the SAS4 sequence of the SCALE-4.4a code system, which is based on the MORSE-SGC, a Monte Carlo code with the (27 neutron – 18 gamma) coupled energy groups cross-section data library. The calculation model used assumes that the materials of fuel assemblies and their sources have been smeared over the horizontal cross section of the fuel assembly.

The reference shielding calculation is based on the MCNP-4C, a Monte Carlo code with proton cross section data library MCPLIB (designed by identifiers ending in .02p) and detailed geometry model of spent fuel containers.

RESULTS

The validity of the Wigner-Seitz cylindrization for the reactor RA unit cell was carefully verified by comparison between the Monte Carlo and transport calculations of the cylindrized cell and the initial square cell in 3D geometry. Results for various burnup conditions obtained as ^{137}Cs evolution without any buckling corrections to the neutron spectrum are shown in figure 3.

The results obtained for the photon source spectrum from one fuel element of the RA reactor (with 2% of ^{235}U in metal uranium) are given in figure 4.

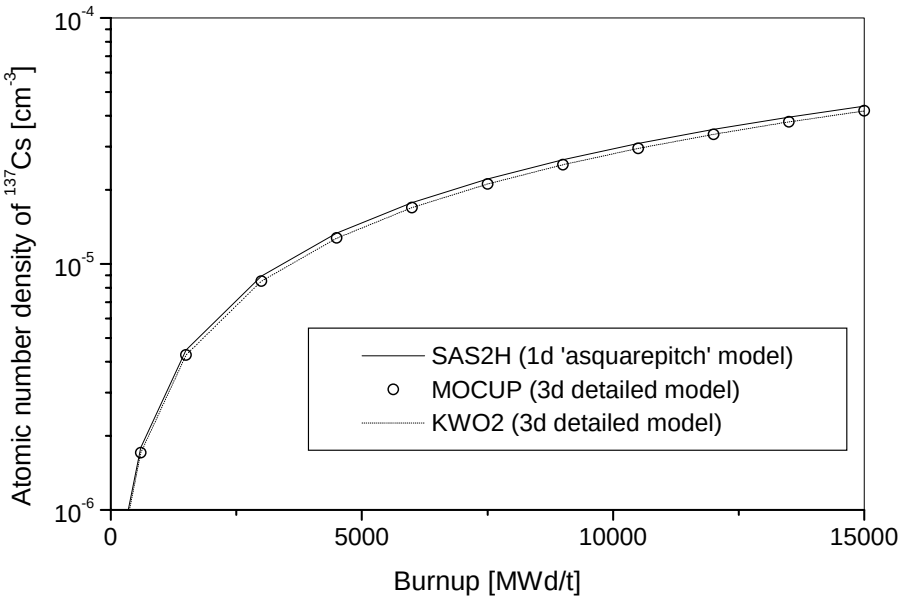

Fig. 3. Comparison of ^{137}Cs density during the fuel burnup

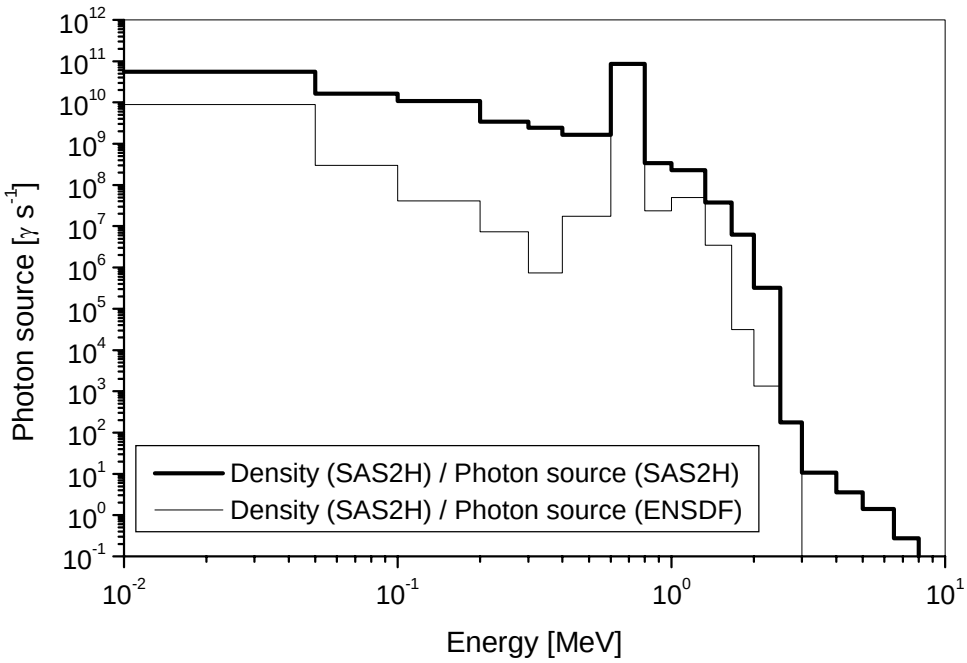

Fig. 4. Comparison of energy spectra of photon sources obtained with the ORIGEN-S code (based on the ENDF/B-IV, i.e., DLC-7E library) and ENSDF oriented calculation (based on the corrected ENDF/B-VI.8, i.e., EPDL-97 library) for one fuel slug of the RA reactor with fuel burnup of 6000 MWd/t and cooling time equal to 40 years

The results given in figure 4 are obtained by eliminating the double accounting of photon emissions for parents and metastable daughters in the data evaluations of the Evaluated Nuclear Structure Data Files (ENSDF). The photon source strengths obtained with ORIGEN-S procedure were about 50% higher than the new ENSDF based results. The high difference is observed for the low energy photons, the photons that do not play a significant role in the shielding

calculations. For ^{137m}Ba photons (at 661.7 keV), covering about 90% of photon source strength, the photon sources agreed within a few percent.

Analysis of those results has allowed for further discussion in choosing the appropriate values of the radiation source terms needed for the shielding investigations of the stainless steel and aluminium containers with spent fuel slugs of the RA reactor.

As a representative fuel assembly for the shielding calculation, the aluminium barrel with 30x6 spent fuel slugs of initial enrichment equal to 2% of ^{235}U (figure 2) surrounding with 100 cm of light water is chosen. In this case the aluminium barrel with spent fuel slugs has been considered appropriate to model the average sources corresponding to fuel burnup of 6000 MWd/t and cooling time equal to 40 years. The source energy spectra and nuclide densities of these fuel elements provided by the ORIGEN-S and ENSDF based calculations have been used as input for the further Monte Carlo transport calculations. Constant radial and axial distributions for active part of the sources have been assumed.

The results for energy spectra of photon flux at the outer surface of the radial light water reflector surrounding the aluminium barrel with 30x6 spent fuel slugs (at the average fuel burnup of the RA reactor equal to 6000 MWd/t and with cooling time equal to 40 years) are shown in figure 5, and obtained equivalent gamma dose at distances equal 1 cm and 100 cm from radial light water reflector are given in tables I, II and III.

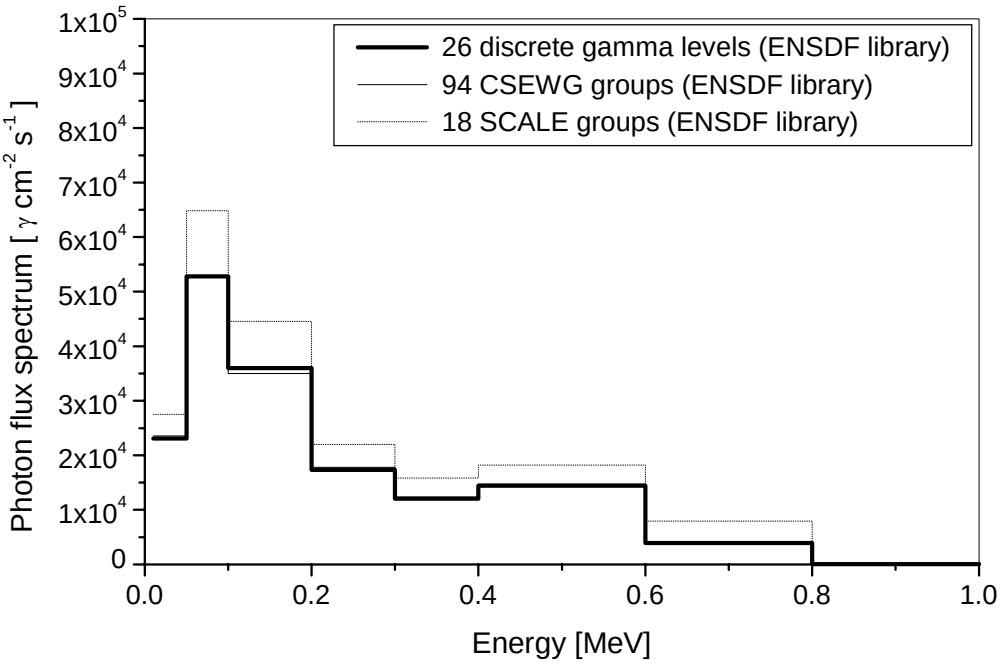

Fig. 5. Comparison of energy spectra of photon flux at the outer surface of light water reflector surrounding aluminium barrel with 30x6 spent fuel slugs obtained for different photon sources based on the ENSDF based libraries

Table I. Comparison of equivalent gamma dose rate outside the light water reflector surrounding aluminium barrel with 30x6 spent fuel slugs obtained with different MCNP™ codes

Case	Atomic Number Density	Photon source			Equivalent gamma dose rate [mSv/h]	
	Used code	Used code	Strength (γ s^{-1})	Source definition	1 cm from H_2O reflector	100 cm from H_2O reflector
MCNP-4C/MCPLIB2 (ENDF/B-IV, DLC-7E)						
1	SAS2H/ ORIGEN-S	ENSDF based routine	$1.81 \cdot 10^{13}$	26 (8+18[1]) discrete levels	0.94 (±1.4%)	0.20 (±1.7%)
MCNP-5[2]/MCPLIB4 (ENDF/B-VI.8, EPDL97)						
2	SAS2H/ ORIGEN-S	ENSDF based routine	$1.81 \cdot 10^{13}$	26 discrete levels	0.91 (±1.4%)	0.20 (±1.8%)

Table II. Comparison of equivalent gamma dose outside the light water reflector surrounding aluminium barrel with 30x6 spent fuel slugs (at the average fuel burnup of the RA reactor equal to 6000 MWd/t and with cooling time equal to 40 years) obtained for different photon sources

Case	Atomic Number Density	Photon source			MCNP-4C/MCPLIB2 Equivalent gamma dose [mSv/h]	
	Used code	Used code	Intensity (γ s^{-1})	Source definition	1 cm from H_2O reflector	100 cm from H_2O reflector
ORIGEN-S and ORIGEN-2.1 libraries						
1	SAS2H/ ORIGEN-S	ORIGEN-S	$3.20 \cdot 10^{13}$	Histogram 18 groups	1.22 (±1.2%)	0.26 (±1.5%)
2	SAS2H/ ORIGEN-S	ORIGEN-S	$4.32 \cdot 10^{13}$	18-discrete levels	0.96 (±1.5%)	0.20 (±1.9%)
3	MOCUP/ ORIGEN-2	ORIGEN-2	$4.41 \cdot 10^{13}$	18-discrete levels	0.62 (±1.8%)	0.13 (±2.3%)
ENSDF based library						
4	SAS2H/ ORIGEN-S	ENSDF based routine	$1.81 \cdot 10^{13}$	Histogram 18 groups (SCALE)	1.21 (±0.9%)	0.26 (±1.1%)
5	SAS2H/ ORIGEN-S	ENSDF based routine	$1.81 \cdot 10^{13}$	Histogram 94 groups (CSEWG)	0.95 (±1.0%)	0.20 (±1.3%)
6	SAS2H/ ORIGEN-S	ENSDF based routine	$1.81 \cdot 10^{13}$	8+18 discrete levels	0.94 (±1.0%)	0.20 (±1.3%)

[1] The first 8 gamma lines represents 99.9% of photon source, and other 18 effective gamma lines cover the rest of photon source (0.1%).

[2] This results was obtained with the MCNP-5.0 utility code licensed to M. Pešić from the Vinča Institute of Nuclear Sciences

Table III. Comparison of equivalent gamma dose outside the light water reflector surrounding aluminium barrel with 30x6 spent fuel slugs (at the average fuel burnup of the RA reactor equal to 6000 MWd/t and with cooling time equal to 40 years) obtained for different geometrical models used with the MORSE-SGC code

Case	Atomic Number Density	Photon source			MORSE-SGC (18-groups) Equivalent gamma dose [mSv/h]	
	Used code	Used code	Intensity (γ s^{-1})	Source definition	1 cm from H_2O reflector	100 cm from H_2O reflector
Detailed (heterogeneous) geometry model						
1	SAS2H/ ORIGEN-S	ORIGEN-S	$3.20 \cdot 10^{13}$	Histogram 18 groups	1.26 (±1.6%)	0.28 (±1.2%)
Simplified geometry model (with volume homogenised fuel cell)						
2	SAS2H/ ORIGEN-S	ORIGEN-S	$3.20 \cdot 10^{13}$	Histogram 18 groups	1.22 (±1.0%)	0.27 (±0.8%)

The resulting dose rate values will be used to assist the development of the working procedures that have to be observed during the loading of the cask as well as for the audit activities. The conservative approximations used have to be taken into account in order to have more precise basic input needed to establish the conditions of personnel operation in the working area, and to ensure the safety of the staff under ALARA principles.

CONCLUSIONS

This study shows:

- A good agreement between two MCNP™ calculations, the MCNP-4C code (with the MCPLIB2, the ENDF/B-IV, i.e., DLC-7E based library) and MCNP-5 code (with the MCPLIB4, the new ENDF/B-VI.8, i.e., EPDL97 based library);
- A good agreement between results for the equivalent gamma dose rate obtained with two different discrete levels models of photon source (i.e., with the ORIGEN-S code and new ENSDF based model);
- A high difference (about 30%) between these models and discrete levels model used in the ORIGEN2.1 code, since this code assigns the ^{137m}Ba line (at 661.7 keV) to pre-defined line at 575 keV);
- A good agreement between the 18 groups results obtained with two libraries (i.e., with the ORIGEN-S code and new ENSDF based model), and a high difference (about 25%) of these results and results obtained with the ORIGEN-S and ENSDF discrete levels models; and
- A good agreement between the 18 groups results obtained with two geometry models used with MORSE-SGC code (i.e., with the detailed representation of fuel elements and with volume homogenised fuel cells), and a high difference (about 30%) of these results and results obtained with the ORIGEN-S and ENSDF discrete levels models.

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