The Monte Carlo Method: Versatility Unbounded In A Dynamic Computing World
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CANDU-SEU ENRICHMENT EFFECTS ON SPENT FUEL MONTE CARLO SHIELDING ANALYSIS

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ABSTRACT

Shielding analyses are an essential component of the nuclear safety, the estimations of radiation doses in order to reduce them under specified limitation values being the main task here.

Last decade, both for operating reactors and future reactor projects, a general trend to raise the discharge fuel burnup has been world wide registered. For CANDU type reactors, the most attractive solution seems to be SEU and RU fuels utilization.

The paper aims to study the effects induced by fuel enrichment variation on spent CANDU SEU fuel photon dose rates for a Monte Carlo shielding analysis applied to spent fuel transport after a defined cooling period in the NPP pools.

Spent fuel inventories and photon source profiles are obtained by means of ORIGEN-S code. The shielding calculations have been performed by using Monte Carlo MORSE-SGC code. Both codes are included in the ORNL's SCALE 5 programs package. The photon dose rates to the shipping cask wall and in air, at different distances from the cask, have been estimated. Finally, a photon dose rates comparison for different fuel enrichments has been performed.

Key Words: CANDU-SEU spent fuel, fuel enrichment, photon dose rate, shipping cask

1 INTRODUCTION

The promises for higher nuclear fuel utilization have lead to advanced cycles development need, but all the problems associated with radioactive waste have generated an increase in the interest for these fuel cycles.

Last decade, both for operating reactors and future reactor projects, a general trend to raise the discharge fuel burnup has been registered. The fuel burnup raise associated consequences are very important: spent fuel mass reduction for 1 MWh generated electric power; actinides mass significant reduction in the spent fuel; more rarely refueling, leading to impressing raises in installed capacity utilization; about 15%-35% reduction in costs associated with nuclear fuel, for 1 MWh generated electric power. For CANDU type reactor, one of the most attractive solutions is the use of SEU (Slight Enriched Uranium) fuel. The use of SEU is the next logical step from natural uranium fuel in CANDU [1]. By enriching natural uranium from 0.7% to between 0.9 and 1.2% in Uranium-235, fuel costs are lowered because less uranium and fewer bundles are needed to fuel the reactor. This in turn reduces the quantity of used fuel and its subsequent waste management costs. Figure 1 shows the evolution of CANDU fuel projects [2].

AECL (Atomic Energy of Canada Limited), along with the Korea Atomic Energy Research Institute (KAERI), has developed CANFLEX (CANDU FLExible fuelling), an advanced fuel bundle design, to increase fuel performance and cost efficiency through improved heat transfer characteristics, and to maximize advanced fuel cycle options in CANDU reactors [3]. It will deliver many benefits (greater operating and safety margins, extended plant life, increased power and better economics) to current and future CANDU reactors - using natural uranium or other advanced fuel cycles.

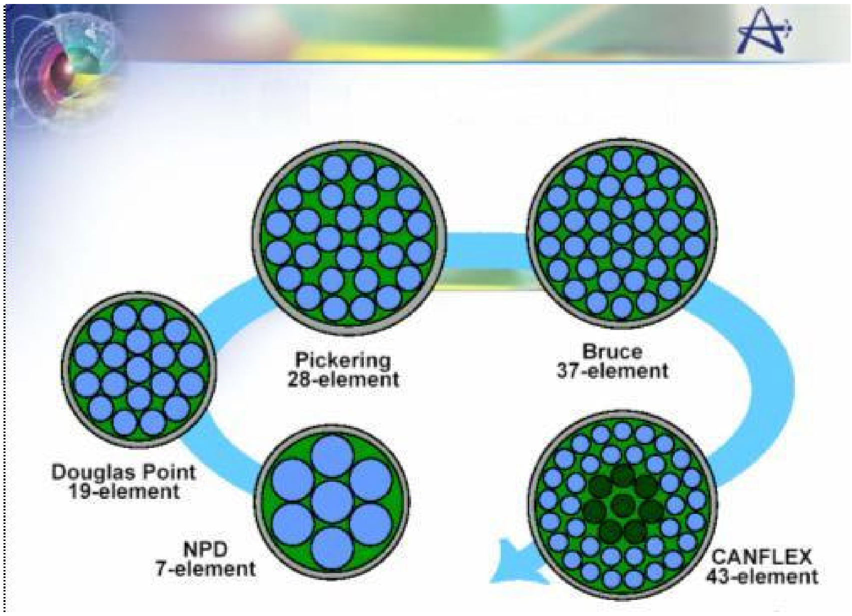

Figure 1. Evolution of CANDU fuel projects

The nuclear energy world wide development is accompanied by huge quantities of spent nuclear fuel accumulation. Taking into account for the possible impact on the human and environment, in all activities associated to nuclear fuel cycle - transport, storage, reprocessing and final disposal - the spent fuel or radioactive waste characteristics must be well known.

According to IAEA (International Atomic Energy Agency) data, more than 10 millions packages containing radioactive materials are annually world wide transported. Therefore, all the problems arisen from the maximum safe radioactive materials transport must be carefully settled. The regulations for safety radioactive materials transport must ensure the population, the authorized personnel and the environment protection against radiation exposures in a possible radioactive materials dispersion event. The protection/ safety must be oriented for maintaining individual doses, number of exposed humans and irradiation probability to the lowest values.

As Romania has decided to use the open fuel cycle, considering spent fuel as radioactive waste, the objective of Romanian spent fuel management policy is to ensure safe management of spent fuel, according to the IAEA Safety Principles. By ratifying the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management, Romania has shown its willingness to undertake all the necessary steps for achieving the required level in the spent fuel and radioactive waste safe managing. As a member of IAEA, Romania allows the radioactive materials transportation only packed in homologated containers. The shipping casks, B type, are used for: radioisotopes needed for industrial radiographies, spent nuclear fuel and highly activated radioactive wastes.

During transport, problems related to the spent fuel manipulation, shielding and cooling may occur. Therefore, at least for 6 months before sending the spent fuel bundles to final disposal or fuel reprocessing facilities, an intermediate storage inside the Nuclear Power Plant is mandatory. The spent fuel bundles discharged from the reactor core are stored in NPP cooling pools with concrete walls, stainless-steel reinforced, at adequate distances to avoid critical mass formation. As shielding material and cooling agent, the light water is used.

About 2 decades ago, Romania has opted for a heavy water nuclear power plant, CANDU6, the decision being based on the fact that the nuclear fuel and the heavy water needed for the NPP could be home-bred. Actually, Romania has only one nuclear power plant, Cernavoda NPP, equipped with 5 reactors PHWR CANDU 6 type, 705 MW(e) each. Unit 1 is in commercial operation since December, 1996, Unit 2 is under construction, the rest of 3 units being under preservation stage. In the first 3 years of commercial operation, Cernavoda NPP Unit1 has given ~10% from the total electricity produced in Romania, this percent assuming to increase at around 17-20% after 2005 (with both Unit1 and Unit2 in operation).

The Romanian specialists have been analyzed many advanced fuel cycles, the estimations giving the best chance for SEU and RU fuel cycles application [4]. In the Institute for Nuclear Research Pitesti there is an active preoccupation, with promising evaluations, for the development of a fuel bundle CANDU-SEU [5] corresponding to the CANFLEX.

1.1 The Paper Objectives

The basic tasks accomplished by the shielding calculations in a nuclear safety analysis consist in dose rates calculation, to prevent any risks both for personnel protection and impact on the environment during the spent fuel manipulation, transport and storage.

The paper aims to study the effects induced by fuel enrichment variation on CANDU-SEU spent fuel photon dose rates for a Monte Carlo shielding analysis applied to spent fuel transport after a defined cooling period in the NPP pools.

The fuel bundles projects considered here have 43 Zircaloy rods, filled with SEU fuel pellets, the fuel having different enrichment in U235. All the geometrical and material data related on the cask were considered according to the shipping cask type B model.

After a photon source profile calculation by using ORIGEN-S code, in order to perform the shielding calculations, Monte Carlo MORSE-SGC code has been used, both codes being included in the ORNL's SCALE 5 programs package. The photon dose rates to the shipping cask wall and in air, at different distances from the cask, have been estimated. Finally, a photon dose rates comparison for different fuel enrichments has been performed.

1.2 Shielding Problem General Description

1.2.1 The source of radiation

A single spent fuel bundle, CANDU-SEU type with 43 Zircaloy rods, filled with SEU fuel pellets, has been used as source of radiation. Fuel characteristics and isotopic composition were those for CANDU-SEU fuel bundle developed by the specialists from INR Pitesti [5].

Figure 2 presents the CANDU-SEU fuel bundle geometrical arrangement.

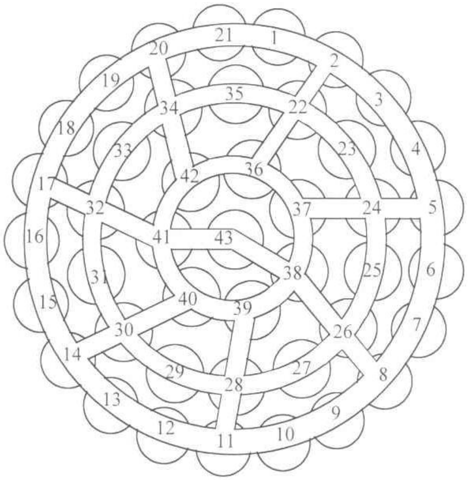

Figure 2. Geometrical arrangement for CANDU-SEU fuel bundle

The following enrichments (wt% in U235) have been used: 0.9, 0.96, 1.0, 1.2 and 1.5, respectively. The residence period inside the reactor core, corresponding to the fuel enrichment, is presented in Table 1. The table also contains the identification for each study case. After the discharge from the reactor core, the spent fuel is cooled down up to 10 years, in the NPP pools.

Table I. CANDU-SEU cases identification

Case	**1**	**2**	**3**	**4**	**5**
Enrichment [wt% U235]	0.9	0.96	1.0	1.2	1.5
Burnup[Mwd/tU]	10000	13500	20000	24000	30000
Residence time [days]	231.26	312.21	462.53	555.04	693.80

1.2.2 The shipping cask

All the geometrical and material data related to the shipping cask were considered according to the shipping cask type B model. The shipping cask prototype has been realized and tested in INR Pitesti.

1.2.3 The shielding problem theoretical model set-up

The fuel bundle was represented by 3 right, circular cylinders (1st for the central element and 7 fuel rods inner ring, the 2nd for the 14 fuel rods intermediate ring and the 3rd for the 21 fuel rods outer ring), containing a fuel, clad and structure materials homogenous mixture named "fuel", with respect for the volume conservation [6].

As regarding the shipping cask, the geometrical model consists in right circular cylinders of shielding materials with a central cavity to accommodate the source region. Figure 3 presents the geometrical configurations both for the source and the source-shipping cask assembly.

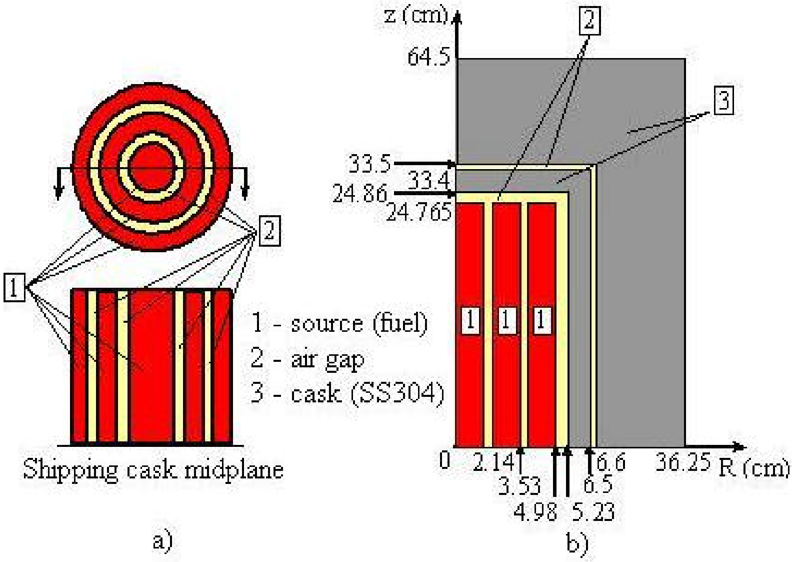

**Figure 3. 2-dimensional geometrical configuration for:
a) source; b) source-shipping cask assembly**

1.2.4 The shielding calculations

The photon dose rates calculations have been performed by means of Monte Carlo code MORSE-SGC in SAS4 sequence (3D Monte Carlo shielding calculations using an automated biasing procedure), included in the SCALE5 system (Standardized Computer Analyses for Licensing Evaluation) developed by Oak Ridge National Laboratory. The shielding calculations were preceded by photon source calculations using ORIGEN-S code, included in SCALE5 system, also. In the shielding calculations, the (27n-18g) coupled nuclear data library (27 neutron and 18 gamma energy groups) and the ANSI standard flux-to-dose conversion factors (dose rates will be in rem/h) were used. As regarding the Monte Carlo simulation, 1000 bunches of 2000 particles each, have been generated.

1.3 Spent Fuel Analysis

1.3.1 Results

Fuel characteristics and isotopic composition were considered according to [4]. Data regarding the structural materials and nuclides isotopic composition from [7] have been used. For the shielding analysis have been used: a) 7 cooling times: 182 days, 1 year, 2 years, 3 years, 4 years, 5 years, 10 years; b) 4 photon dose rates measuring points: to the cask wall and in the air, at 0.5m, 1m and 2m distance from the shipping cask.

The spent fuel is characterized by the values of concentration, radioactivity, thermal and γ power, both for nuclides and elements; other characteristic parameters are the total values for radioactivity, thermal and γ power. It would be very interesting to present all the data regarding the evolution of the spent fuel characteristics during the established cooling period, but, because of the limited space, I have to make the hard decision to indicate the references for all these characteristics and to present only a few representative data here.

In Table II, III and IV, the evolution of the spent fuel total values for radioactivity, thermal and γ power, during the cooling period, is shown. The studied cases are individualized.

Table II. Total radioactivity [Ci] evolution during the cooling period

Cooling time	182 days	1 year	2 years	3 years	4 years	5 years	10 years
Case 1							
Light elements	$2.23\ 10^{1}$	4.33	$5.78\ 10^{-1}$	$2.68\ 10^{-1}$	$1.57\ 10^{-1}$	$1.01\ 10^{-1}$	$2.38\ 10^{-2}$
Actinides	$5.47\ 10^{-1}$	$5.43\ 10^{-1}$	$5.39\ 10^{-1}$	$5.36\ 10^{-1}$	$5.32\ 10^{-1}$	$5.28\ 10^{-1}$	$5.13\ 10^{-1}$
Fission prod.	$3.42\ 10^{3}$	$1.04\ 10^{3}$	$3.89\ 10^{2}$	$2.23\ 10^{2}$	$1.50\ 10^{2}$	$1.15\ 10^{2}$	$7.05\ 10^{1}$
Case 2							
Light elements	$2.75\ 10^{1}$	5.36	$7.18\ 10^{-1}$	$3.35\ 10^{-1}$	$1.95\ 10^{-1}$	$1.26\ 10^{-1}$	$2.98\ 10^{-2}$
Actinides	$7.57\ 10^{-1}$	$7.53\ 10^{-1}$	$7.44\ 10^{-1}$	$7.37\ 10^{-1}$	$7.27\ 10^{-1}$	$7.20\ 10^{-1}$	$6.86\ 10^{-1}$
Fission prod.	$4.48\ 10^{3}$	$1.37\ 10^{3}$	$5.17\ 10^{2}$	$2.98\ 10^{2}$	$1.99\ 10^{2}$	$1.53\ 10^{2}$	$9.36\ 10^{1}$
Case 3							
Light elements	$3.63\ 10^{1}$	7.07	$9.60\ 10^{-1}$	$4.48\ 10^{-1}$	$2.62\ 10^{-1}$	$1.69\ 10^{-1}$	$4.00\ 10^{-2}$
Actinides	1.29	1.27	1.25	1.23	1.21	1.18	1.09
Fission prod.	$6.19\ 10^{3}$	$1.92\ 10^{3}$	$7.29\ 10^{2}$	$4.19\ 10^{2}$	$2.81\ 10^{2}$	$2.16\ 10^{2}$	$1.31\ 10^{2}$
Case 4							
Light elements	$3.63\ 10^{1}$	7.03	$9.65\ 10^{-1}$	$4.50\ 10^{-1}$	$2.64\ 10^{-1}$	$1.71\ 10^{-1}$	$4.04\ 10^{-2}$
Actinides	1.34	1.33	1.30	1.28	1.25	1.23	1.13
Fission prod.	$7.33\ 10^{3}$	$2.27\ 10^{3}$	$8.69\ 10^{2}$	$4.98\ 10^{2}$	$3.35\ 10^{2}$	$2.57\ 10^{2}$	$1.58\ 10^{2}$
Case 5							
Light elements	$3.59\ 10^{1}$	6.98	$9.69\ 10^{-1}$	$4.54\ 10^{-1}$	$2.66\ 10^{-1}$	$1.72\ 10^{-1}$	$4.07\ 10^{-2}$
Actinides	1.41	1.40	1.37	1.34	1.31	1.29	1.18
Fission prod.	$8.97\ 10^{3}$	$2.81\ 10^{3}$	$1.08\ 10^{3}$	$6.18\ 10^{2}$	$4.17\ 10^{2}$	$3.20\ 10^{2}$	$1.97\ 10^{2}$

Table III. Total thermal power [W] evolution during the cooling period

Cooling time	182 days	1 year	2 years	3 years	4 years	5 years	10 years
Case 1							
Light elements	$1.00\ 10^{-1}$	$1.63\ 10^{-2}$	$8.17\ 10^{-4}$	$3.14\ 10^{-4}$	$2.10\ 10^{-4}$	$1.51\ 10^{-4}$	$3.92\ 10^{-5}$
Actinides	$1.37\ 10^{-2}$	$1.37\ 10^{-2}$	$1.37\ 10^{-2}$	$1.37\ 10^{-2}$	$1.37\ 10^{-2}$	$1.37\ 10^{-2}$	$1.37\ 10^{-2}$
Fission prod.	$1.34\ 10^{1}$	3.96	1.31	$6.73\ 10^{-1}$	$4.13\ 10^{-1}$	$2.98\ 10^{-1}$	$1.86\ 10^{-1}$

Table III. Total thermal power [W] evolution during the cooling period (cont'd)

Cooling time	182 days	1 year	2 years	3 years	4 years	5 years	10 years
Case 2							
Light elements	1.24 10^{-1}	2.01 10^{-2}	1.01 10^{-3}	3.91 10^{-4}	2.62 10^{-4}	1.88 10^{-4}	4.89 10^{-5}
Actinides	1.70 10^{-2}	1.70 10^{-2}	1.70 10^{-2}	1.70 10^{-2}	1.70 10^{-2}	1.70 10^{-2}	1.70 10^{-2}
Fission prod.	1.76 10^{1}	5.25	1.74	8.98 10^{-1}	5.49 10^{-1}	3.96 10^{-1}	2.47 10^{-1}
Case 3							
Light elements	1.63 10^{-1}	2.64 10^{-2}	1.34 10^{-3}	5.25 10^{-4}	3.52 10^{-4}	2.53 10^{-4}	6.58 10^{-5}
Actinides	2.29 10^{-2}	2.29 10^{-2}	2.29 10^{-2}	2.31 10^{-2}	2.31 10^{-2}	2.31 10^{-2}	2.31 10^{-2}
Fission prod.	2.44 10^{1}	7.35	2.46	1.27	7.77 10^{-1}	5.58 10^{-1}	3.48 10^{-1}
Case 4							
Light elements	1.62 10^{-1}	2.62 10^{-2}	1.35 10^{-3}	5.28 10^{-4}	3.53 10^{-4}	2.55 10^{-4}	6.64 10^{-5}
Actinides	2.33 10^{-2}	2.33 10^{-2}	2.33 10^{-2}	2.33 10^{-2}	2.33 10^{-2}	2.33 10^{-2}	2.34 10^{-2}
Fission prod.	2.88 10^{1}	8.70	2.92	1.51	9.26 10^{-1}	6.68 10^{-1}	4.19 10^{-1}
Case 5							
Light elements	1.60 10^{-1}	2.59 10^{-2}	1.34 10^{-3}	5.32 10^{-4}	3.57 10^{-4}	2.59 10^{-4}	6.70 10^{-5}
Actinides	2.36 10^{-2}	2.36 10^{-2}	2.36 10^{-2}	2.38 10^{-2}	2.38 10^{-2}	2.38 10^{-2}	2.38 10^{-2}
Fission prod.	3.52 10^{1}	1.07 10^{1}	3.61	1.88	1.15	8.31 10^{-1}	5.23 10^{-1}

Table IV. Total γ power [W] evolution during the cooling period

Cooling time	182 days	1 year	2 years	3 years	4 years	5 years	10 years
Case 1							
Light elements	9.08 10^{-2}	1.45 10^{-2}	5.54 10^{-4}	1.97 10^{-4}	1.43 10^{-4}	1.09 10^{-4}	3.01 10^{-5}
Actinides	3.81 10^{-6}	3.78 10^{-6}	3.81 10^{-6}	3.83 10^{-6}	3.85 10^{-6}	3.87 10^{-6}	3.94 10^{-6}
Fission prod.	6.51	1.11	1.36 10^{-1}	8.82 10^{-2}	7.38 10^{-2}	6.66 10^{-2}	5.47 10^{-2}
Case 2							
Light elements	1.12 10^{-1}	1.78 10^{-2}	6.86 10^{-4}	2.46 10^{-4}	1.79 10^{-4}	1.36 10^{-4}	3.76 10^{-5}
Actinides	4.48 10^{-6}	4.46 10^{-6}	4.52 10^{-6}	4.56 10^{-6}	4.61 10^{-6}	4.65 10^{-6}	4.84 10^{-6}
Fission prod.	8.56	1.46	1.82 10^{-1}	1.19 10^{-1}	9.90 10^{-2}	8.93 10^{-2}	7.29 10^{-2}

Table IV. Total g power [W] evolution during the cooling period (cont'd)

Cooling time	182 days	1 year	2 years	3 years	4 years	5 years	10 years
Case 3							
Light elements	1.47 10^{-1}	2.33 10^{-2}	9.11 10^{-4}	3.29 10^{-4}	2.40 10^{-4}	1.83 10^{-4}	5.06 10^{-5}
Actinides	5.77 10^{-6}	5.77 10^{-6}	5.91 10^{-6}	6.05 10^{-6}	6.16 10^{-6}	6.29 10^{-6}	6.79 10^{-6}
Fission prod.	1.18 10^{1}	2.01	2.60 10^{-1}	1.70 10^{-1}	1.42 10^{-1}	1.27 10^{-1}	1.03 10^{-1}
Case 4							
Light elements	1.46 10^{-1}	2.33 10^{-2}	9.11 10^{-4}	3.33 10^{-4}	2.42 10^{-4}	1.85 10^{-4}	5.10 10^{-5}
Actinides	5.93 10^{-6}	5.95 10^{-6}	6.10 10^{-6}	6.25 10^{-6}	6.38 10^{-6}	6.51 10^{-6}	7.05 10^{-6}
Fission prod.	1.39 10^{1}	2.38	3.11 10^{-1}	2.03 10^{-1}	1.69 10^{-1}	1.52 10^{-1}	1.24 10^{-1}
Case 5							
Light elements	1.45 10^{-1}	2.29 10^{-2}	9.10 10^{-4}	3.35 10^{-4}	2.44 10^{-4}	1.86 10^{-4}	5.15 10^{-5}
Actinides	6.21 10^{-6}	6.25 10^{-6}	6.42 10^{-6}	6.57 10^{-6}	6.71 10^{-6}	6.86 10^{-6}	7.44 10^{-6}
Fission prod.	1.70 10^{1}	2.92	3.83 10^{-1}	2.51 10^{-1}	2.10 10^{-1}	1.90 10^{-1}	1.54 10^{-1}

The photon dose rates to the cask wall and in air, at different distances from the shipping cask have been estimated. Their evolution during the cooling period is shown in the following figures, for each fuel enrichment case.

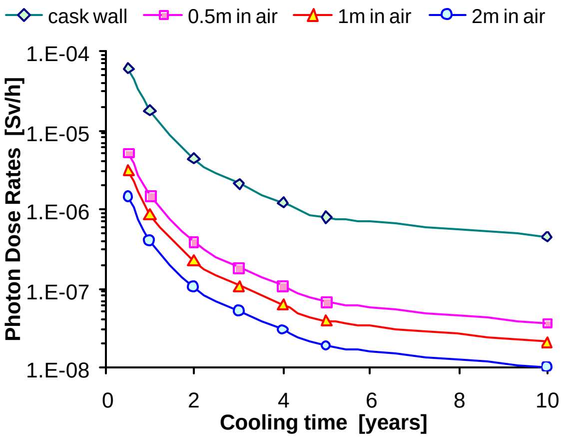

Figure 4. Photon dose rates [Sv/h]
Case 1

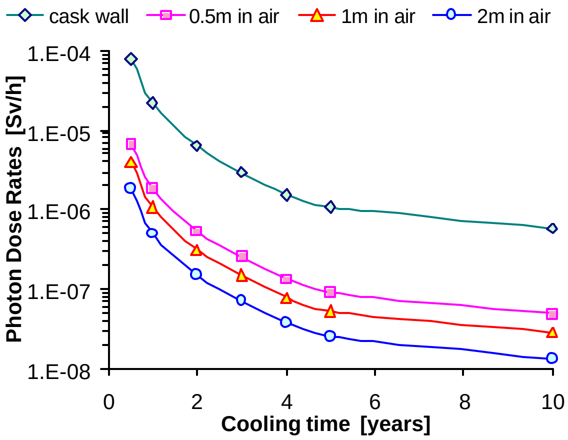

Figure 5. Photon dose rates [Sv/h]
Case 2

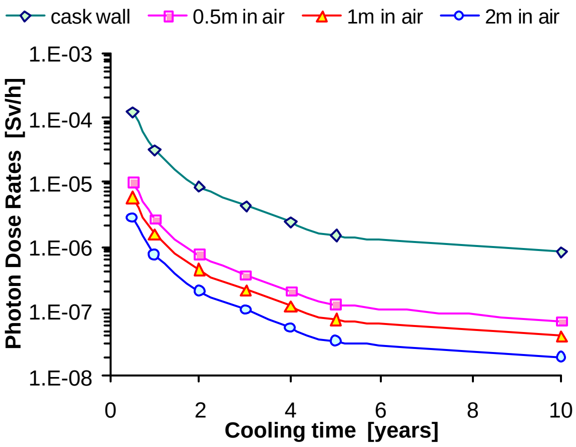

Figure 6. Photon dose rates [Sv/h] Case 3

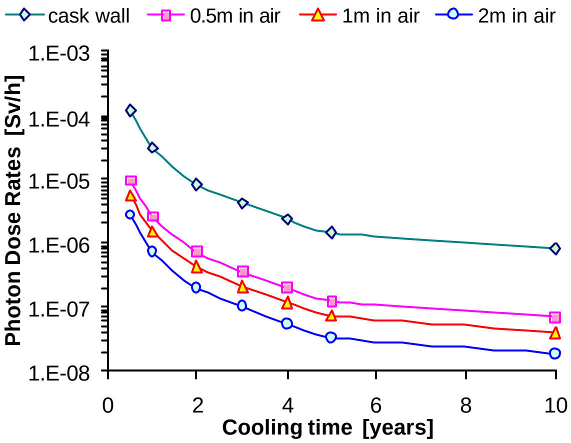

Figure 7. Photon dose rates [Sv/h] Case 4

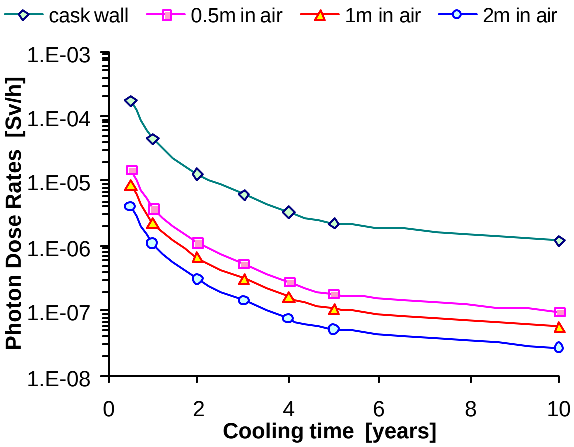

Figure 8. Photon dose rates [Sv/h] Case 5

1.3.2 Fuel enrichment variation effects

To study the effects induced by the fuel enrichment variation, we decided to perform a photon dose rates comparison for different fuel enrichment. The comparison is presented in the figures below, for different photon dose rates measuring points.

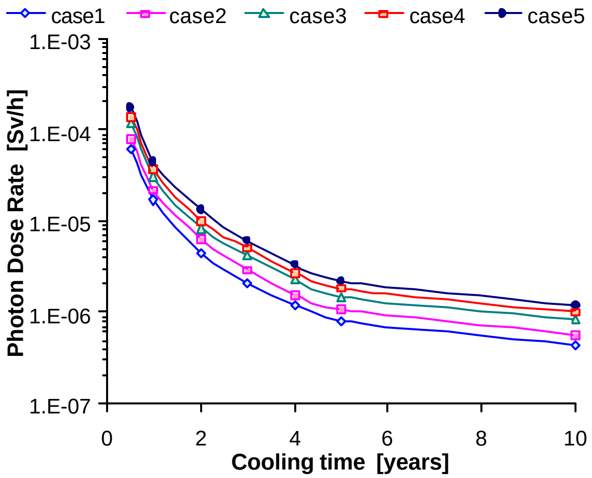

Figure 9. Photon dose rates comparison to the cask wall [Sv/h]

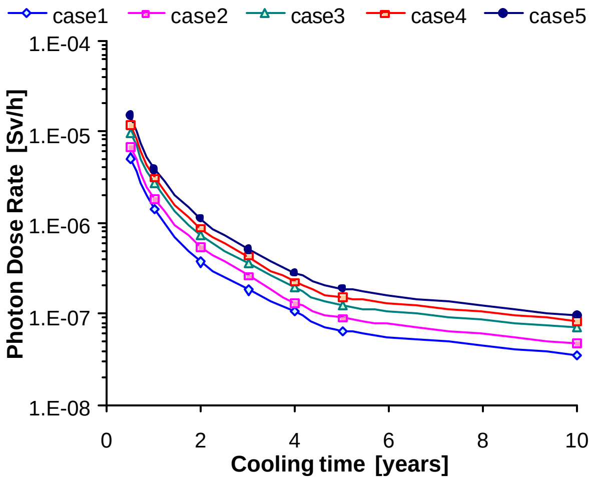

Figure 10. Photon dose rates comparison at 0.5m from the cask [Sv/h]

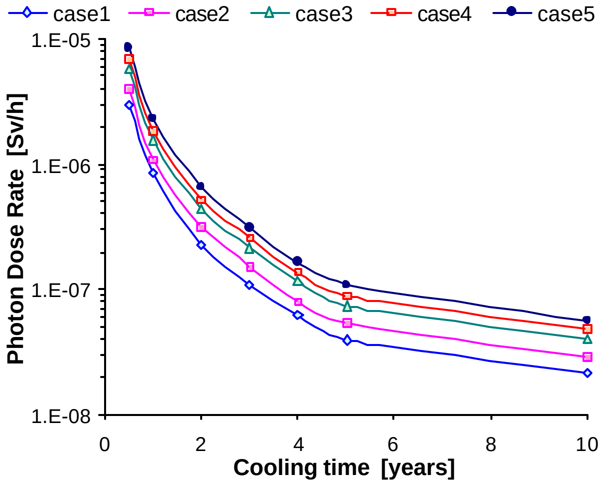

Figure 11. Photon dose rates comparison at 1m from the cask [Sv/h]

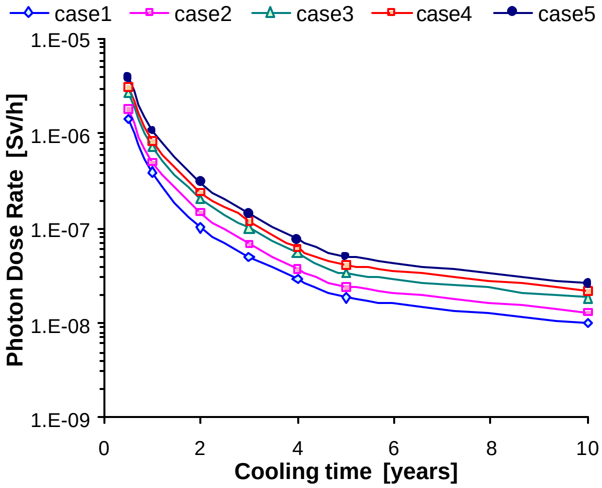

Figure 12. Photon dose rates comparison at 2m from the cask [Sv/h]

2 CONCLUSIONS

Nuclear energetic is called now to find the optimal solution in solving a series of political and public problems: nuclear accidents risks, radioactive waste disposal, nuclear weapon proliferation and nuclear plant decommissioning.

Shielding analyses are an essential component of the nuclear safety, the estimations of radiation doses in order to reduce them under specified limitation values being the main task here. According to IAEA data, more than 10 millions packages containing radioactive materials are annually world wide transported. All the problems arisen from the safe radioactive materials transport assurance must be carefully settled.

Last decade, both for operating reactors and future reactor projects, a general trend to raise the discharge fuel burnup has been world wide registered. For CANDU type reactors, the most attractive solution seems to be SEU and RU fuels utilization.

The paper aims to study the effects induced by fuel enrichment variation on CANDU-SEU spent fuel photon dose rates for a Monte Carlo shielding analysis applied to spent fuel transport after a defined cooling period in the NPP pools. The CANDU-SEU fuel bundle considered here have 43 Zircaloy rods, filled with SEU pellets, and it was developed in INR Pitesti. The SEU fuel enrichment in U235 was between 0.9 and 1.5 wt%. All the geometrical and material data related on the cask were according to the shipping cask type B model, developed in INR Pitesti.

After a photon source profile and radioactive inventory calculation by using ORIGEN-S code, in order to perform the shielding calculations, Monte Carlo MORSE-SGC code has been used, both codes being included in the ORNL's SCALE 5 programs package. The photon dose rates to the shipping cask wall and in air, at different distances from the cask, have been estimated. Finally, in order to study the effects of fuel enrichment variation, a photon dose rates comparison for different fuel enrichments has been performed.

One first observation is that the fuel burnup raise is associated both with spent fuel mass reduction for 1 MWh generated electric power and actinides mass significant reduction in the spent fuel.

The spent fuel characteristics (radioactivity, thermal and γ power) increase with the fuel enrichment raise, but the calculated relative differences between successive cases does not exceed 16%.

For all cases, the estimated photon dose rates values for spent fuel transport are small, allowing a shipping cask safe manipulation. The dose rates to the cask wall decrease from 10^{-5} Sv/h (first year from discharge) to 10^{-6} Sv/h (after 2 years of cooling) and reach 10^{-7} Sv/h (after 10 years of cooling). At different distances from the shipping cask, the corresponding values were one or two degrees smaller.

As regarding the effect of fuel enrichment variation on the photon dose rates, the corresponding relative differences were: 21% - 26% Case2 against Case 1; 25% - 31% Case 3 against Case 2; 14% - 18% Case4 against Case 3; 16% - 21% Case 5 against Case 4. The most significant effects occur between 0.96% and 1% fuel enrichment in U235.

For Cernavoda NPP Unit3, CANDU6 reactor is already an option. For the rest of 2 units, the principal candidate to CANDU 6 project seems to be another AECL project, namely CANDU NG (Next Generation) or CANDU ACR-700 (Advanced CANDU Reactor). Taking into account for the Romania future integration in European Union, another possibility is the option for an advanced PWR reactor. The opportunity to apply one of these solutions stands both in provability and desire to invest.

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