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THE KEY-ROLE OF SHIELDING ANALYSIS IN ADVANCED CANDU FUEL BUNDLES NUCLEAR SAFETY IMPROVEMENT FOR SOME ACCIDENTAL CRITICALITY SCENARIOS

Cristina Alice Margeanu, Andrei Rizoiu, Gheorghe Olteanu

Institute for Nuclear Research Pitesti, No. 1 Campului Street, 115400-Mioveni, Romania, malice@scn.ro

Abstract

The paper aims to present the source term and photon dose rates estimation for advanced CANDU fuel bundles in some accidental criticality scenarios. As reference, the CANDU standard fuel bundle has been used. The scenarios take into account for a very short-time irradiated or spent fuel bundles for some configurations closed to criticality. In order to estimate irradiated fuel characteristic parameters and radiation doses, the ORNL's SCALE5 codes ORIGEN-S and Monte Carlo MORSE-SGC have been used. The paper includes the irradiated fuel characteristic parameters comparison for the considered CANDU fuel bundles, providing also a comparison between the corresponding radiation doses.

Keywords: *advanced CANDU fuel*, *accidental criticality scenarios*, *fuel activity*, *fuel thermal power*, *fuel gamma energy*, *radiation doses*

1. INTRODUCTION

In the Romanian Government policy, the nuclear energetic is considered as optimal solution for the national energetic needs. In 2002, the Government decided to approve the Nuclear National Strategy for the next 50 years [1]. The Fundamental Objective specifies that in 2025-2050 period Romanian NPPs must provide (20-40) % from the total electricity generated in Romania, with respecting both of the competitive costs conditions and the nuclear safety assurance at international agreed standards.

About 4 decades ago, Romania made the option for a heavy water nuclear power plant, CANDU6 type, located at Cernavoda, the decision being based on the fact that the needed nuclear fuel and heavy water could be home-bred. The Cernavoda NPP is equipped with 5 reactors PHWR CANDU 6 type, CANDU 6 standard series - 706 MW(e) each. Unit 1 is in-service since December, 1996, Unit 2 began the commercial operation in October, 2007, Units

3 and 4 are under construction, the projected in-service being 2012-2014, and Unit 5 is under preservation stage.

In present, 34 CANDU reactors are in-service around the world in: Argentina, Canada, India, Pakistan, People's Republic of China, Republic of Korea and Romania, [2]. In Figure 1 CANDU 6 type reactors operating around the world are presented.

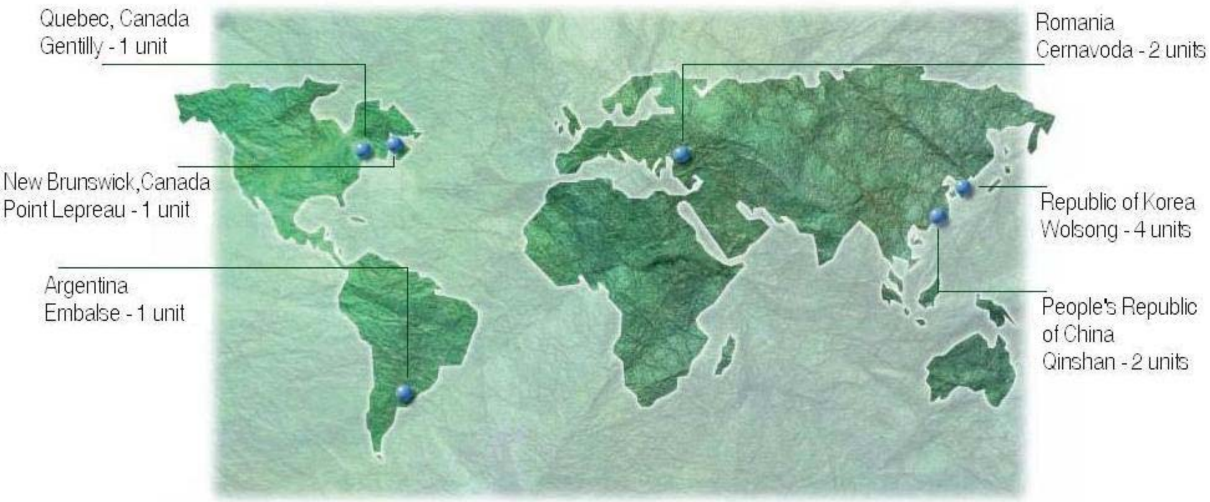

Figure 1. CANDU 6 reactors operating around the world, [2]

In about a decade of commercial operation, Cernavoda NPP Unit1 has given ~10% from the total electricity produced in Romania, this percent assuming to be around 20% after 2007 (with both Unit1 and Unit2 in operation) [3], as Figure 2 illustrates.

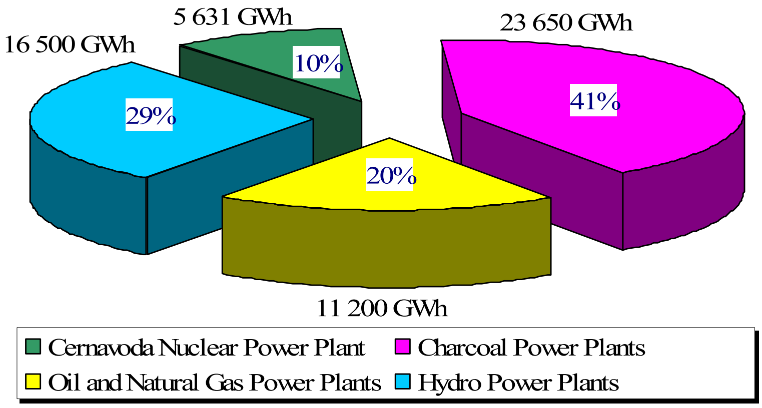

Figure 2. Coarse Power structure in Romania, end of 2006, [3]

At the end of 2006, Unit1 has registered an average capacity factor of 91.37%, being on the forth position from the world wide CANDU 6 type NPP. Over the year 2006, Cernavoda NPP-Unit 1 generated an amount of electricity of 5,631 GWh, out of which 5,118 GWh were supplied to National Electric System [4], and thermal energy of about 44,000 Gcal, respectively. Every year, Unit 1 operation allows avoiding the release of 4 million tones CO2 in the atmosphere and the import of about 1.4 million tones fluid fuels [4].

In the last 15 years, both for operating reactors and future reactor projects, a general trend to increase discharge fuel burnup (energy released by the bundle) has been world wide 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 each MWh generated electric power. For CANDU reactors, one of the most attractive solutions seems to be SEU (Slight Enriched Uranium) fuels utilization. By enriching natural uranium from 0.7% to (0.9 - 1.2) % in U235, 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. AECL (Atomic Energy of Canada Limited), along with the KAERI (Korea Atomic Energy Research Institute), 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, including greater operating and safety margins, extended plant life, better economics and increased power [5].

Romanian specialists have been analyzed many advanced fuel cycles, the estimations giving the best chance for SEU and RU fuel cycles application. In INR Pitesti there is an active preoccupation, with promising evaluations, for the development of a fuel bundle CANDU SEU type, similar to Canadian CANFLEX.

The paper aims to estimate the irradiated fuel characteristic parameters for some hypothetical criticality scenarios and the radiation doses characterizing the fuel bundle transport after a defined residence period inside the reactor core followed by a cooling period in the NPP pools up to 10 years. Two types of fuel bundles have been considered, as follows: CANDU standard fuel bundle [6] and the Romanian CANDU SEU43 fuel bundle [7], respectively. The source term assessment and fuel characteristic parameters estimation were done by means of ORIGEN-S code. In order to estimate the corresponding radiation doses, the Monte Carlo MORSE-SGC code was used, both codes being included in Oak Ridge National Laboratory's SCALE 5 (Standardized Computer Analyses for Licensing Evaluation) programs package [8]. A comparison between CANDU SEU study cases was obtained, the results for CANDU standard fuel bundle being used as reference.

2. SHIELDING ANALYSES

2.1. Shielding Problem - General Description

From the nuclear safety analysis point of view is very important to know the fresh and spent fuel characteristic parameters: radionuclides inventory – masses and activities, decay heats, neutron and photon generation rates, during both the residence period in the reactor core and the cooling period inside the NPP spent fuel bay (concrete walls, stainless-steel reinforced; light water used as shielding material and cooling agent).

The following hypothetical accidental criticality scenarios were considered: a) fuel bundle extraction from the reactor and immersion in the NPP spent fuel bay, following a fuel bundle damage identification at k-eff (effective multiplication constant) maximum value – ″d″ (damaged) upper index for identification; b) spent fuel bundles extraction from the reactor when k-eff reaches the critical value (=1.0) and immersion in the NPP spent fuel bay – ″s″ (spent) upper index for identification. For both cases, a cooling period up to 10 years in the NPP damaged or spent fuel bay is considered. Figure 3 presents the k-eff evolution with the irradiation period (WIMS calculation) for the considered fuel bundle projects.

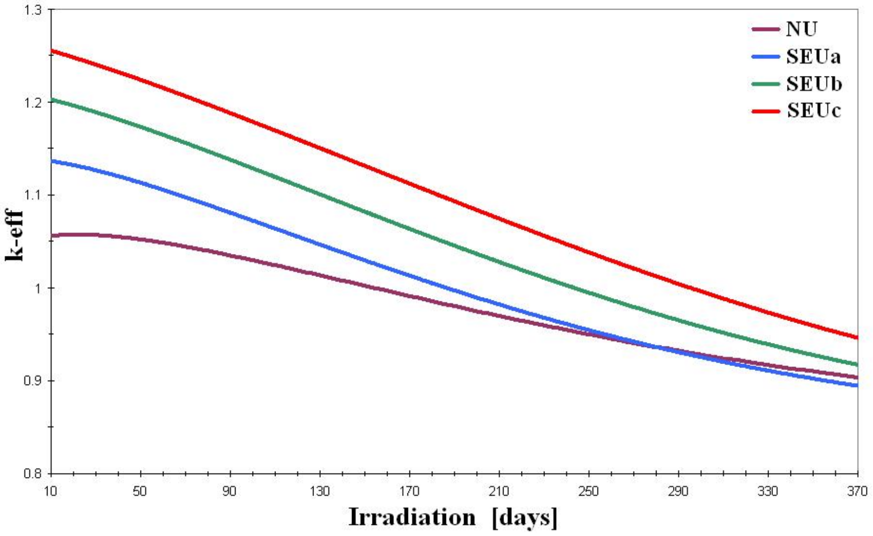

Figure 3. K-eff evolution with irradiation time

Two CANDU fuel bundle projects have been analyzed, namely: CANDU standard fuel bundle with 37 fuel rods filled with natural uranium dioxide pellets (NU), and CANDU SEU43 fuel bundle with 43 fuel rods filled with SEU pellets (SEU). For SEU, the following enrichments in U235 were considered: 0.9 wt% (SEUa), 1.1 wt% (SEUb) and 1.3 wt% (SEUc), respectively. Table 1 permits an easy identification for each study case. Along with the case notation, the characteristic data about the enrichment in U235, burnup and residence time inside the reactor core, respectively, are presented.

Table 1. Study cases identification

Type of fuel	**Study Case**	**Enrichment** [wt% U235]	**Burnup** [MWd/tU]	**Residence time** [days]
Short-time irradiated fuel	NU$^{(d)}$	0.72	797.27	23
	SEUa$^{(d)}$	0.9	648.60	15
	SEUb$^{(d)}$	1.1	648.60	15
	SEUc$^{(d)}$	1.3	648.60	15
Spent fuel	NU$^{(s)}$	0.72	5,362	154.688
	SEUa$^{(s)}$	0.9	8,051	186.198
	SEUb$^{(s)}$	1.1	10,507	242.997
	SEUc$^{(s)}$	1.3	12,756	295.0

The geometrical arrangement of the bundle rods (see Figure 4) consists in 3 concentrical rings (for NU – 6 elements on the inner ring, 12 elements on the intermediate ring and 18 elements on the outer ring, respectively; for SEU – 7 elements on the inner ring, 14 elements on the intermediate ring and 21 elements on the outer ring, respectively) and one central element, [9].

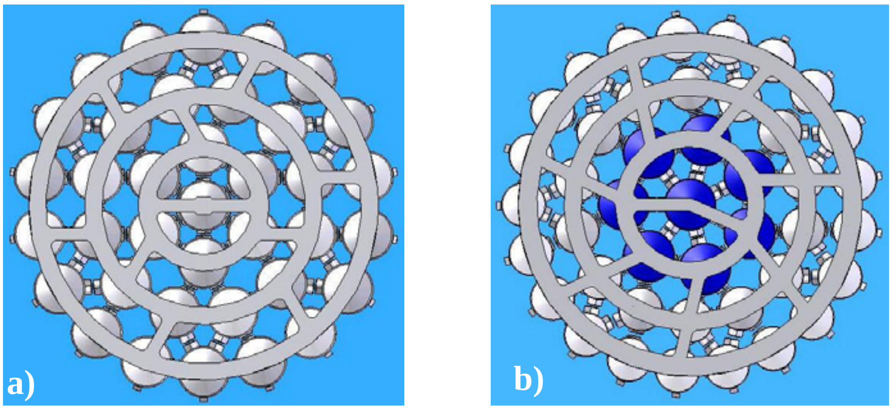

Figure 4. Geometrical arrangement for the considered CANDU fuel bundles:
a) CANDU standard; b) CANDU SEU43

For CANDU standard fuel bundle, all the 37 fuel rods have the same diameter, no matter if they are central or on inner, intermediate or outer ring. The CANDU SEU43 fuel bundle distinctive characteristic is that the 8 inner rods (the central element and the other 7 rods from the inner ring) are thicker than the CANDU standard fuel element, the other 35 outer rods (14 rods from the intermediate ring and 21 rods from the outer ring) being thinner than the CANDU standard fuel element.

Both for short-time irradiated and spent fuel bundles, the considered cooling period in the NPP spent fuel bay was up to 10 years, the residence periods being those specified in Table 1. Fuel characteristics and isotopic composition were considered according to [6, 7].

Radionuclide inventory and irradiated fuel characteristics have been obtained by taking into account for all relevant isotopes generation and depletion during both the irradiation and cooling phases of the fuel history. 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 were used. As regarding the Monte Carlo simulation, 1000 bunches of 2000 particles each, have been generated.

2.2. Short-time irradiated fuel analysis

The considered scenario takes into account for a fuel bundle damaging identification after k-eff reaches the maximum value, followed by fuel bundle extraction and storage in the NPP defected fuel bay.

In Table 2 the evolutions of the short irradiated fuel total radioactivity, total thermal power and total gamma energy during the specified cooling period are presented.

Table 2. Short-irradiated fuel characteristic parameters evolution

Parameter	**Study Case**	**Cooling time** [years]				
		0.5	***1***	***3***	***5***	***10***
Total Radioactivity [Ci]	NU[d]	7,924.0	2,506.0	550.7	280.6	169.9
	SEUa[d]	6,629.0	2,050.0	443.4	227.0	138.4
	SEUb[d]	6,655.0	2,053.0	443.5	227.7	139.3
	SEUc[d]	6,672.0	2,056.0	443.7	228.3	140.0
Total Thermal Power [W]	NU[d]	31.15	9.63	1.71	0.75	0.47
	SEUa[d]	26.05	7.86	1.36	0.60	0.38
	SEUb[d]	26.15	7.86	1.35	0.60	0.38
	SEUc[d]	26.21	7.86	1.35	0.60	0.38
Total Gamma Energy [W]	NU[d]	15.04	2.59	2.27	0.17	0.13
	SEUa[d]	12.66	2.16	1.76	0.13	0.11
	SEUb[d]	12.71	2.17	1.76	0.13	0.11
	SEUc[d]	12.75	2.17	1.76	0.13	0.11

Table 3 contains the corresponding relative differences for SEU - SEU and SEU - NU comparisons.

Table 3. Short-irradiated fuel characteristic parameters relative differences

Comparison	**Relative differences** [%]		
	Total Radioactivity	**Total Thermal Power**	**Total Gamma Energy**
NU - SEU43a	18.66	19.21	19.73
NU - SEU43b	18.42	19.30	19.88
NU - SEU43c	18.24	19.36	19.98
SEU43b – SEU43a	0.30	- 0.11	- 0.20
SEU43c – SEU43a	0.51	- 0.19	- 0.33
SEU43c – SEU43b	0.22	- 0.08	- 0.13

2.3. Spent fuel analysis

The residence periods for spent fuel cases were thus selected to justify the accidental criticality scenario. Figure 5 illustrates the photon source profiles during cooling period for NU and SEUc cases.

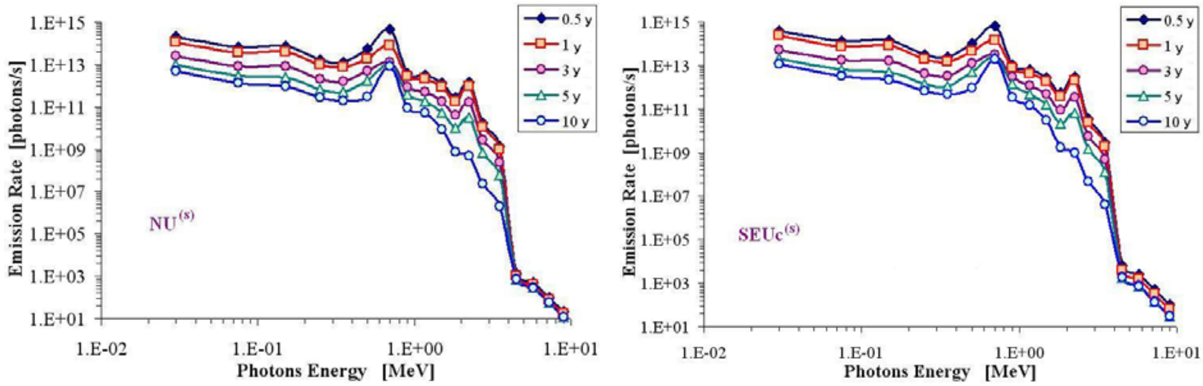

Figure 5. Photon source profile during cooling period

In Figure 6 the evolution of spent fuel total radioactivity, total thermal power and total gamma energy during the specified cooling period, for the considered CANDU fuel projects is illustrated.

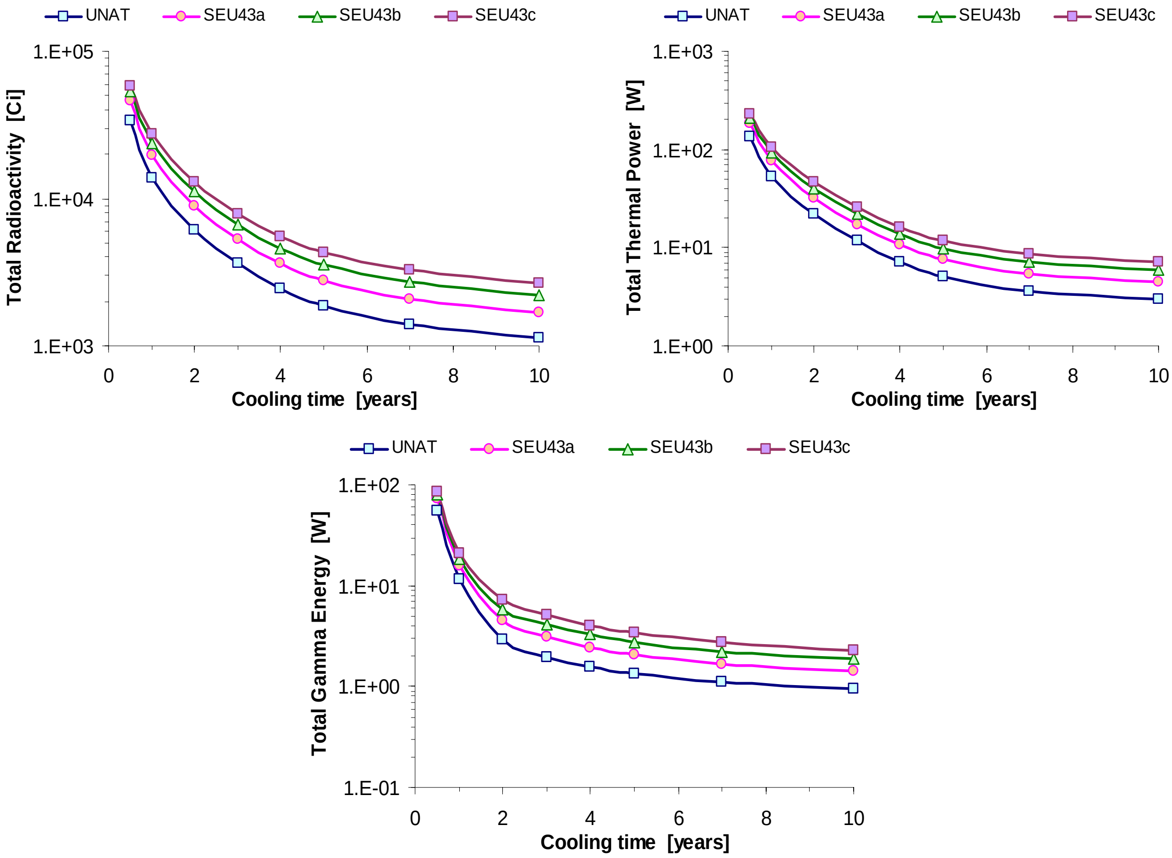

Figure 6. Spent fuel characteristic parameters evolution during cooling period

The corresponding relative differences obtained for SEU - SEU and SEU - NU comparisons are presented in Table 4. The considered comparisons are completed by the relative differences obtained for some long life nuclides radioactivity (see Table 5).

Table 4. Spent fuel characteristic parameters relative differences

Comparison	**Relative differences** [%]		
	Total Radioactivity	**Total Thermal Power**	**Total Gamma Energy**
SEU43a - NU	31.23	31.68	32.99
SEU43b - NU	45.13	45.64	47.25
SEU43c - NU	52.89	53.70	55.42
SEU43b – SEU43a	20.29	20.53	21.62
SEU43c – SEU43a	32.02	32.55	33.99
SEU43c – SEU43b	14.87	15.15	16.14

Table 5. Long life nuclides radioactivity relative differences

Comparison	**Relative differences** [%]					
	Kr85	***Sr90***	***Cs137***	***Pu239***	***Pu240***	***Pu241***
SEU43a - NU	32.43	32.48	32.60	3.42	27.81	39.43
SEU43b - NU	48.42	48.66	48.32	4.29	39.48	53.44
SEU43c - NU	57.78	58.14	57.40	4.47	45.59	60.20
SEU43b – SEU43a	23.67	23.95	23.33	0.90	16.17	23.13
SEU43c – SEU43a	37.52	38.00	36.80	1.09	24.62	34.29
SEU43c – SEU43b	18.14	18.47	17.57	0.19	10.09	14.51

2.4. Radiation doses estimation

All the geometrical and material data were considered according to the shipping cask type B model, whose prototype (see Figure 7) has been designed, manufactured and tested in INR Pitesti, [10].

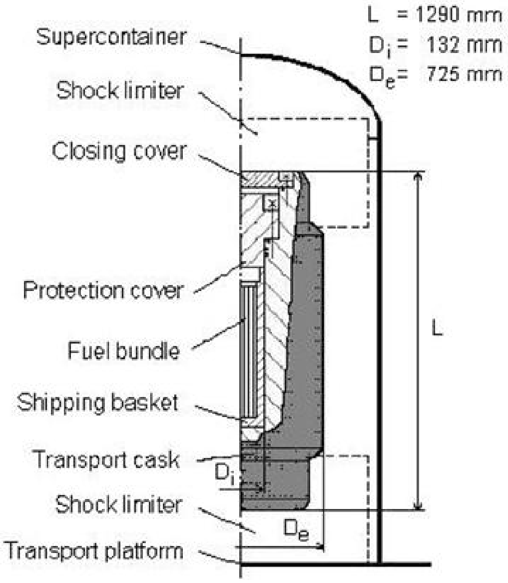

Figure7. Spent fuel shipping cask

Must to be mentioned that the shipping cask used for the spent fuel radiation doses estimation performed in the paper is authorized only for CANDU standard spent fuel transport. If the Romanian authorities will make the decision to use advanced fuel in our CANDU reactors, then a corresponding shipping cask will be proposed for authorization.

The geometrical model for the shipping cask consists in right circular cylinders of shielding materials with a central cavity to accommodate the source region [10]. In Figure 8, the geometrical configurations both for the source and the source-shipping cask assembly are presented.

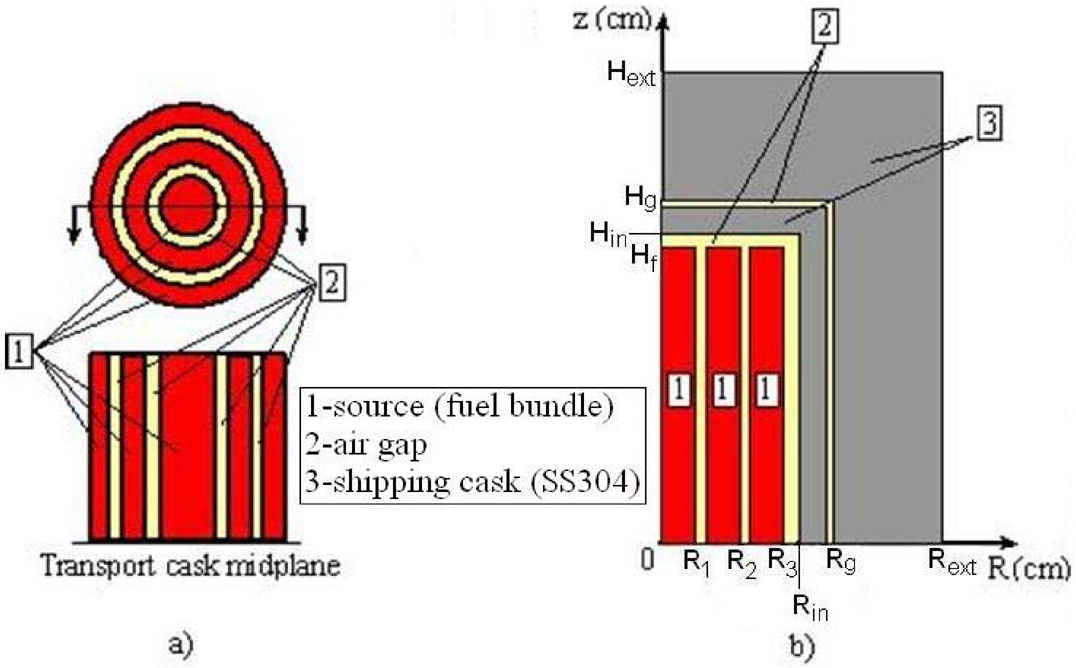

Figure 8: Two-dimensional geometrical configuration for:
a) source; b) source-shipping cask assembly

The photon dose rates to the shipping cask wall have been estimated. Their evolution during the cooling period is shown in the following tables, both for short-time irradiated fuel and spent fuel.

Table 6. Short-time irradiated fuel dose rates to the shipping cask wall, [Sv/h]

Short-time irradiated fuel	**Cooling period [years]**				
	0.5	***1***	***3***	***5***	***10***
NU	9.58×10^{-5}	5.06×10^{-5}	8.52×10^{-6}	2.16×10^{-6}	2.72×10^{-7}
SEUa	6.77×10^{-5}	3.31×10^{-5}	5.47×10^{-6}	1.05×10^{-6}	1.64×10^{-7}
SEUb	5.96×10^{-5}	3.34×10^{-5}	5.42×10^{-6}	1.11×10^{-6}	1.55×10^{-7}
SEUc	7.86×10^{-5}	3.15×10^{-5}	5.57×10^{-6}	1.34×10^{-6}	1.56×10^{-7}

Table 7. Spent fuel dose rates to the shipping cask wall, [Sv/h]

Spent fuel	**Cooling period [years]**				
	0.5	***1***	***3***	***5***	***10***
NU	5.60×10^{-4}	3.00×10^{-4}	5.58×10^{-5}	1.32×10^{-5}	2.37×10^{-6}
SEUa	5.66×10^{-4}	3.58×10^{-4}	6.28×10^{-5}	1.51×10^{-5}	2.51×10^{-6}
SEUb	6.48×10^{-4}	4.01×10^{-4}	8.06×10^{-5}	1.90×10^{-5}	3.51×10^{-6}
SEUc	7.94×10^{-4}	4.80×10^{-4}	1.36×10^{-4}	2.24×10^{-5}	4.38×10^{-6}

In the tables below the relative differences for the considered cases are shown.

Table 8. Short-time irradiated fuel relative differences to the cask wall, [%]

Comparison	**Cooling period [years]**				
	0.5	***1***	***3***	***5***	***10***
NU - SEU43a	29.39	34.50	35.79	51.30	39.90
NU - SEU43b	37.77	33.95	36.34	48.84	42.91
NU - SEU43c	17.97	37.76	34.58	38.07	42.84
SEU43b – SEU43a	-13.47	0.84	-0.87	4.79	-5.28
SEU43c – SEU43a	13.92	-5.24	1.85	21.36	-5.14
SEU43c – SEU43b	24.13	-6.13	2.69	17.40	0.13

Table 9. Spent fuel relative differences to the cask wall, [%]

Comparison	**Cooling period [years]**				
	0.5	***1***	***3***	***5***	***10***
SEU43a - NU	1.22	16.05	11.14	12.67	5.46
SEU43b - NU	13.64	25.07	30.73	30.83	32.35
SEU43c - NU	29.52	37.40	58.96	41.28	45.82
SEU43b – SEU43a	12.58	10.75	22.04	20.80	28.44
SEU43c – SEU43a	28.65	25.43	53.81	32.77	42.70
SEU43c – SEU43b	18.39	16.45	40.75	15.11	19.92

3. CONCLUSIONS

By increasing the CANDU fuel enrichment in U235 a rise in the discharge fuel burnup is registered, associated with the actinide mass reduction in the spent fuel. Meanwhile, for the same amount of electric energy generated, CANDU SEU fuel cycle produces a smaller, but more radioactive weight of spent fuel than the natural UO2 one. The spent fuel characteristic parameters sustain this affirmation.

For the short-time irradiation conditions, CANDU SEU fuel characteristic parameters are less than the natural UO2 ones, the relative differences being, as follows: 18% in radioactivity, 19% in thermal power and about 20% in gamma energy.

For the fuel irradiation until k-eff reaches the critical value, CANDU SEU fuel characteristic parameters are greater than the natural UO2 ones, the following relative differences being obtained: (31.23-52.89)% in radioactivity, (31.68-53.70)% in thermal power and (32.99-55.42)% in gamma energy, respectively.

The estimated photon dose rates values for both short-time irradiated and spent fuel transport are small, according to Romanian and internationally agreed safety standards.

For short-time irradiated fuel, CANDU SEU fuel photon dose rates to the cask wall are smaller than the natural UO2 ones. Estimated dose rates decrease from 10-5 Sv/h (in first year of cooling) to 10-6 Sv/h (after 2 – 5 years of cooling) and reach 10-7 Sv/h (after 10 years of cooling).

For the fuel irradiation until keff reaches the critical value, CANDU SEU fuel photon dose rates to the cask wall are greater than the natural UO2 ones. Estimated dose rates decrease from 10-4 Sv/h (in first year of cooling) to 10-5 Sv/h (after 2 – 5 years of cooling) and reach 10-6 Sv/h (after 10 years of cooling).

In order to allow the spent fuel decay and reactivity decreasing, at least for 6 months before sending the spent fuel bundles to final disposal or fuel reprocessing facilities, an intermediate storage inside the NPP spent fuel bay is mandatory. In the NPP spent fuel bay the irradiated fuel bundles discharged from the reactor core are stored on racks, at adequate distances to avoid critical mass formation, light water being used as shielding material and cooling agent.

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.

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