

## MONTE CARLO METHOD APPLICATION TO SHIELDING CALCULATIONS

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CANDU spent fuel discharged from the reactor core contains Pu, so it must be stressed in two directions: tracing for the fuel reactivity in order to prevent critical mass formation and personnel protection during the spent fuel manipulation.

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

To perform photon dose rates calculations the Monte Carlo MORSE-SGC code incorporated in SAS4 sequence from SCALE system was used.

The paper objective was to obtain the photon dose rates to the spent fuel transport cask wall, both in radial and axial directions.

As source of radiation one spent CANDU fuel bundle was used. All the geometrical and material data related to the transport cask were considered according to the shipping cask type B model, whose prototype has been realized and tested in the Institute for Nuclear Research Pitesti.

### INTRODUCTION

CANDU reactor is a PHWR reactor cooled and moderated with heavy water that uses natural Uranium as fuel. Here are some characteristics for CANDU reactor:

- using of natural Uranium makes the fuel cycle simple, economic, without needs for reprocessing.
- the heavy water allows a high reproduction coefficient (0.7 – 0.9) for fuel with thermal neutrons.
- the refueling is done on line during the operation, spent or damaged fuel bundles being withdrawn without needs for a shutdown.
- the moderator is completely separated from the heat transport system, involving an easy operation for all the regulating and control systems.

The fuel irradiated in a CANDU type reactor contains Plutonium, so must be stressed in two directions: tracing of the fuel criticality in order to avoid critical mass formation and the personnel protection during the manipulation of the spent fuel. These are the reasons imposing the spent fuel temporary storage inside the NPP for a period about 6 months before sending to storage facilities. For this temporary storage the NPP is equipped with special pools, concrete and stainless steel walls, where the spent fuel is stored under water.

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

The paper goal is to obtain the photon dose rates to the wall of a spent fuel transport cask, both in radial and axial directions, after a defined cooling period.

These calculations have been performed by means of Monte Carlo code MORSE-SGC /1/ included in the SCALE programs package, developed by ORNL. For the radiation source calculation, the ORIGEN-S code /2/ included in SCALE system was used. Previously, the shielding problems were solved in the Institute for Nuclear Research Pitesti by means of another calculation recipe, namely: using of ORIGEN-JR code (the Japanese version of

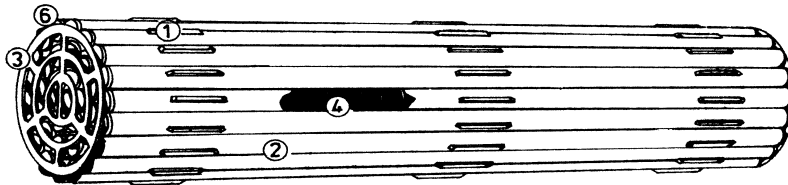
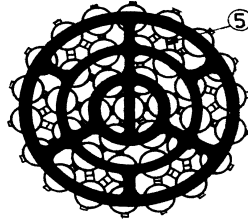
ORIGEN2 code) for photon source calculation, followed by shielding calculations performed with PESHIE-3 code, that uses the point kernel method.

**SHIELDING PROBLEM DEFINITION**

**1. The source**

As source of radiation a single spent fuel bundle was considered. Fuel characteristics and isotopic composition were those for CANDU fuel bundles used at Cernavoda NPP /3/. 37 zircalloy rods, filled with natural UO<sub>2</sub> pellets compose the CANDU fuel bundle. The geometrical arrangement of the bundle consists in 3 concentric rings (of 6, 12 and 18 rods, respectively) and 1 central rod (Figure 1).

- 1 zircalloy bearing pads
- 2 zircalloy sheath
- 3 zircalloy end plate
- 4 UO<sub>2</sub> pellets
- 5 zircalloy spacers
- 6 zircalloy end cap

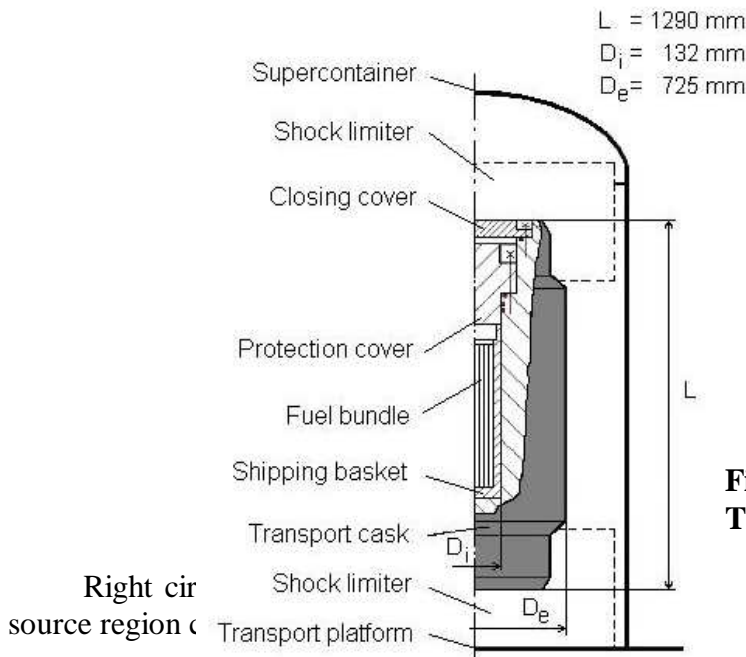


**Figure 1 - CANDU Fuel Bundle with 37 rods**

The fuel bundle was irradiated in the reactor core for 291.16 days, followed by a cooling period of 360 days before the introduction in the transport cask.

**2. The transport cask**

All the geometrical and material data related to the transport cask were considered according the shipping cask type B model, whose prototype has been realized and tested in INR Pitesti /4/. In Figure 2 is presented a transversal section through the cask.



**Figure 2  
The spent fuel transport cask**

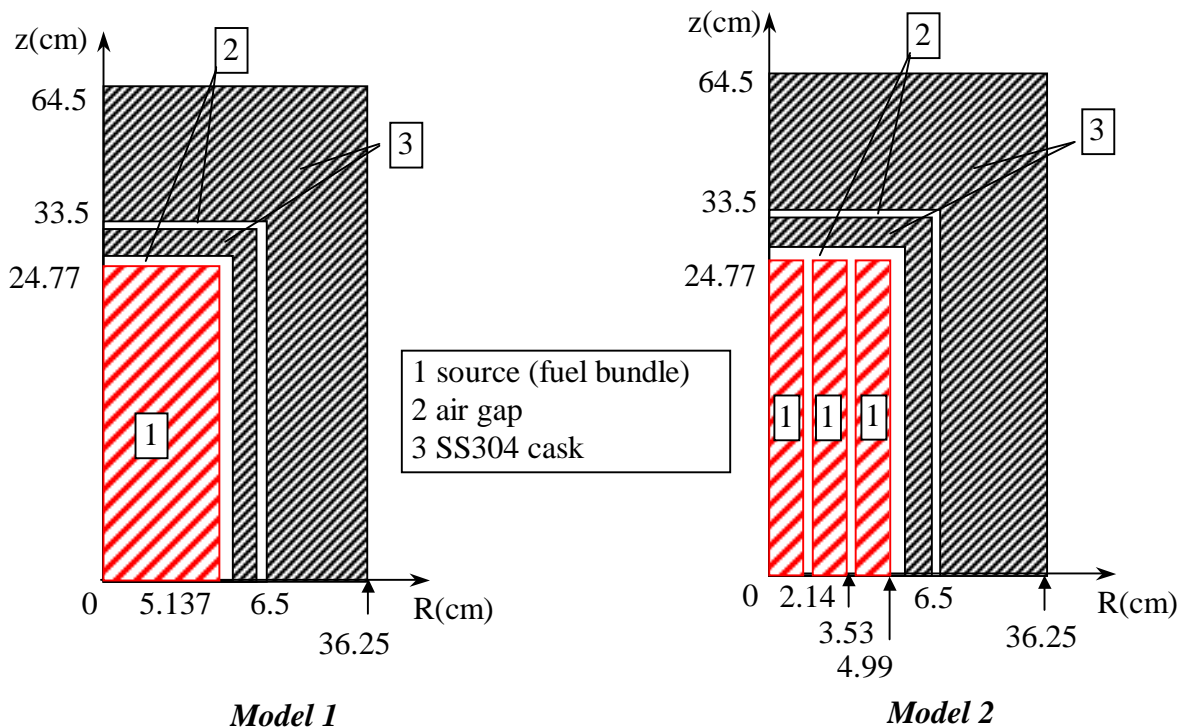
central cavity to accommodate the transport cask.

### 3. Shielding calculations

For fuel bundle representation 2 geometrical models were used, both with respect of the volume conservation:

- Model 1: the fuel bundle was represented as a single cylinder containing a homogenous mixture of fuel, clad and structure materials, named further “fuel”
- Model 2: the fuel bundle was represented as 3 concentric cylinders (1<sup>st</sup> for the central rod and the 6 fuel rods inner ring, 2<sup>nd</sup> for the 12 fuel rods ring, 3<sup>rd</sup> for the 18 fuel rods outer ring). All these 3 cylinders contain the homogenous mixture of fuel, clad and structure materials, named further “fuel”.

Figure 3 presents the geometrical configurations for the source-cask assembly.



**Figure 3 - Source-cask assembly geometrical configurations**

In Table 1 the homogenized “fuel” densities for both fuel bundle models described before, are given.

**Table 1 – Homogenized “fuel density” for fuel bundle**

Element	Density [g/cm <sup>3</sup> ]	
	Model 1	Model 2
Uranium 234	$2.487 \times 10^{-4}$	$2.800 \times 10^{-4}$
Uranium 235	$3.332 \times 10^{-2}$	$3.752 \times 10^{-2}$
Uranium 238	4.661	5.248
Oxygen	$6.312 \times 10^{-1}$	$7.108 \times 10^{-1}$
Zircalloy 4	$5.346 \times 10^{-1}$	$6.019 \times 10^{-1}$

To perform the shielding calculations, SAS4 standard sequence from SCALE system was used. SAS4 sequence allows 3-D shielding calculations using Monte Carlo method and an automatic biasing procedure for a spent fuel storage and shipping cask /5/.

The automated biasing procedure basically involves:

- calculation of adjoint fluxes of a simplified 1-D slab model of a cask using XSDRNPM functional module,
- processing of the adjoint fluxes into different biasing parameters by SAS4 control module,
- application of the biasing parameters to particle random walk in MORSE calculation,
- estimation of radiation doses exterior to the cask.

To calculate dose rates by means of MORSE-SGC code, the (27n-18g) coupled library was used for nuclear data. Bunches of 1000 particles were generated, and ANSI standard flux-to-dose conversion factor (meaning that the dose rates are in rem/h) was applied.

## RESULTS AND DISCUSSIONS

In Tables 2 and 3, the photon dose rates to the cask wall after a cooling period of 90 days and 360 days, respectively, are presented.

**Table 2 - Photon dose rates after 90 days of cooling**

Geometrical model	Model 1		Model 2	
Calculation type	radial	axial	radial	Axial
Photon dose rate [rem/h]	$1.82 \times 10^{-1}$	$5.83 \times 10^{-3}$	$2.21 \times 10^{-1}$	$5.71 \times 10^{-3}$

**Table 3 - Photon dose rates after 360 days of cooling**

Geometrical model	Model 1		Model 2	
Calculation type	radial	axial	radial	Axial
Photon dose rate [rem/h]	$6.69 \times 10^{-2}$	$1.23 \times 10^{-3}$	$7.88 \times 10^{-2}$	$1.18 \times 10^{-3}$

This kind of shielding problem was previously solved in ICN Pitesti using ORIGEN-JR code for source calculation, followed by shielding calculations with PELSHIE-3 code /6/.

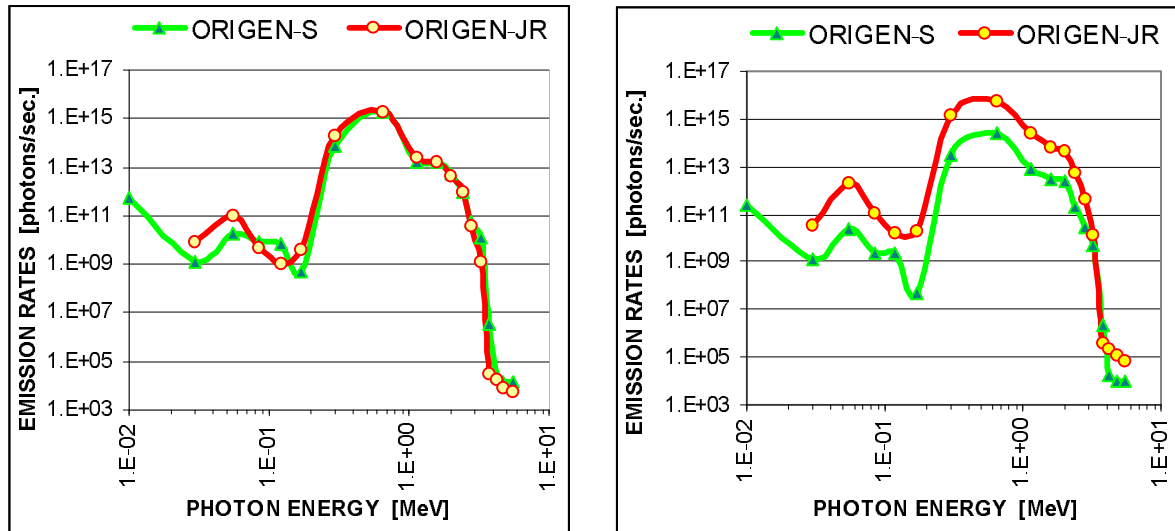
Using Model 1 for fuel representation, a comparison between MORSE-SGC and PELSHIE-3 radial photon dose rates, for 2 values of cask wall thickness, was done and relative differences were calculated (Table 4).

**Table 4 - Relative differences in radial photon dose rates**  
(MORSE-SGC - PELSHIE-3 comparison using Model 1 for fuel representation)

<b>Cooling period</b> [days]	<b>90</b>		<b>360</b>	
<b>Cask wall thickness</b> [cm]	28	30	28	30
<b>Relative difference</b> [%]	17.08	20.5	13.07	10.6

These differences could be explained, mainly, in terms of source imprecision caused by the cross sections library used. The libraries included in SCALE system contain about 1700 nuclides, while ORIGEN-JR code was used along with cross sections libraries containing only 800 nuclides.

Figure 4a, b illustrates gamma sources profiles given by means of ORIGEN-S and ORIGEN-JR codes, for a cooling period of 90 days and 360 days, respectively.



a) 90 days cooling time

b) 360 days cooling time

**Figure 4 – Gamma sources profiles**

A further investigation of source influence on radial photon dose rates has lead to the relative differences presented in Table 5.

**Table 5 – Source influence on dose rates**  
(cooling period: 360 days; cask wall thickness: 30 cm)

<b>Method of calculation</b>	<b>Relative difference [%]</b>
- ORIGEN-S source	
- PELSHIE-3 against MORSE-SGC shielding calculations	2.76
- ORIGEN-JR against ORIGEN-S source	
- MORSE-SGC shielding calculations	20.92
- ORIGEN-JR against ORIGEN-S source	
- PELSHIE-3 shielding calculations	8.02

## CONCLUSIONS

The photon dose rates strongly depend on the cross sections library used in order to calculate the radiation source.

The comparison between PELSHIE-3 and MORSE-SGC results using the same source of radiation has lead to a drastic reduction in the relative difference value.

The radiation dose rates characterizing a cooling period of 90 days sustain the safety point of view: the fuel bundles irradiated in the reactor core must be stored in the reactor pool at least for 6 months, in order to allow the spent fuel decay and reactivity decrease.

It must be noted that the MORSE-SGC model for the cask is more realistic than the one used in PELSHIE-3 calculations, as it includes the most relevant design details of the cask.

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