A MODEL OF THE DOSE RATE CALCULATION FOR A SPENT FUEL STORAGE STRUCTURE BY MONTE CARLO METHOD USING THE MODULATED CODE SYSTEM SCALE 4.4a

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ABSTRACT

The modulated code system SCALE is used to perform a standardized shielding analysis for any facility containing spent fuel: handling devices, transport cask, intermediate and final storage facility. The neutron and gamma sources as well as the dose rates can be obtained using either discrete-ordinates or Monte Carlo methods.

The shielding analysis control modules (SAS1, SAS2H and SAS4) provide a general procedure for cross-section preparation, fuel depletion/decay calculation and general one-dimensional or multi-dimensional shielding analysis.

The module SAS4 used in the analysis presented in this paper, is a three-dimensional Monte Carlo shielding analysis module, which uses an automated biasing procedure specialized for a nuclear fuel transport or storage container.

The Spent Fuel Interim Storage Facility in our country is projected to be a parallelepiped concrete monolithic module, consisting of an external reinforced concrete structure with vertical storage cylinders (pits) arranged in a rectangular array.

A pit is filled with sealed cylindrical baskets of stainless steel arranged in a stack, and with each basket containing spent fuel bundles in vertical position. The pit is closed with a concrete plug.

The cylindrical geometry model is used in the shielding evaluation for a spent fuel storage structure (pit), and only the active parts of the superposed bundles is considered.

The dose rates have been calculated in both the axial and radial directions using SAS4.

1 INTRODUCTION

The analysis of the radiation shielding, performed with SAS4, includes the following steps:
- preparation of a problem-dependent cross-section library;
- depletion of the fuel;
- evaluation of the source-term (neutron and gamma);
- evaluation of the dose rate outside of shield (occupational doses and dose taken of an individual of critic group).
The obtained results will be compared with those evaluated previously with other codes (MICROSHIELD 5 and SAS 2).

2 METHODOLOGIES AND INPUT DATA

2.1 Description of the methodology for SAS4 module

SAS4 calculates radiation doses exterior to a nuclear spent fuel cask with the Monte Carlo method. The method uses the existing functional modules and cross-section and subroutine libraries in the SCALE system to carry out the analysis.

The functional modules executed by SAS4 are BONAMI, NITAWL-II, XSDRNPM and MORSE-SGC. An outline of the SAS4 calculation sequence is shown in Fig. 1. The basic function of each functional module, as applied to SAS4, is described below.

- **BONAMI** applies the Bodnarenko method for resonance self-shielding calculations for nuclides whose cross sections include Bodnarenko data.
- **NITAWL-II** performs the Nordheim resonance self-shielding corrections to cross sections for nuclides whose cross sections include resonance parameters.
- **XSDRNPM** performs two 1-D discrete-ordinates calculations to provide two very different function in SAS4. First, it is used to produce cell-weighted cross sections if the LATTICECELL option is chosen for the fuel lattice type. Then, it is executed in the adjoint mode with slab geometry to calculate adjoint fluxes that are used to generate biasing parameters for the Monte Carlo analysis.
- **MORSE-SGC** performs a Monte Carlo fixed-source analysis that calculates dose rates outside a transport or storage cask. All the standard biasing options, including source biasing, splitting and Russian roulette, path length stretching and collision energy biasing are invoked.

All the required biasing parameters are derived from results of the adjoint XSDRNPM calculation and are automatically input to MORSE.

2.2 Other computer codes

MICROSHIELD is a 3-D code that analyses the shielding and estimates the exposure from gamma radiation, using the point kernel method and Gaussian quadrature numerical integration for 16 source geometries.

The SAS2H control module is a coupled one-dimensional depletion and shielding only on radial direction analysis module. It uses two models for cell-weighted cross-sections: a unit fuel-pin cell and a larger cell (a fuel assembly).

For both programs, the source term was evaluated using ORIGEN-S code.
2.3 Description of the input data

SAS4 module uses a cask model to facilitate implementation of the simplified geometry input and the automated biasing procedure in MORSE-SGC calculations. The main input data for SAS4 basically consist of three parts, material information data, adjoint discrete-ordinates data, and Monte Carlo data:

- neutron-gamma cross-section library: 27n+18COUPLE;
- type of fuel lattice: LATTICECELL;
- number of mixtures: 6;
- type of rod-lattice: SQUAREPITCH;
- center-to center spacing between fuel pins: 1.47683 cm;
- number of assemblies in the basket: 60.

It is important to note that SAS4 module is specialized for a squarepitch lattice of the fuel pins inside a PWR assembly; for this reason, we tried to apply the module, in a forced way, for a CANDU bundle having 37 pins distributed on concentric circles and for cylindrical arrangement of bundles. For an equivalent squarepitch, we obtained:

- number of unit cells on each side of an assembly: 7;
- number of vacant cells in an assembly: 12;
- outside dimension of a fuel assembly: 11.81 cm.

Necessary data for Monte Carlo calculation are strength source information and geometry information. Strength source information refers to gamma and neutron spectra. The radiation source was evaluated both for irradiation of the fuel into reactor core and for post-irradiation time, using ORIGEN-S code with the following characteristics:

- irradiation time of the fuel in reactor core: 247.3 days
- burnup: 7500 MWd/tU
- cooling time of the spent fuel after discharge: 7 years.

Basket geometry was modeled using the concept of the material zone construction using basic combinatorial bodies of the MARS module. In this approach the homogenized fuel assemblies have been modeled as 120 bodies (60 fuel bundles with active fuel part and hardware part), arranged on 4 concentric circles corresponding to 8 input zones.

For the initial Monte Carlo evaluation, a number of 1000 (for gamma) or 100 (for neutrons) source particles per batch and 700 batches (for gamma) and 400 (for neutrons) per running, have been taken into account.

2.3 Description of the pit

A pit is filled with 10 sealed cylindrical baskets of stainless steel arranged in a stack each basket containing 60 spent fuel bundles (of 50 cm total length with active part of 48.03 cm) in vertical position. The pit is closed with a concrete plug. The basket provides containment and confinement of radioactive materials.

Table 1 presents a description of the problem to be solved.
Table 1 Description of the dose rate evaluation for the pit loaded with 10 baskets

<table>
<thead>
<tr>
<th>Description</th>
<th>Radiation source</th>
<th>No. of fuel bundles</th>
<th>Shield</th>
</tr>
</thead>
<tbody>
<tr>
<td>10 baskets completely loaded with spent fuel inside the pit</td>
<td>10 full baskets</td>
<td>600</td>
<td>Reinforced concrete (105 cm radial and 110 cm axial)</td>
</tr>
</tbody>
</table>

The cylindrical geometry model (shown in figure 2) has been used in the shielding evaluation. Dose points are included in the figure.

The 600 bundles have been modeled as 60 longer bundles with length of 480.3 cm considering only the active parts of the pins. The results obtained for this case are presented in Table 2.

Table 2 Dose rate for a pit loaded with 10 baskets (µSv/h)

<table>
<thead>
<tr>
<th>Detector</th>
<th>SAS4 (neutrons) (*)</th>
<th>SAS4 (gamma) (*)</th>
</tr>
</thead>
<tbody>
<tr>
<td>D&lt;sub&gt;1&lt;/sub&gt; (on surface, axial)</td>
<td>D&lt;sub&gt;p&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 7.06 E - 4 (22.5%)</td>
<td>D&lt;sub&gt;p&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 2.55 (25.8%)</td>
</tr>
<tr>
<td></td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 7.52 E - 4 (10%)</td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 2.63 (6.5%)</td>
</tr>
<tr>
<td>D&lt;sub&gt;2&lt;/sub&gt; (at 1 m, axial)</td>
<td>D&lt;sub&gt;p&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 2.35 E - 4 (8.8%)</td>
<td>D&lt;sub&gt;p&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 9.36 E - 1 (5.7%)</td>
</tr>
<tr>
<td></td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 1.23 E - 4 (9.4%)</td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 4.70 E - 1 (4.6%)</td>
</tr>
<tr>
<td>D&lt;sub&gt;3&lt;/sub&gt; (at 2 m, axial)</td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 6.43 E - 5 (9.7%)</td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 3.03 E - 1 (5.3%)</td>
</tr>
<tr>
<td>D&lt;sub&gt;1&lt;/sub&gt; (on surface, radial)</td>
<td>D&lt;sub&gt;p&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 9.15 E - 4 (36%)</td>
<td>D&lt;sub&gt;p&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 6.59 (38.6%)</td>
</tr>
<tr>
<td></td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 1.11 E - 3 (3.5%)</td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 4.56 (1.2%)</td>
</tr>
<tr>
<td>D&lt;sub&gt;2&lt;/sub&gt; (at 1 m, radial)</td>
<td>D&lt;sub&gt;p&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 7.24 E - 4 (11%)</td>
<td>D&lt;sub&gt;p&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 2.18 (4.5%)</td>
</tr>
<tr>
<td></td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;n&lt;/sup&gt; = 4.13 E - 4 (3.5%)</td>
<td>D&lt;sub&gt;S&lt;/sub&gt;&lt;sup&gt;γ&lt;/sup&gt; = 1.90 (1.5%)</td>
</tr>
</tbody>
</table>

*) Indexes n, γ refer to neutron and gamma doses and p, s refer to punctual and surface detectors.
2.4 Description of the storage facility calculation

The Spent Fuel Interim Storage Facility has a unit of storage as a parallelepipedic concrete monolithic module, consisting of an external reinforced concrete structure with 20 storage vertical cylinders (pits) of galvanized carbon steel, inside. The considered dimensions of the module are 21.60 m x 8.10 m x 7.50 m [2].

The cylinders are arranged in a rectangular array, disposed in two rows.

The shielding analysis for the whole concrete module was performed taking into account a single pit, and adding the contributions of first and second order neighbor pits, in a conservative way.

A concrete thickness of side shield of 90 cm was considered for a conservative approach.

3 RESULTS

Table 3 shows a comparison of the dose rates from shielding analysis for Spent Fuel Interim Storage Facility obtained in this work and calculated using different computer codes specialized for shielding assessment: MICROSHIELD and SAS2H module.

All values in the table are considered on the surface of the shield (on side and on top).

<table>
<thead>
<tr>
<th>No.</th>
<th>Description of the case</th>
<th>SAS2H</th>
<th>MICROSHIELD</th>
<th>SAS4</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Pit loaded with 10 baskets, 600 fuel bundles</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>- axial</td>
<td>-</td>
<td>-</td>
<td></td>
</tr>
<tr>
<td></td>
<td>- radial</td>
<td>-</td>
<td>D_γ = 1.39 E +0</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>D_n = 4.96 E - 4</td>
<td>-</td>
<td>D'_n = 7.06 E - 4 (22.5%)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>D_γ = 4.20 E +0</td>
<td>-</td>
<td>D'_γ = 2.55 E +0 (25.8%)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>D'_γ = 2.63 E +0 (6.5%)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>D_p = 7.06 E - 4 (25%)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>D_s = 7.52 E - 4 (10%)</td>
</tr>
<tr>
<td>2</td>
<td>Storage facility completely loaded with fuel bundles</td>
<td>D_n = 4.15 E - 3</td>
<td>-</td>
<td>D_n = 9.15 E - 4 (36%)</td>
</tr>
<tr>
<td></td>
<td>- radial</td>
<td>D_n = 4.67 E +1</td>
<td>D_γ = 2.5 E +1</td>
<td>D_n = 9.08 E - 2</td>
</tr>
<tr>
<td></td>
<td></td>
<td>D_γ = 6.59 E +0</td>
<td>D_γ = 4.56 E +0 (1.2%)</td>
<td></td>
</tr>
</tbody>
</table>

*) Notes:
- Indexes n, γ refer to neutron and gamma doses and p, s refer to punctual and surface detectors (only SAS4 makes distinction between punctual and surface detectors).
- The values missing for SAS2H mean that this code can’t calculate dose rate in axial direction;
- The values missing for MICROSHIELD mean that: a) concerning neutrons, this code can’t calculate dose rate, and b) concerning gammas, the value isn’t available in this moment;
- For the last two values corresponding to SAS4 are not specified because they were obtained by indirect way (based on the results for one pit).
4 CONCLUSIONS

From the results presented in this paper, we may conclude the following:
- the most complete results can be obtained with the SAS4 module;
- the SAS4 results have associated errors, which permit to decide on their confidence (a better result implies larger computing time);
- the results obtained with SAS4 and SAS2H are quite similar, while those obtained with MICROSHIELD are smaller;
- the use of controle module SAS4 of SCALE system to evaluate the radiation shielding in a conservative way, is more appropriate.

REFERENCES


Cell weighting is performed if the LATTICECELL option is input for fuel lattice type

Fig. 1 SAS4 analytic sequence
Fig. 2 Geometrical model of a pit with 10 baskets

All dimensions are in cm

Not to scale

Spent fuel in air
Zircalloy 4
SS304
Reinforced concrete
Void
Carbon steel

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