NTech Lehrstuhl für Nukleartechnik

Development of a statistical sampling method for the propagation of nuclear cross section uncertainties through the SCALE6.1 lattice physics calculations.

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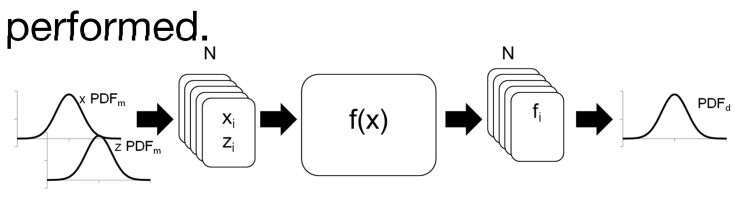
Uncertainty Analysis

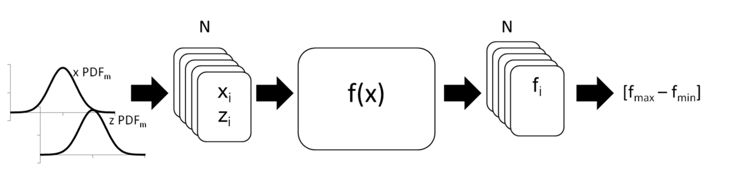
In order to assess how uncertainties in nuclear cross sections affect lattice physics calculations, their distribution functions are randomly sampled with the aim to create *N* sets of input parameters. The *N* calculations are then performed.



An accurate characterization of the output distribution in terms of average and standard deviation would require *N* to be relevant under a statistical point of view. This would dramatically increase the need in term of computational time. An alternative way to evaluate the results of separate calculations is provided by the mathematical model of the "Two-Sided Statistical Tolerance Intervals" elaborated by Wilks. The Wilks' formula defines the minimum number of calculations *N* as a function of probability content (γ) and confidence level (β).

β↓ γ→	One sided statistical tolerance limits			Two-sided statistical tolerance limits		
	0.90	0.95	0.99	0.90	0.95	0.99
0.90	22	45	230	38	77	388
0.95	29	59	299	46	93	473
0.99	44	90	459	64	130	662

Once γ and β have been chosen, a reasonable amount of calculations will define the possible maximum and minimum values for an output random variable.



Calculation scheme

A dedicated program has been written to sample uncertainty data provided in the 44-group SCALE covariance library. The random values obtained varying simultaneously the values of all uncertain cross sections are then applied in perturbing the 238-group ENDF/B-VII library used as input for SCALE.

Since for some cross sections relatively high uncertainties were found ($\sigma/\mu > 0.5$), the sampling from a multidimensional Gaussian distribution of correlated variables gave several negative values, which for cross sections are not acceptable. For those cases a correction was introduced, paying attention not to compromise the randomness of the sample.

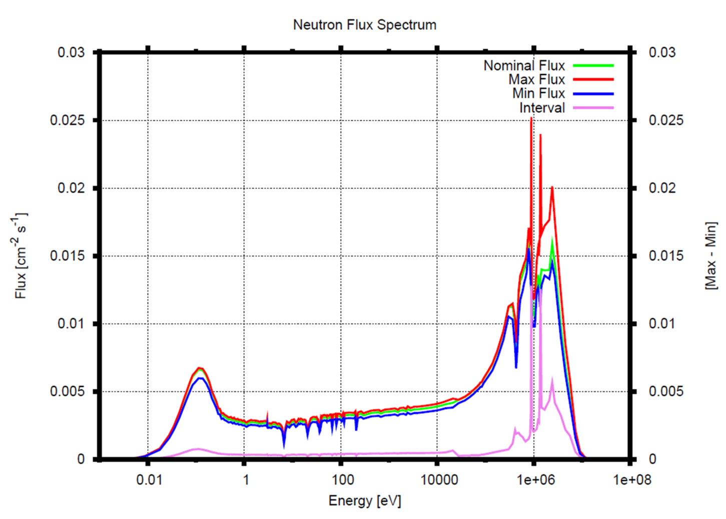
Any arbitrary number of randomly sampled input sets can be calculated; for each of them a lattice calculation will take place. At the end, an output distribution is obtained.

Results for a PWR lattice

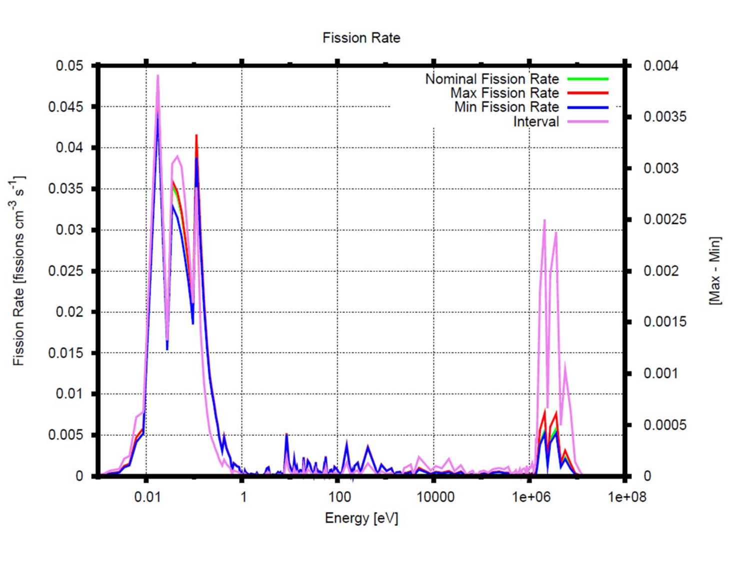
This methodology is here applied to calculate the k effective for the 15x15 PWR lattice, proposed in the exercise I-2 of the UAM Benchmark. A total number of 120 code runs were performed. According to the Wilks' formula at least the 95% of the possible values of the random variable k_{eff} will fall in the interval [kmin, kmax], with a confidence level of the 95%.

Nominal Value	Maximum Value	Minimum Value	Interval
1.423525	1.437192	1.378727	5.8465E-2

One could also be interested in seeing how the neutron flux spectrum, the normalized pinwise power and the fission rate are affected by the input uncertain variables.



Nominal Value Maximum Value Minimum Value [K_{max} – K_{min}]							
0.9667 0.9715 0.9660 5.50E-03	0.9259 0.9322 0.9252 7.00E-03	0.9579 0.9614 0.9572 4.20E-03	0.9928 0.9941 0.9918 2.30E-03	1.0200 1.0202 1.0190 1.20E-03	1.0343 1.0346 1.0326 2.00E-03	1.0250 1.0253 1.0234 1.90E-03	1.0157 1.0160 1.0143 1.70E-03
0.9259 0.9322 0.9252 7.00E-03	0.3203 0.3506 0.3155 3.51E-02	0.8896 0.8946 0.8887 5.90E-03	0.9632 0.9643 0.9617 2.60E-03	1.0033 1.0034 1.0016 1.80E-03	1.0529 1.0532 1.0512 2.00E-03	0.9995 0.9997 0.9978 1.90E-03	0.9795 0.9796 0.9775 2.10E-03
0.9579 0.9614 0.9572 4.20E-03	0.8896 0.8946 0.8887 5.90E-03	0.9490 0.9512 0.9482 3.00E-03	1.0386 1.0396 1.0375 2.10E-03	1.0718 1.0723 1.0697 2.60E-03	0.0000 0.0000 0.0000 0.00E+00	1.0412 1.0415 1.0392 2.30E-03	0.9836 0.9838 0.9819 1.90E-03
0.9928 0.9941 0.9918 2.30E-03	0.9632 0.9643 0.9617 2.60E-03	1.0386 1.0396 1.0375 2.10E-03	0.0000 0.0000 0.0000 0.00E+00	1.0899 1.0905 1.0874 3.10E-03	1.0794 1.0799 1.0767 3.20E-03	1.0030 1.0033 1.0010 2.30E-03	0.9739 0.9742 0.9719 2.30E-03
1.0200 1.0202 1.0190 1.20E-03	1.0033 1.0034 1.0016 1.80E-03	1.0718 1.0723 1.0697 2.60E-03	1.0899 1.0905 1.0874 3.10E-03	1.0530 1.0535 1.0506 2.90E-03	1.0785 1.0791 1.0755 3.60E-03	1.0085 1.0094 1.0063 3.10E-03	0.9811 0.9819 0.9791 2.80E-03
1.0343 1.0346 1.0326 2.00E-03	1.0529 1.0532 1.0512 2.00E-03	0.0000 0.0000 0.0000 0.00E+00	1.0794 1.0799 1.0767 3.20E-03	1.0785 1.0791 1.0755 3.60E-03	0.0000 0.0000 0.0000 0.00E+00	1.0606 1.0620 1.0580 4.00E-03	1.0093 1.0098 1.0068 3.00E-03
1.0250 1.0253 1.0234 1.90E-03	0.9995 0.9997 0.9978 1.90E-03	1.0412 1.0415 1.0392 2.30E-03	1.0030 1.0033 1.0010 2.30E-03	1.0085 1.0094 1.0063 3.10E-03	1.0606 1.0620 1.0580 4.00E-03	1.0321 1.0337 1.0294 4.30E-03	1.0621 1.0635 1.0591 4.40E-03
1.0190 1.0193 1.0176 1.70E-03	0.9795 0.9796 0.9775 2.10E-03	0.9836 0.9838 0.9819 1.90E-03	0.9739 0.9742 0.9719 2.30E-03	0.9813 0.9820 0.9793 2.70E-03	1.0128 1.0133 1.0103 3.00E-03	1.0621 1.0635 1.0591 4.40E-03	0.0000 0.0000 0.0000 0.00E+00



Further developments

The 2-group parameters calculated by the lattice code will be then passed to a core calculation code with the intent of defining conservative limits for the core parameters.

Monte Carlo-Simulation of Photo-neutron Doses from Radiation Treatment with Linear Accelerators

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Introduction

Radiotherapy with Megavoltage linear accelerators (LINACs) is the most common form of radiation therapy to control and treat tumor diseases. In high-energy treatments, photons produced by the Linac are able to induce photonuclear reactions and consequently cause secondary neutron radiation. This undesired neutron contamination typically occurs at beam energies from 10 to 18 MV. In order to assess contribution to non-target organ and total doses, Monte-Carlo-Simulations with a Linac model based on a Varian Clinac machine were performed.

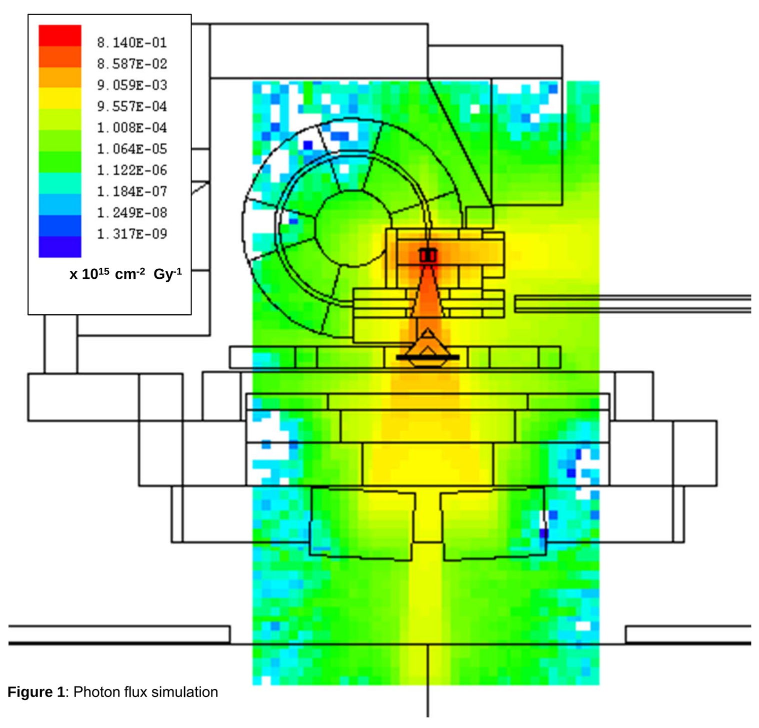

Figure 1: Photon flux simulation

Methods and Materials

For radiation transport calculations the Monte Carlo code MCNPX 2.7e was used. Cross section data libraries included default neutron interaction datasets and the ENDF/B-VII photonuclear data library *endf7u* to simulate (γ,n)-reactions. This ENDF/B sublibrary is essentially based on the IAEA Photonuclear Data Library. As (γ,n)-cross sections are highly dependent on the type of isotopes present in the material, special attention was paid to the isotopic material description of all components of the LINAC. The model setup covers all beam-line components like target, primary collimator, flattening filter, ionization chamber, multi leaf collimator and jaws.

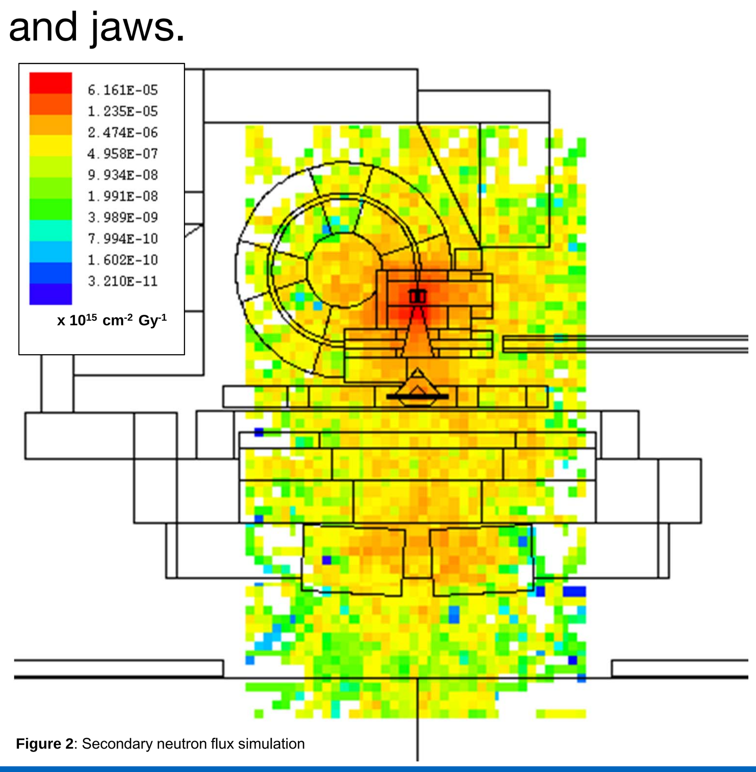

Figure 2: Secondary neutron flux simulation

To accurately predict dose distributions outside the treatment field all other components in the accelerator head were considered as well. The entire geometry can be seen in figures 1 and 2. For an 18 MV initial beam source electrons impinging on a tungsten target were simulated to yield a corresponding Bremsstrahlung-spectrum.

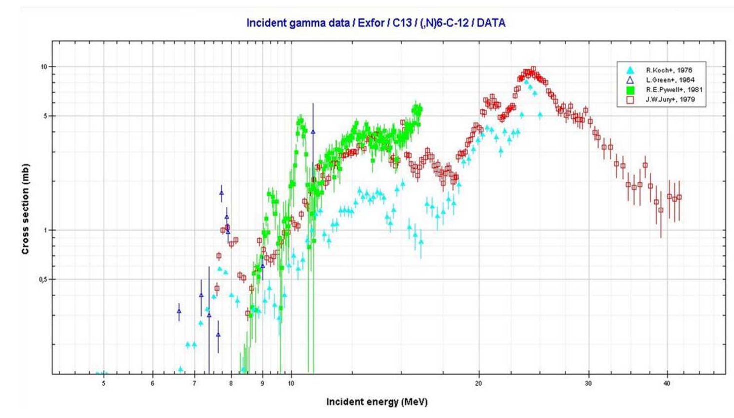

Figure 3: Photonuclear cross section measurements, EXFOR database, IAEA

Results and Discussion

In air neutron flux of 7.46 x 10^6 n cm^{-2} per Gy photon dose (under reference irradiation conditions: at d_{max} from a 10 cm x 10 cm field incident on a water tank) at a distance of 21 cm from the central axis of the beam was calculated. Some Input parameters of the simulations performed, however, are subject to significant uncertainties such as the

- Photonuclear dataset
- Initial electron beam parameters and resulting high energy Bremsstrahlung- spectrum
- Isotopic material composition.

Thus, a comprehensive multidimensional analysis of possible influences on the output is indicated. Figure 3, for example, illustrates a wide spread in the variance of the data available to reproduce photonuclear reaction. Finally, an extension of the model towards a complete treatment scenario including anthropomorphic voxel-based phantoms and detailed resolution of non-target organ doses is intended.

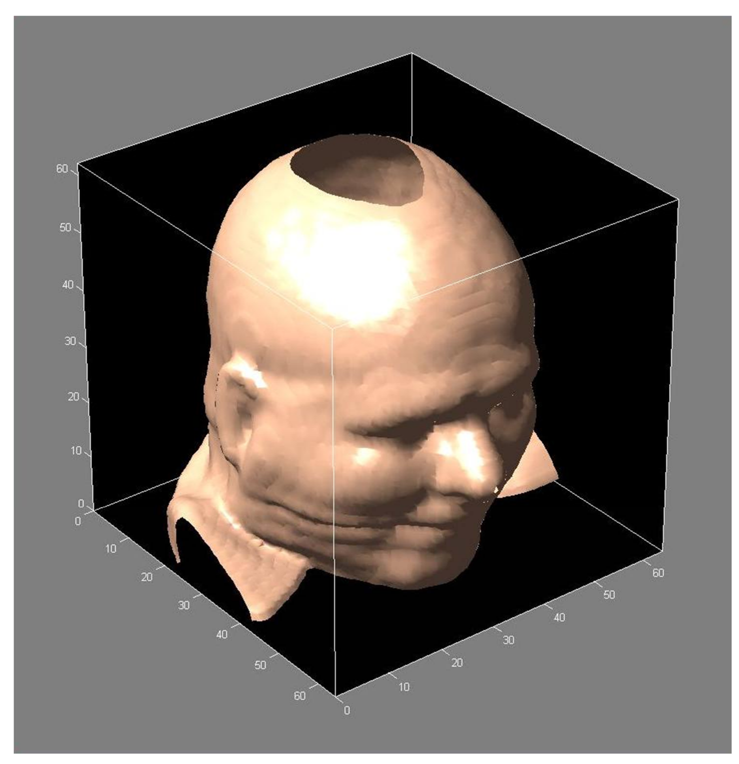

Figure 4: Voxel-based Head Phantom

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