TEACHING COMPUTATIONAL CAPABILITY FOR RADIATION PROTECTION AND SHIELDING IN SCALE

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ABSTRACT

This paper presents the work carried out within a training course in computational capability for Radiation Protection and Shielding, applied to criticality calculation, deep penetration problems, radiation transport, and neutron flux calculation. The course was offered in July 2016 by the Polytechnic University of Valencia (UPV) and was sponsored by The Spanish Nuclear Safety Council (CSN) within the Vicente Serradell Chair of Nuclear Safety to foster the education of highly qualified nuclear safety and radiation protection professionals. It was designed for workers of the nuclear sector as well as for graduate and postgraduate students. The proposed approach was focused on theoretical lectures and practical exercises in order to simulate real problems like nuclear reactors fuel assemblies, spent fuel pools, deep penetration transport simulations and shielding design. To solve the proposed problems, the authors employed two sequences of SCALE code: KENO-VI and MAVRIC, both based on Monte Carlo method for solving criticality and radiation transport problems. The main feature of these SCALE sequences is that they calculate accurately real radiation transport problems in reasonable computation times. This work presents an overview of the course, a brief explanation of the teaching method and the practical exercises.

1. Introduction

This paper presents the work carried out within a training course in computational capability for Radiation Protection and Shielding, applied to criticality calculation, deep penetration problems, radiation transport, and neutron flux calculation. The course was offered in July 2016 by the Polytechnic University of Valencia (UPV) and was sponsored by The Spanish Nuclear Safety Council (CSN) within the Vicente Serradell Chair of Nuclear Safety to foster the education of highly qualified nuclear safety and radiation protection professionals. It was designed for workers of the nuclear sector as well as graduate and postgraduate students. The total number of trainees was 27 and they evaluated positively the course.

Shielding and radiation protection are of concern to many areas, such as: medical facilities, outer space, accelerators, fission and fusion reactors, and nuclear waste management. All these areas involve criticality safety and radiation transport calculation. Criticality safety attempts to prevent nuclear accidents by analysing all possible conditions in fissile material operation, studying the most important parameters affecting the criticality of the system. Radiation transport calculation determines the flux distribution and the dose rate.

Criticality and dose rate calculations are very important for professionals working in shielding and radiation protection areas. Consequently, these workers need theoretical and practical knowledge in methods and codes for performing these calculations.

In a training course of shielding and radiation protection, there are two main objectives. First, to teach the trainees to simulate practical situations as realistic as possible. Second, teach the trainees to calculate fluxes and dose rates with low uncertainties in reasonable times, even for deep penetration problems.

The best method for performing criticality and dose rate calculations in real problems is the Monte Carlo method. One should learn several theoretical concepts for applying this method, like: Theory of probability, Cross Section Libraries, Continuous-energy and Multi-group library, and Material Information Processor. Fortunately, there are several codes for solving radiation transport problems with the Monte Carlo method, which simplify the calculation. Consequently, the trainees should also acquire computational skills to create the input files and analyse the output data.

Among these codes, the authors highlight the SCALE code, in particular the version 6.2, because of two reasons. First, SCALE is an important code for the U.S. Nuclear Regulatory Commission. Second, it includes the state-of-the-art algorithms for criticality safety and radiation shielding.

The teaching methodology applied in the course follows the simulation-based learning methodologies. It combines master classes about theory contents with the application of the theoretical concepts to real cases using computational codes. In lecture classes the teacher creates an environment that propitiates the participation of the trainees. The good understanding of the theory basis is very important to be able to follow the practical classes and to tackle other problems that trainees can find in their professional work.

The outline of this paper is as follows. Section 2 includes a brief description of SCALE. This section includes two subsections describing the major modules for criticality and radiation shielding calculation: KENO-VI and MAVRIC. Section 3 explains the learning method. Section 4 summarises the conclusions.

2. SCALE

The SCALE code system [1] is a widely used modelling and simulation suite for nuclear safety analysis and design, which is developed by the Reactor and Nuclear Systems Division (RNSD) of the Oak Ridge National Laboratory (ORNL) [2]. SCALE provides a comprehensive, verified and validated, user-friendly tool set for criticality safety, reactor physics, radiation shielding, radioactive source term characterization, and sensitivity and uncertainty analysis. SCALE includes nuclear data libraries for continuous energy, multigroup neutronics and coupled neutron-gamma calculations, as well activation, depletion, and decay calculations. Moreover, SCALE includes unique capabilities for automated variance reduction for shielding calculations, as well as sensitivity and uncertainty analysis.

2.1. KENO-VI

The KENO-VI [3] is a 3D multigroup and continuous energy eigenvalue Monte Carlo analysis sequence with criticality search capability. KENO-VI uses the SCALE Generalized Geometry Package, which provides a quadratic based geometry system with much greater flexibility in modelling with slower runtimes. KENO-VI performs eigenvalue calculations for neutron transport to calculate multiplication factors (k_{eff}), fluxes and energy distributions of a criticality problem, useful as a source in shielding problems. On the other hand, this sequence calculates angular fluxes and flux momentums useful to the sensitivity analysis.

The geometry package in KENO-VI is capable of modelling any volume that can be constructed using quadratic equations. Special features include simplified data input, supergrouping of energy-dependent data, and the use of quadratic equations to represent geometry input, a P_n scattering treatment, extended use of differential albedo reflection, and an improved restart capability [4]. Other calculated quantities are neutron lifetime, generation time, energy-dependent leakages, energy- and region-dependent absorptions, fissions, fluxed, and fission densities.

The principal applications of the KENO-VI sequence are listed below.

For criticality calculations:

- Nuclear reactors.
- Spent fuel/refuelling pools
- Spent fuel dry storage casks.
- Research reactors.

With the purpose of calculating the fission source term:

- Depth penetration transport simulations.
- Shielding design.
- Critically accident alarm systems (CAAS).

2.2. MAVRIC

MAVRIC is a 3D sequence for continuous energy and multigroup fixed-source Monte Carlo analysis with automated variance reduction [5, 6]. MAVRIC is based on the Consistent Adjoint Driven Importance Sampling (CADIS) methodology [7], which uses an importance map and a biased source that are derived to work together. Its primary feature is the capability to calculate fluxes and dose rates with low uncertainties in reasonable times, even for deep penetration problems. This sequence automatically performs a coarse mesh 3D discrete ordinates transport calculation using Denovo [8] to determine the adjoint flux as a function of position and energy. This adjoint flux is used for applying variance reduction techniques in the shielding calculation, which is performed by Monaco sequence [9, 10].

The principal applications of the MAVRIC sequence are listed below.

- Perform radiation transport on problems that are too challenging for standard, unbiased Monte Carlo methods.
- Calculate fluxes and dose rates with low uncertainties in reasonable times even for deep penetration problems.
- Shielding calculations in low times.
- Calculation of deep penetration problems in reasonable times.
- Dose rate analysis in a high capacity nuclear spent fuel storage system.
- Dose analysis in pools of spent fuel.
- Gamma ray litho-density logging tools in well-logging studies.
- Radiation transport in ex-core calculations.
- Neutron flux calculation in the pressure vessel.

3. Method

The learning method is based on a theoretical explanation of the calculation method and its application to real problems. As mentioned before, the calculation method is the Monte Carlo Method with variance reduction techniques and is applied to fixed-source radiation transport problems. In case of criticality sources, the source is calculated in the first place by solving an eigenvalue problem.

The major concepts that are explained are the following ones: transport equation, continuous energy and multigroup approach, eigenvalue problem, deterministic method, Monte Carlo method, random number generation, distribution functions and sampling, variance reduction techniques. The instructors highlight the importance of the variance reduction techniques and explained those used in MAVRIC. These techniques are the source biasing and the weight windows; these weight windows are based on the adjoint flux calculated with a deterministic method.

The instructors point out the advantages and drawbacks of the method. In particular, the Monte Carlo method is the most accurate method for solving radiation transport problems in complex geometries, but it might require long run times for obtaining low uncertainties. Thus, variance reduction techniques are applied to the Monte Carlo method to solve accurately the problem in reasonable computation times.

The major applications of this method are shielding calculations and deep penetration problems. Therefore, the theoretical learning was combined with three different practical problems. First, a dose rate analysis in nuclear spent fuel storage system. Second, application of gamma ray litho-density logging tools in well-logging studies. Third, neutron transport in ex-core calculations.

In each of these three problems, the instructors define accurately the problem. Then, instructors and learners make the input for the code together. Next, the cases are run. Finally, instructors and learners discuss the results. The following subsections describe each problem.

3.1. Dose rate analysis in nuclear spent fuel storage system

This example was chosen because it uses several capabilities of MAVRIC code, such as: multiple sources, user-defined distributions for these sources, macro-materials for improved S_N calculations and the automated variance reduction technique.

This model contains PWR spent fuel assemblies, each with specified neutron and photon sources, which are placed inside a shielding cask. Fig 1 shows the geometry of the model. The goal of this example is to calculate the total dose rate within two meters of the cask surface.

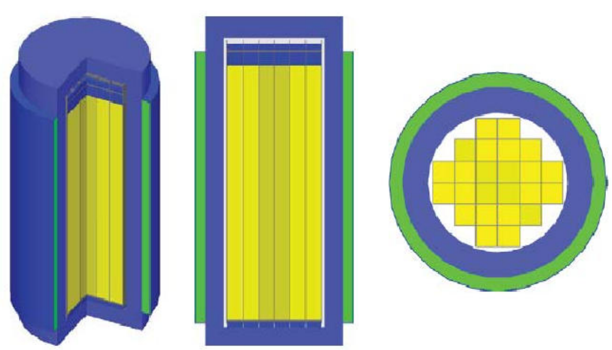

Fig 1. Geometry of the nuclear spent fuel storage system.

The detailed definition of the input includes the following issues: cross sections, materials, geometry, distributions, sources, responses, parameters for the Monte Carlo simulation and parameters for the variance reduction techniques.

Two cases were simulated: one with variance reduction technique and other without it. The comparison of these simulations is used to point out that the variance reduction techniques reduce the uncertainties of the results. The dose rates are also analysed to check the shielding of the cask.

3.2. Gamma ray litho-density logging tools in well-logging studies

The authors chose this example because it is a different application of radiation transport problems. The model of this problem is simpler than the previous one: the source is punctual and with a discrete spectrum and the geometry contains few cylinders as shown in Fig 2.

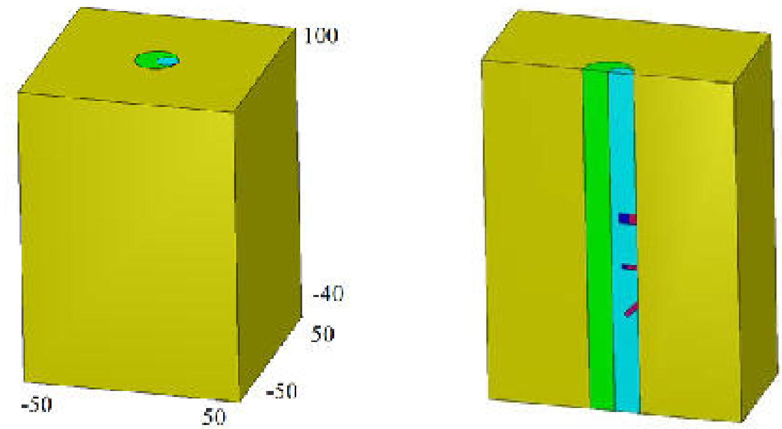

Fig 2. Geometry of the well-logging problem.

As in the first problem, a detailed definition of the input is performed. By contrast, five cases were simulated, which differ in the variance reduction technique. Thus, the instructors and learners discuss the results to find out the optimal technique in terms of computational time and low uncertainty.

3.3. Neutron transport in ex-core calculations

This example consists in calculating the dose in 16 ex-core detectors of a reactor. The example was chosen because of three reasons. First, the complexity of the geometry. Second, one should solve the criticality problem to determine the source, because the neutron source is the fissions from the reactor. Third, the variance reduction technique is crucial to obtain low uncertainties.

A detailed definition of the input is also performed for this case. Figs 3 and 4 show different cross sections of the reactor. As regards the simulations, 19 cases were simulated varying parameters of the variance reduction technique.

Finally, the power of this problem is the analysis of the results. This problem is a fantastic example to show the capability of the variance reduction techniques in both computational time and uncertainties.

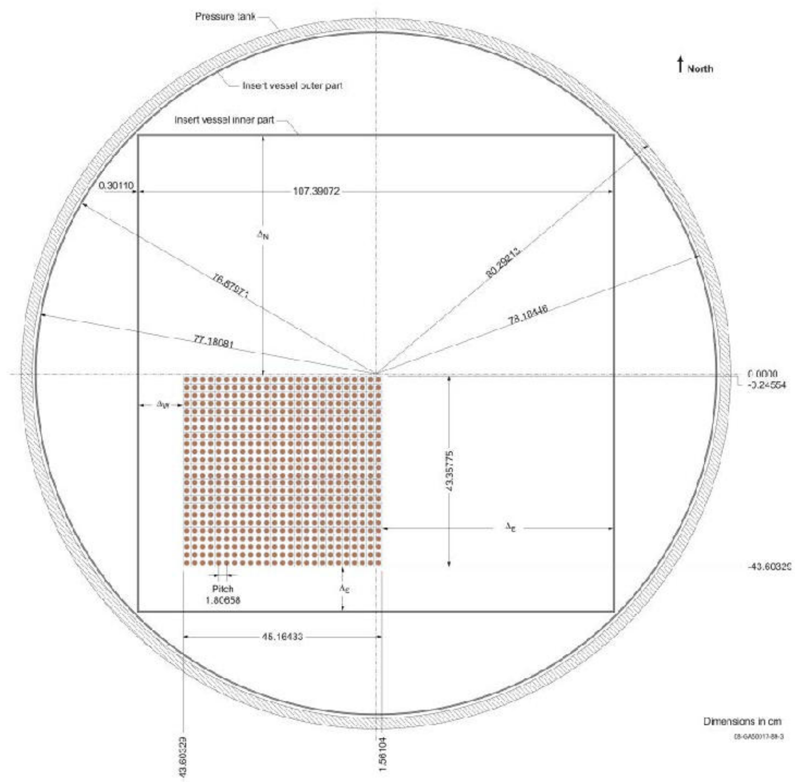

Fig 3. Axial cross section of the reactor.

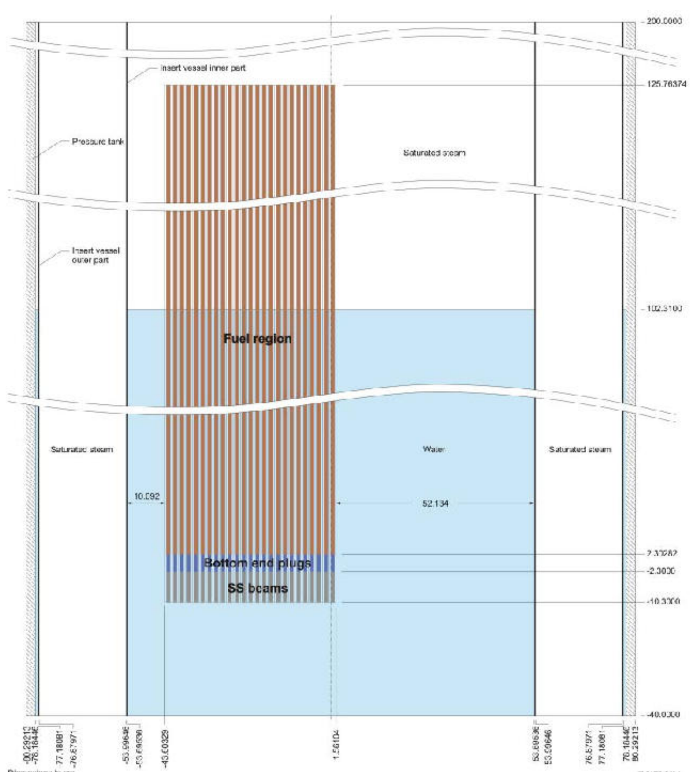

Fig 4. Frontal cross section of the reactor.

4. Conclusions

This work describes the learning method to calculate radiation protection and shielding problem with SCALE code. This method was applied in a training course of criticality and radiation transport calculations, offered by Polytechnic University of Valencia, and sponsored by The Spanish Nuclear Safety Council (CSN) within the Vicente Serradell Chair of Nuclear Safety.

A complete explanation of KENO-VI and MAVRIC modules of SCALE code is provided, especially the input files construction and output files understanding.

The learning method is based on a theoretical explanation of the calculation method and its application to real problems. Since the calculation method is complex (Monte Carlo with variance reduction technique), the application of the theory to real problems is crucial for achieving a good level of understanding.

The instructors applied the calculation method to three real problems. First, a dose rate analysis in nuclear spent fuel storage system. Second, application of gamma ray litho-density logging tools in well-logging studies. Third, neutron transport in ex-core calculations. Each problem includes a detailed definition of the input, several simulation, and analyses of the results. It is important to highlight that the discussion of the results is the important part to test the understanding of the method.

5. References

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