Abstract

An Introduction to MCNP

This part introduces basic notions and concepts to model a radiological scene with the MCNP (Monte-Carlo N-Particle) transport code. The block structure of a typical input file are discussed. The elementary surfaces and geometries enabling the construction of cells are described and applied in simple examples. The other features such as particle characteristic (type, energy, flux,...), evaluated quantities, physics parameters and biasing, will be specifically addressed in applications developed throughout this book. Two case studies will be used to illustrate the different points addressed, as the explanations to create an MCNP input file will be detailed.

Chapter 1: Calculation of Radiometric and Dosimetric Quantities (Fluence, Kerma, Dose and Ambient Dose Equivalent)

1.I Basic Calculations of the Fluence Rate for Sources of Different Geometric Shapes

We propose here to calculate the fluence rate for basic sources: point isotropic source, parallel beam, line source and disk-shaped source. For all of the cases, results of the analytical and numerical solutions will be compared.

Objectives:

- Application of the analytical formalism for fluence calculations.
- Fluence calculation for a point source (analytical and MCNP code).
- Fluence calculation for a parallel beam (analytical and MCNP code).
- Fluence calculation for a line (analytical and MCNP code).
- Fluence calculation for a disk (analytical and MCNP code).

1.II Basic Calculations of Ambient Dose Equivalent

Basic ambient dose equivalent rates for isotropic point sources are calculated for different radionuclides. For each case study, the analytical and numerical approaches are presented.

Objectives:

- Ambient dose equivalent rate calculations for isotropic point-sources.
- Application of the inverse-square law.
- Use of the ICRP "conversion coefficient for the ambient dose equivalent".
- Numerical calculation of dose equivalent.
- Normalization of the MCNP results for multi-source problems.
- Definition of multiple sources in a MCNP file.

1.III Dose and Kerma Calculation for Photons in a Medium Composed of Interfaces

In this problem, dose and kerma curves, as a function of depth, are performed in a set of media.

Objectives:

- Analytical calculation of the kerma and the depth dose for photons.
- Calculation of kerma and dose at the interface between different materials.
- Numerical simulation of kerma and dose.
- Calculation of the Build-up factor in soft tissue.

1.IV Calculation of the Ambient Dose Equivalent Rate in a Uniformly Contaminated Pipe

For this application, the ambient dose equivalent rate within a pipe of which the inner wall is uniformly contaminated by several radionuclides is calculated. The close-form solution of this classic problem of external exposure is discussed below and compared to the numerical result.

Objectives:

- Application of the analytical form for the fluence and ambient dose equivalent.
- Analytical and numerical calculations of the ambient dose equivalent rate for a surface contamination.

1.V Calculation of the In-Depth Absorbed Dose for β Spectra

We propose to calculate the in-depth absorbed dose for β spectra in a water phantom, seen as tissue-equivalent. In a first section, semi-empirical and numerical models are detailed for calculation with a ^{90}Y point-source in contact with the inlet face of the phantom. In a second part are detailed the results of the first section which lead to obtaining the dose equivalent to extremities $H'(0.07)$ for this point-source. In the last section, calculations are achieved for a 50 cm^2 surface contamination for the same radionuclide.

Objectives:

- Semi-empirical calculation of the in-depth absorbed dose for β particles relying on two different models: a fairly simple one that approximates about a part of the maximum range of β particles and a complex accurate one that assesses the actual range.
- Numerical calculation of the absorbed in-depth dose for β spectra.
- Calculation of the absorbed dose with a 3D-mesh in MCNP.
- Semi-empirical and numerical calculations of a dose equivalent to extremities for a β point source.

1.VI Calculation of the Absorbed Dose Profile to Water for 170 MeV Protons

In this section, a calculation of the absorbed dose profile to water for 170 MeV protons is undertaken. The first part draws upon the 1997 work of Bortfield, to suggest an analytical formulation of the absorbed dose as a function of depth and to determine the resulting profile. The second part aims to describe the numerical simulation of the absorbed dose to water and compare the results with those obtained in the first part. The absorbed dose and then the directional dose equivalent under 70 μm at a zero angle of incidence are determined analytically and numerically in the third part.

Objectives:

- Analytical calculation of the absorbed dose to water as a function of the depth for protons.
- Tracking of the different particles set in motion throughout nuclear interactions using MCNP.
- Creation of a mesh for the absorbed dose with MCNP.

- Determination of the mass stopping power at any point of a light ion range by means of the MCNP card “ft let”.
- Representation of a microdosimetric spectrum using the MCNP card “ft let”.
- Calculation of the operational quantity: directional dose equivalent under 70 μm for light ions.

1.VII Neutron Kerma Calculation and Microdosimetric Spectrum Reconstruction with a Tissue Equivalent Proportional Counter

In this problem, the “fluence – first collision kerma” factor for 13.9 MeV neutron is calculated for a low-pressure Tissue Equivalent Proportional Counter (TEPC). The value found is compared to the theoretical value. This factor is then compared to different values of the “fluence – mean organ dose” factor from ICRU (1998a), for the purpose of assessing its ability to estimate these macroscopic dosimetric quantities. The ability to assess the ambient dose equivalent, $H^*(10)$, from the neutron kerma is studied as well. Furthermore, a challenge of this problem is the reconstruction of microdosimetric spectra based on MCNP results. In a last step, the characteristic dosimetric quantities drawn from the MCNP spectra are compared to the reference values to judge the reliability of the proposed method.

Objectives:

- Analytical calculation of the first collision kerma for monoenergetic neutron.
- Study of microdosimetric spectrum and first collision kerma measurement technique with the standard spherical TEPC, *i.e.*, Rossi-type detector.
- Comparison of the first collision kerma with mean organ doses and evaluation of the ambient dose equivalent.
- Microdosimetric spectrum calculation with MCNP simulation using the “anti-coincidence pulse-height” card.
- Neutron mean quality factor calculation from a microdosimetric spectrum.

Chapter 2: Detection Principles, Detector Response to Reference Physical Quantities and Operational Dosimetric Quantities

2.I Gas-Filled Detector Operating in Ionization Chamber Mode for Photons and in Proportional Counter Mode for Neutrons

A gas-filled detector is successively operated in current mode (ionization chamber) for a photon field and in pulse mode (proportional counter) for a neutron field. This application is aimed at evaluating separately dosimetric quantities with both operating modes. This problem highlights the possibility of determining different dosimetric quantities depending on the particle type and the operating mode.

Objectives:

- Establish if the chamber wall is tissue-equivalent.
- Determination of the cavity type according to the cavity theory.
- Calculation of the current in a small cavity.
- Evaluation of an operational quantity from a physical quantity measured by the ionization chamber in a photon field.
- Understand the associated electronic counting system, from the radiation interaction in the detector to the pulse counted and recorded, for a proportional counter exposed to a neutron field.
- Assessment of a protection dosimetric quantity from the fluence.
- Calculation of the neutron kerma in the detector wall.
- Numerical simulation of the energy response of a chamber.

2.II Calibration of an Ionization Chamber with a ^{60}Co Source

In this application, the calibration in terms of ambient dose equivalent, $H^*(10)$, for a radiation survey meter (ionizing chamber type) with a ^{60}Co source is studied. This calibration operation is processed in a secondary laboratory and requires a device for perfect measurement of air kerma. The first part is focused on the evaluation of the air kerma with a reference air-equivalent chamber (Sp01 type) in a metrology primary laboratory. In the second part, the calibration in terms of ambient dose equivalent is performed in a secondary laboratory. The third part is devoted to the study of the survey meter response to ^{241}Am gamma-rays. The main objective of this last part is to check if the deviation between measured and actual values of the operational quantity is within the limit recommended by standards.

Objectives:

- Understand the calibration process of a radiation protection device, from the primary laboratory to the operational use.
- Calculate the theoretical expression of the air kerma with an ionization chamber.
- Determine the energy response of a survey meter with MCNP.
- Assess a wall-scattering factor with MCNP.
- Assess the relevance of the calibration carried out, depending on the energy measured in a real situation.

2.III Calculation of a Dose Rate in a LiF Pellet

In the past few years, Haider *et al.* (1997) have improved Burlin intermediate cavity theory (association of a small cavity and a large cavity) mostly by taking into account the electrons backscattered in the rear face of the chamber. In the first part, the new cavity theory is applied to evaluate the mean absorbed dose in a LiF pellet surrounded by aluminum ("buffer layer"). In this theory, this mean dose can be determined from the absorbed dose at a point under the aluminum inlet face of the detector with an analytical expression. In the second part, the analytical result is verified with MCNP simulations.

Objectives:

- Application of the new cavity theory.
- Calculation of the mean absorbed dose in a sensitive volume from the absorbed dose at a point under the aluminum inlet face of the detector.
- Dose calculation at an interface.
- Correction for photon scattering calculated with MCNP.

2.IV Measurement of the Operational Quantity $H'(0.07, 0°)$ *for a* β *Spectrum with an Extrapolation Chamber*

In a first part, the directional dose equivalent at a zero angle of incidence, $H'(0.07, 0°)$, is evaluated for a reference ^{90}Sr/^{90}Y source, using an extrapolation chamber, model 23392 PTW. This device is recommended by the ISO 6980 standard for the measurement of absorbed dose under 70 μm of tissue, $D_t(0.07)$ for β radiation. For this chamber model, a micrometer screw on a piston enables to adjust the depth of the sensitive volume of the chamber. This feature allows to determine the slope of the ionization current *versus* the depth of the chamber. A check of the slope value with a numerical simulation is performed in a second part.

Objectives:

- Calculation of the dose at the entrance window for an extrapolation chamber.
- Calculation of the theoretical current generated in the sensitive volume using the energy deposition tally.

2.V Fast Monoenergetic Neutron Energy and Fluence Measurements with a Proton Recoil Telescope

This application is aimed at evaluating energy and fluence of monoenergetic neutrons with a proton recoil telescope. These neutrons emerge from a ^{3}H target impinged by a deuteron beam (150 keV) according to the reaction ^{3}H(d,n)^{4}He. This method is highly recommended to carry out a reference measurement of the fluence and the energy of a fast neutron field for an external dosimetry purpose. In a next step, the ambient dose equivalent rate at the point at which the neutron beam interacts with the telescope radiator is inferred. A comparison between results drawn from theoretical and numerical methods is performed.

Objectives:

- Theoretical calculation of the energy of fast neutrons undergoing elastic scattering in an organic radiator and evaluation of the associated uncertainty.
- MCNP simulation of a proton recoil telescope.
- Calculation of a detector efficiency.
- Calculation of the energy resolution.

2.VI Fast Monoenergetic Neutron Time of Flight Energy Measurement

In this problem, the time of flight (ToF) technique is used to measure the energy of monoenergetic neutrons originating from the ^{3}H(p,n)^{3}He reaction. In the first part, an analytical calculation of the ToF is detailed for the experimental data. In the second part, this calculation is done for a *d-d* neutron spectrum simulated with MCNP.

Objectives:

- Theoretical calculation of the time-of-flight (ToF) and matching neutron energy.
- ToF technique with a coincident secondary particle.
- ToF uncertainty and energy resolution calculations.
- Calculation of the mean ToF with MCNP using a fusion spectrum.

2.VII Measurement of a Realistic Neutron Spectrum Using Bonner Spheres

In this application, we propose to determine the neutron fluence- and ambient dose equivalent- rates for three energy regions (thermal – epithermal – fast) by employing three Bonner spheres spectrometer (BSS). In the first part, the response of each Bonner sphere is simulated with MCNP for the three energy regions. In the second part, the neutron fluence is calculated for each energy range through the least squares method. Note that the multi Bonner spheres method is a reference to assess the overall fluence and the spectrum.

Objectives:

- Solving of an equation system through the least squares method.
- Study the principle of an unfolding algorithm to derive a spectrum from the signal provided by several detectors.
- Calculation of the matrix response of a BSS with MCNP.
- Study the principle of Bonner spheres to evaluate a reference neutron spectrum.
- Calculation of a mean "physical quantity-to-dosimetric quantity" conversion coefficient for an energy spectrum.

2.VIII Absorbed Dose Measurement in a Water Calorimeter for a 170 MeV Proton Beam Used in Proton Therapy

This application is aiming at measuring the absorbed dose deposited by a proton beam of 170 MeV maximum energy, for proton therapy. An proton box, corresponding to the Bragg curve plateau region in water, is scanned with an hermetic calorimeter. In the first part, the absorbed dose to water is evaluated from the thermistors temperature change. A subsidiary challenge will be to assess the fluence rate at the water phantom entrance face required for the proton beam to induce such a dose. In the second part, we will evaluate the charge measured by an Exradin T1 ionization chamber located at the reference measurement point for the same proton beam. Finally, in an independent third part, we will point out the difference in the holding time for the temperature variation, owing to the deposited dose, for both a graphite and a water calorimeter exposed to a ^{60}Co radiation field.

Objectives:

- Calculation of the absorbed dose from the temperature rise in a water calorimeter.
- Determination of the proton beam fluence as a function of the absorbed dose in depth.
- Study of the calibration process of an ionization chamber in air for the assessment of the proton dose to water.
- Evaluation of the holding time of the temperature variation in the calorimeter.

2.IX Calibration of a Neutron Survey Meter with a ^{252}Cf Source

This application is aimed at describing the calibration of a neutron survey meter. In the first section, the device is calibrated in terms of μSv.h^{-1}/c.s^{-1} with a ^{252}Cf source using the shadow cone method described in the ISO standard 8529-2 (ISO, 2000). The device is a spherical ^{3}He-filled proportional counter surrounded by a polyethylene layer, identical to the 8″ Bonner sphere studied in the application 2.VII.

In the second section, the survey meter is used to measure a typical workplace neutron field, emitted by a PuF_4 solution considered as a point-like source. At the end, the reliability of the ^{252}Cf calibration is checked with respect to the deviation between the readout of the calibrated survey meter and the actual dose equivalent rate of the PuF_4 solution.

Objectives:

- Application of the shadow cone method described in the ISO 8529-2 standard.
- Calculation of the efficiency and energy response of a Bonner sphere using MCNP.
- Evaluation of a calibration coefficient for a proportional counter.
- Check of the calibration reliability depending on the characteristics of the workplace radiation field.

Chapter 3: Shielding and Activation Calculations for Several Types of Facilities or Devices (Radioactive Sources, X-ray Generators, Accelerators...)

3.I Shielding Calculation for an AmBe Neutron Source

In this application, the thickness of a polyethylene shielding is determined to ensure a dose equivalent of 2 μSv/h at 1 m of a 60 GBq Am-Be source. The calculated thickness derives from a semi-empirical model, checked by a numerical simulation.

Objectives:

- Semi-empirical and numerical calculations of a shielding for a neutron source.
- Use of neutron transmission coefficient.
- Neutron spectrum implementation in MCNP.
- Numerical calculation of dose equivalents for a mixed particle field composed of neutrons and photons.

3.II Calculation of a ^{60}Co Photon Source Ambient Dose Equivalent Rate Behind a 4 m Water Shielding

In this problem, the ambient dose equivalent rate of a ^{60}Co photon source of 20 TBq is assessed behind a 4 m water shielding. The distance between the source and the calculation point is 4.2 m.

Objectives:

- Semi-empirical calculation of a Shielding thickness for a ^{60}Co source.
- Use of the build-up factor for photons.
- Numerical calculation of a very thick shielding.
- Use of a variance reduction technique for numerical calculations.

3.III Radiological Shielding for an X-ray Generator

The design of a radiological shielding for an X-ray generator is discussed in this problem. In the first part, a semi-empirical method is applied, and results are compared to a numerical approach in the second part.

Objectives:

- Semi-empirical calculation of a radiological shielding for an X-ray generator.
- Use of the transmission factor for an X-ray spectrum.
- Numerical simulation of an X-ray spectrum as a primary photon source.
- Application of a biasing technique to shielding calculations.

3.IV Evaluation of Ambient Dose Equivalent from Neutron Activation

The irradiation of a ^{59}Co sample with a monoenergetic neutron beam, resulting in the formation of ^{60}Co by activation is studied in this application. In a first part, the ^{60}Co activity is calculated analytically, as well as the resulting fluence and ambient dose equivalent rate at 1 m from the sample. In a second part, the same quantities are numerically evaluated and compared to that of the first part. The third part points out the possibility to assess fluence and dose equivalent rates for different irradiation scenarios and cooling steps, and therefore to perform the time profile of the external exposure.

Objectives:

- Analytical calculation of sample activation by neutrons, resulting fluence and ambient dose equivalent rates.
- Same as above with MCNP.
- MCNP calculations with the "act" card.
- Setting of an irradiation time profile with MCNP.
- Calculation of a prompt and delayed photon spectrum with MCNP during a continuous irradiation.
- Achievement of the time profile of the fluence rate for successive irradiation and cooling steps.

3.V Calculation of the Scattered Ambient Equivalent Dose Rate for an X-ray Generator Beam on a Water Phantom

In this application, the ambient dose equivalent rate resulting from the scattering of an X-ray generator beam on a water phantom is evaluated.

Objectives:

- The ambient equivalent dose rate for an X-ray scattering field.
- MCNP and semi-empirical models of the scattered radiation and its related ambient dose equivalent.
- Variance reduction techniques with use of the pseudo-particle and of the DXTRAN sphere.
- Single-step calculation of dosimetric quantities related to the bremsstrahlung radiation with MCNP.

3.VI Calculation of Radiological Shielding for a 20 MeV Electron Accelerator with an X-ray Conversion Target

This application presents recommendations and technical information related to the design of radiological shielding for a 20 MeV electron accelerator. The electron beam is fully stopped in a target of high-Z material (tungsten). Related issues of radiological exposure as air activation, skyshine and ozone production are also addressed.

Objectives:

- Complete study of source-terms for the irradiation facility (analytical and MCNP).
- Calculation of the ambient equivalent dose rate due to bremsstrahlung produced by the interaction of electrons in tungsten.
- Assessment of the photoneutron energy.
- Calculation of the ambient equivalent dose rate due to neutrons produced by photonuclear reactions.
- Calculation of the shielding around an electron accelerator with high-Z target.
- Calculation of the external exposure component due to skyshine.
- Numerical calculation of the saturation activity in the air accelerator room.
- Air activity evolution as a function of time with an air change rate.
- Calculation of ozone concentration arising from photon interaction with air molecules.