Practical Monte Carlo using MCNP - Assessed Exercise

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The MCNP assignment is set to PTNR, 4^{th} year Nuclear Engineering and NDAWM students. The problem set in Exercise 1 is the basis for some of the geometry used in the second part of the assessment (Exercise 2 or 3). Use the MCNP manual and primer documents to aid you before asking for help.

You need to complete Exercise 1, and then choose ONE of either Exercise 2 or 3.

The assessment will take the form of a single short report with a **maximum length of five pages of text**. This should be supplemented by up to three pages of results tables and figures in an appendix at the end of the report. The report will include both exercises and must take the following format:

1. **Abstract**: A brief description of the work and the key results and conclusions (200 words maximum)
2. **Introduction**: Explain in outline what your modelling is trying to achieve and what you are setting out to calculate. Provide some relevant background information.
3. **Method**:

   3.1 Exercise 1: Describe the MCNP model you use for Exercise 1, including geometry, materials, source, tallies and anything else of relevance.

   3.2 Exercise 2 or 3: In a similar way describe the model you use for Exercise 2/3

   This description can include how you tackled the problem and, where choices are required, what choices you made and why.
4. **Results**:

   4.1 Exercise 1: Present the results obtained for Exercise 1, referring to tables and figures in the appendix.

   4.2 Exercise 2 or 3: Present the results for Exercise 2 or 3, referring to tables and figures in the appendix.

   For both exercises choose carefully which results to present and how to present them, bearing in mind the assessment questions.
5. **Analysis & Discussion**:

   5.1 Exercise 1: Use appropriate analysis and discussion to answer the assessment questions.

   5.2 Exercise 2 or 3: Use appropriate analysis and discussion to answer the assessment questions.

   If there are other relevant points to make about the results and analysis, feel free to included them in your discussion, subject to the total page limit.
6. **Conclusions:** Summarise what the outcome of your investigation was, paying attention to the set questions.

Appendix: This will contain your results tables and figures.

Target Readership: Write the report as if it is to be read by somebody who is already familiar with Monte Carlo radiation transport methods.

Although no marking scheme is provided, the allocated marks are heavily weighted towards your results, how you present them and the conclusions you draw from them. We will not assess any textual material within this document beyond the first five pages, but we will look in detail at your figures and tables.

You will also need to submit a single MCNP input file for each of Exercise 1 and your choice of either Exercise 2 or Exercise 3.

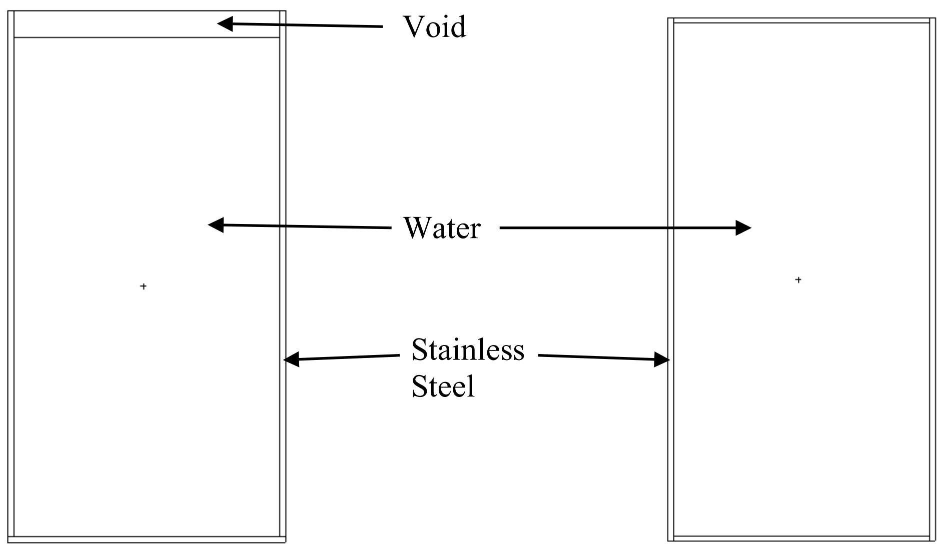
You will be expected to justify your choice of calculation method and any choice of calculation parameters and analysis techniques.

Your report is due for submission electronically using the Turnitin system by no later than 4pm on Monday the **25th March**. You must add your name to the top of your report and your MCNP input files (using a comment; you will be assessed on your use of comments).

Exercise 1 Simple Neutron Source in a Bucket of Water

In this exercise you will define simple MCNP surface and cell cards from a given geometry description. You will use the source definition (SDEF) card and simple tallies.

Geometry: You are required to model the geometry for a rectangular bucket filled with pure water. The bucket has internal base dimensions of 10 cm x 20 cm and a wall and base thickness of 2 mm and an external height of 20 cm. The water is filled to 10 mm below the top of the bucket, but all other space can be considered as a void (don't worry about the air). The elevation and plan views below show what it looks like.



Sectional Elevation *Sectional Plan View*

Define this geometry in an MCNP input file. Choose a sensible place for the origin of your model. Remember MCNP requires all space to be defined and you need to define the outer boundary of your geometry.

The bucket is constructed from stainless steel with a density of 7.92 g/cm^3 and a chemical composition defined as:

Element	Weight %	Atomic Number
Iron	74 wt%	Z = 26
Chromium	18 wt%	Z = 24
Nickel	8 wt%	Z = 28

Source: The source in this problem is an isotropic neutron source with an energy distribution defined by a Watt fission spectrum from thermal neutron-induced fission of ^{235}U. This is located within the water on the centreline of the bucket, 20 mm up from the bottom.

For the source energy spectrum make use of the appropriate pre-defined probability distribution within MCNP (Watt fission spectrum). This takes the form:

$$P(E) = C \exp^{(-E/a)} \sinh(bE)^{½}$$

By default in MCNP a = 0.965 MeV and b = 2.29 MeV^{-1}. You will need to select the appropriate parameters for the spectrum required, referring to Appendix H of the manual.

Tallies: Use two surface tallies to calculate the neutron fluences averaged over the external vertical sides of your bucket. One tally for the average fluence over the two short sides and one for the two long sides. Add another surface flux tally for the base of the bucket. The energy bins for these tallies should be equally spaced on a logarithmic scale from 10^{-9} MeV to 10 MeV. For example, use simple decade steps over this range, or a finer spacing (e.g. 5 bins per decade). The finer the bins the more refined your spectrum will be but the statistical uncertainties will be larger (it is always a compromise).

Run this simulation for 20,000 neutron source histories.

Questions:

1. Are these results reliable? Explain your answer.
2. What are the total (all energy) neutron fluences at the external long and short vertical surfaces and at the base?
3. Compare the neutron spectra and the fluences for the tallies over these sides and the base of the bucket. Explain any observed differences.

Run this simulation for a more suitable number of particle histories (you choose) to obtain successfully converged results. Discuss these results in relation to the three questions posed above.

If your source represents 10^{10} neutrons per second what are the calculated all-energy neutron fluxes in n/cm^2/s ?

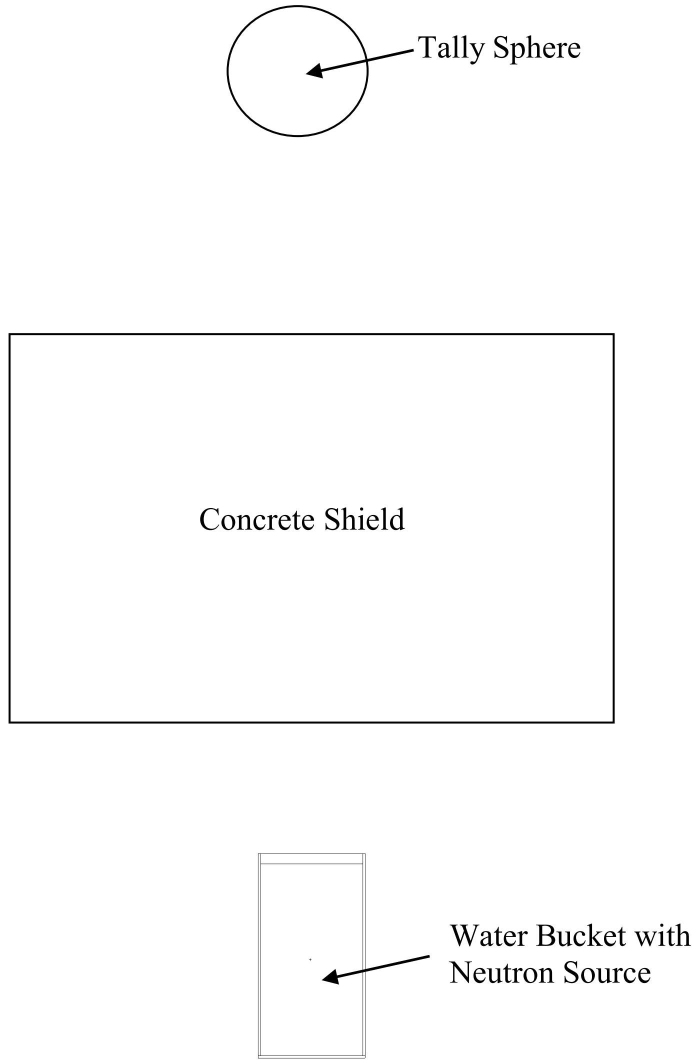
Submit this final input file.

Now **choose** either Exercise 2 or Exercise 3 for the remainder of the assessment.

Exercise 2: Deep penetration neutron shielding problem

This exercise is a deep penetration shielding problem and we are interested in the neutron flux and dose rate on the far side of a thick concrete shield.

Geometry: Modify the geometry described in Exercise 1 by adding a 60 cm thick concrete shield above it. The lower surface of this shield should be 20 cm from the top of the bucket (not the water).



Elevation View of the Bucket, Concrete Shield and Tally Sphere (not to scale).

Carefully consider the lateral dimensions of the concrete given what you know about direct and scattered contributions to fluxes on the far side of a shield.

The density of concrete is 2.3 g/cm^3, and its chemical composition can be modelled approximately as:

Element	Atomic number	Weight %
Oxygen	8	53
Silicon	14	34
Calcium	20	10
Hydrogen	1	3

Source: An isotropic neutron source with the same energy distribution as for Exercise 1. This time it is now uniformly distributed throughout the volume of the water in the bucket. The total neutron emission rate within the bucket is 10^{14} neutrons per second.

Tallies: Define two tallies for this problem.

a) A track length neutron flux tally, averaged over the volume of a 20 cm diameter sphere centred on the centre line of the bucket and 50 cm above the top face of the concrete shield.

b) A point detector tally located at the centre of this tally sphere.

Since this is a shielding assessment problem you need to calculate neutron dose rates in mSv/hour on the far side of the shield. To do this you need to modify your tallies using the Dose Energy (*DE*) and Dose Function (*DF*) cards. Use the ICRP-21 neutron fluence-to-dose conversion factors listed in Appendix H (Table H.1) of the manual to define the appropriate response function.

Without User-Defined Variance Reduction (VR)

Run the simulation for 100,000 histories with no user-defined VR.

Question:

1. Is 100,000 histories enough? Comment on your answer.

With User-Defined Variance Reduction

Use the technique of geometry splitting/Russian roulette and allocate suitable cell importances to improve your simulation by increasing the number of neutrons passing through the shield. This is likely to be an iterative process. Aim to achieve acceptably converged answers.

Taking the source strength into account report the results for both of your tallies in mSv/hour in a suitable format.

Questions:

2. Are the tallies reliably converged? Explain your answer.
3. How do they compare with each other?

Submit this final input file (the one with suitable splitting and cell importances).

Exercise 3: Criticality

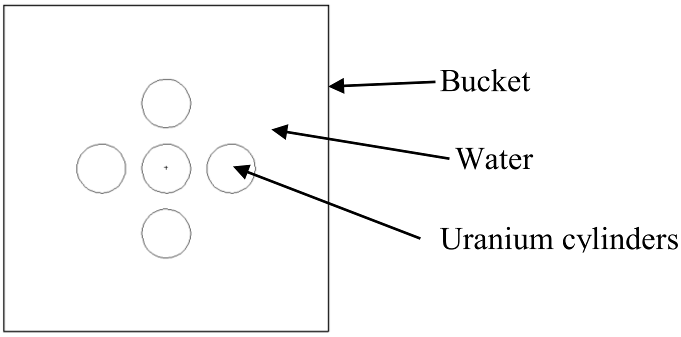
In this exercise you will use MCNP in criticality mode (*KCODE*) to investigate how neutron multiplying systems can be modelled in MCNP. You will model a variation of the simple geometry defined in Exercise 1.

Geometry:

Change the internal dimensions of the bucket to make its internal base 100 cm x 100 cm and its overall height 60 cm. Keep the water level to be 10 mm from the top of the bucket. The bucket now sits on a thick concrete floor. Use the concrete material composition and density defined for Exercise 2.

Reflection from the floor (and other components) is often an important part of a valid criticality assessment. Define suitable dimensions for this concrete slab and justify your choice.

Within the bucket define five 15 cm diameter cylinders, each 25 cm tall and made from 20wt% enriched uranium, (80wt% ^{238}U, 20wt% ^{235}U) with a density of 19.2 g/cm^3. These five cylinders stand vertically on the base of the bucket. Four of them are located equidistant from the bucket centre line with their axes 20 cm from it so that they form a square arrangement, with the sides of the square at 45° angles to the sides of the bucket, as shown in the figure below. The fifth cylinder sits vertically on the centre line of the bucket.



Sectional Plan View

KCODE:

You must make appropriate choices for the number of starting neutrons per cycle (*nsrck*), the initial estimate of the criticality constant (k_{eff} which we denote *rkk*), the number of cycles to skip before a steady state is reached (*ikz* - you won't know this until you've run the simulation), and the total number of cycles to run (*kct* - think about this one). The argument *nsrck* needs to be large enough so that there are sufficient new fission events generated within each cycle. Be careful to choose the other parameters to avoid bias in your final result. Expect to have to rerun this simulation several times until you are satisfied with your choices.

Questions:

1. Examine and report upon the estimate of k_{eff} with cycle number given in the output.
2. Are you confident in the final reported result and its uncertainty? Justify your answer.
3. In addition to Monte Carlo stochastic uncertainties what other uncertainties may need to be considered in a criticality safety assessment?

Increase the uranium enrichment to 25wt% ^{235}U. Run the simulation again to convergence.

Then add some boron-10 to your water as a neutron poison and run the simulation again to convergence. What happens to k_{eff} when the enrichment is increased and when boron is introduced. Explain your observations in terms of the nuclear processes involved.

Try several concentrations of boron-10 to determine what is the minimum concentration required (in parts per million, along with its uncertainty) to lower k_{eff} to less than 0.8 with these five cylinders of 25wt% enriched uranium.

Submit one of these final input files (25wt% ^{235}U with ^{10}B in the water).