GETTING STARTED IN MCNP

This is a short introduction to help you get started in using MCNP in the computer lab. It is designed to be an ‘at a glance’ guide to the input file, which is your first interface with MCNP. The notes are very basic explanations of the elements of the input card, which should give you enough information to start using MCNP and attempt the practice examples.

References are given to sections of the MCNP 4C manual where you can obtain more information as you progress onto more complex problems. This is by no means comprehensive, but is enough to get you moving ahead. A useful section of the manual to read before moving on is Chapter 1 section IV.

Also given is a section on common errors and tips to avoid errors – this will help you to troubleshoot the practice examples.

The sample input file to accompany this guide is given at the end of this handout.

CELL CARDS

Density of material in that cell. The minus sign means that the units are in g/cm^3

Cell number

2 1 –1.0 1 –2 3 –4 5 –6

Material number of material in that cell. M1 corresponds to material card M1, M2 to material card M2 etc

The surface numbers that are used to make the cell. The default is + i.e. if you don’t put a sign, MCNP assumes it is +

Chapter 1 section III A (pg 1-15), section IV B (pg 1-24), and chapter 3 section II A (pg 3-10). Units are summarised at the start of chapter 1 section IV, pg1-21.

SURFACE CARDS

A plane crossing the X-axis at the point (-7.5, 0.0, 0.0)

Surface number

A plane crossing the Y-axis at point (0.0, 7.5, 0.0)

1 PX -7.5
2 PY 7.5
3 SY -2.0 1.0

A sphere with centre on the Y-axis at the point (0.0, -2.0, 0.0) with radius 1.0

Chapter 1 section III B,C (pg 1-20) section IV C (pg 1-25) and chapter 3 section III (pg 3-12). Table 3.1 on page 3-14 has a useful summary of all surface card syntaxes. You may find it helpful to print this off and keep a copy to refer to.

MODE CARD: ***MODE***

This describes the type of particle you want the simulation to transport. The mode can be P (photons), N (neutrons) or E (electrons) or any combination of these three.

Chapter 1 section IV D 1(pg 1-27) and chapter 3 section IV A (pg 3-23)

CELL PHOTON IMPORTANCES: ***IMP***

In this case it is for photons, but it will apply to whatever particles you want to transport. The cell particle importance tells MCNP what weight to give to the particles in a particular cell. The first importance value in the list applies to the first cell, the second applies to the second cell and so on. In this example, cells 1, 2 and 3 have importance 1, and cell 4 has importance 0. An importance of 0 is given to cells that are beyond the volume of interest, usually in the external void.

They can also be used to improve the efficiency of your simulation. For example, if you are only interested in photons moving downwards from a source, you can give a photon importance of 0 to a cell that describes the space above the source. MCNP will not then waste time calculating the transport of photons in a part of the geometry that you are not interested in.

It is important to check that there are the same number of particle importances as there are cells, otherwise MCNP will return an error message.

Chapter 3 section IV C 1. (pg3-33)

SOURCE DEFINITION: ***SDEF***

Specifies all information about the source, i.e. position, energy distribution, direction, type of particle etc. In this example, the position of the source is (0.0, 0.0, 0.0) and its energy is 1.0 MeV. Where a parameter is not specified, MCNP uses default values. The default distribution is isotropic, and the default particle is particle number 1 which is a photon. Source definitions can be quite complex, but to begin with, use a general source definition.

Chapter 3 section IV D (pg3-49). Chapter 1 section II B.(pg 1-28) gives default values.

TALLY CARDS

Tally number. You can have as many tallies as you like

Tally type number.

F12: P 1

Number of the cell or surface over which you want the tally to be counted.

Type of particle to be tallied. Here it is P for photons.

There are numerous tallies. The one shown is a type 2 tally, which measures surface flux.

Chapter 1 section IV D 4, (pg 1-29) gives a summary of tallies and further page references for more information.

ENERGY BINS FOR TALLIES

You can specify in which energy ranges you want your results to be expressed. You may be trying to develop a spectrum, or you may only be interested in all particles regardless of energy. Remember that the more bins you have, the fewer particles may be in each bin and the longer you will have to run the simulation to achieve acceptable statistics in each bin. The identifier number tells MCNP which tallies the energy bin applies to. In this case it is E0, which means that this energy bin distribution will apply to all tallies.

Chapter 1 section VI D 4 a+b,(pg 1-30).

MATERIAL CARDS

Specifies the fraction of the material constituted by that isotope.

Material number

M1 1000 0.6667 8000 0.3333

Specifies the atomic number and mass number of the isotope which constitutes the material

In this example, the material is water, which is made of 0.6667 parts hydrogen, to 0.3333 parts oxygen. Hydrogen is represented as 1000. The 1 refers to an atomic number of 1 and the remaining 3 digits relate to the isotope. In this case, however, because the example uses photon transport only, the isotope number does not have to be given. If no numbers are specified for the mass number, MCNP will assume the natural abundance of that element.

If you want to deviate from the natural abundance, you must specify the isotope number. For example, if your material was heavy water, you would want your hydrogen isotope to be deuterium, so you would write it as 1002. It is usually more important to specify isotopes for neutron problems, as cross sections for interactions are different for different isotopes of the same element.

You can also specify a particular cross section library by adding an extension to the first number.

Chapter 1 section VI D 5,(pg 1-30), chapter 3 section IV F (pg3-107).

PROBLEM CUTOFF CARD

This specifies how you want the run to be terminated e.g. by specifying a maximum number of source particles or a time limit. For acceptable statistics, a large number of histories should be run, but remember that this will increase the time it takes to complete. For error checking at the beginning, use a small number for speed. A rule of thumb is that to half the relative error in the mean of your results, you must quadruple the time.

NPS means ‘number of particles’. **CTME** means computer time cut off (in minutes).

Chapter 3 section IV H (pg 3-123)

GENERAL TIPS AND COMMON ERRORS

- **Geometry** - always check your geometry using the geometry plotter to check for errors before running a simulation.
- **Columns –** the input file is column sensitive. Make sure you haven't deleted or inserted a space. Its best to write a new file by modifying an existing one rather than starting from scratch.
- **Blank line delimiters** -there should be one between the cell cards and surface cards, one between the surface cards and the data cards and no more.
- **Carriage return at end** - after typing the last line of the input file, you must press enter.
- **Tabs** – DON'T USE TABS!
- **Cell importances** – make sure there are the same number of cell importances as cells.
- **Source position** – don't position a source on one of your geometrical surfaces. MCNP doesn't like this and will give fatal error messages in the run even though you won't get an error in your geometry. If your real life situation needs a source on a surface, position it a fraction above or below the surface in MCNP.
- **Account for all regions in your geometry** – a common error is to insert a cell to represent a solid object without changing the air spaces around the object. You will then have one region that is assigned as part of two cells. Similarly you could have a region that is not accounted for in any of your cells.
- **Union or intersection?** – be clear about whether you should be using a union or an intersection operator. It often helps to sketch a diagram.
- **External void –** ensure your external void sphere is large enough to encompass your geometry.
- **Materials** – make sure there are the same number of material data cards as materials assigned to cells.
- **Cells and surfaces –** where you have to give a cell or surface number,e.g. in a tally, don't put a surface number instead of a cell number and vice versa.
- **Comments –** use sensible comments to label your input file. This not only makes it easier for you to monitor what your file is doing, but it helps anyone else looking at it trying to troubleshoot.