Notes for PS#6: MCNP Criticality Calculation

1. Running Criticality problems on MCNP

MCNP can be used to calculate the neutron multiplication constant (eigenvalue) of the reactor, k_{eff}. k_{eff} is the ratio between the number of neutrons in successive generations, with the fission process regarded as the birth event that separates generations of neutrons. To run a criticality problem, in addition to the geometry description and material cards, all that is required is a **KCODE** card and an initial spatial distribution of fission points using either the **KSRC** card, the **SDEF** card, or an **SRCTR** file.

The **KCODE** card specifies the MCNP criticality source that is used for determining k_{eff}. A description of the card can be found in the MCNP manual at Page 3-70 (364). The **KSRC** card described on the next page specify the location(s) of initial source points for a KCODE calculation. An example KCODE card looks like:

```
KCODE   3000  1  5  35
KSRC    0 0 0
```

This **KCODE** card indicates this is a criticality calculation with a nominal source size of **3000** particles, an estimate of k_{eff} of **1.0**, skip **5** cycles before averaging k_{eff} or accumulating the tallies, and run a total of **35** cycles. The initial source location is **(0, 0, 0)**, specified by the **KSRC** card.

This means MCNP will sample and track 3000 starting neutrons for each cycle (or each generation of neutrons). The initial fission sites will be (0, 0, 0), the energy of each particle will be sampled from a Watt fission spectrum. The fission sites generated by each cycle will be used as the neutron starting sites for next cycle. The first 5 cycles (inactive cycles) will be skipped so that the fission sites can reach an equilibrium distribution in the fuel. From the sixth cycle, MCNP starts to average the calculated k_{eff} from each cycle and accumulate the tallies until the total number of cycles specified, 35, is reached (active cycles). A SRCTR file will be generated to store the fission sites of each cycle, with can also be used as the initial spatial distribution of fission sites for other similar problems.

To calculate k_∞, you can define the boundary of your fuel as a reflector so that every neutron trying to escape will be bounced back into the fuel. This is done by adding a '*' in front of your surface card, for example:

```
*1      SO 50
```

2. Using MCNP to estimate various contributions to k_∞

It is possible to use MCNP to separate the contributions to k_∞ into different contributions: η, f, p and ε. To do this, you need to define the probabilities needed to calculate these 4 factors in terms of MCNP tallies. First, you should divide all your tallies into two energy bins: thermal and fast. The energy threshold for thermal region should be around 1 eV. Then, use the track-length flux tally (F4) combined with the FMn (tally multiplier) card to tally various reaction rates.

A description of the FMn card can be found in the MCNP manual at Page 3-87 to 3-91 (381 ~ 385). The FMn card is used to calculate any quantity of the form:

$$C\int\varphi(E)R_m(E)dE\,,$$

where $\varphi(E)$ is the energy-dependent fluence, C is a designated constant, and R(E) is an operator of additive and/or multiplicative response functions. If R(E) is specified as the cross sections of certain reaction, the tallied result will be an indication of the total number of this type of reaction happened in your tallied cell. For example, the following cards

F14:n 1
FM14 (5.23599E+08) (-5.23599E+08 10 (-6) (-6 -7) (-2))

shows how to obtain the total neutron fluence, total fission, total fission neutrons and total neutron absorptions in Cell **1**. Suppose the volume of Cell 1 is **5.23599E+08** cm^3, the first multiplier bin simply multiplies the flux tally by the cell volume to obtain the neutron fluence. In the second multiplier bin, **5.23599E+08** is again the volume of the cell, and **10** is the material number you defined by the **M** card. Three reaction multiplier lists are specified: **(-6)**, **(-6 -7)** and **(-2)**. From the reaction number list on Page 3-89 of the manual, you find:

For neutrons:
-2 absorption cross section
-6 total fission cross section
-7 fission ν

Therefore, **(-6)** gives the total fissions over Material 10 in the cell, **(-6 -7)** gives the total fission neutrons generated, and **(-2)** gives the total absorptions (captures) in the cell. A complete list of available reaction numbers can be found in Appendix G of the MCNP manual.