Chapter 12

CRITICALITY IN MCNP

For criticality calculations, the CODE card is required; together with an initial spatial distribution of fission points using either the KSRC card (with sets of x, y, z point locations), the SDEF cards (for points uniformly in volume) or an SRCTP file (from a previous criticality calculation). The purpose of criticality calculations is to determine the value of the effective multiplication factor, k_{eff}, by estimating the mean number of fission neutrons produced in one generation with respect to that in the previous generation. A generation is the life time of a neutron from birth to death by escape, parasitic capture, or absorption leading to fission. In MCNP, a generation is equivalent to a computed estimate of an actual fission generation (called a cycle). Fission neutrons are terminated in each cycle, but other neutron-producing reactions, such as $(n, 2n)$, do not cause termination. The effect of delayed neutrons is introduced by using the total average number of neutrons per fission, $\bar{\nu}$.

The KCODE card defines the nominal number of source histories, N per cycle, an initial guess for k_{eff}, the number of source cycles to skip before k_{eff} accumulation (useful if the initial guess for fission points is not very reliable) and the total number of cycles, among other parameters.

MCNP provides three main estimated for k_{eff}, based on collision, absorption or track-length. Combined estimates of k_{eff} is made also form these three estimators. Error estimates and corrrlations between the three estimates are also provided. Note that a correlation of unity (one) between two estimators means that no new information is gained from the second estimator. A zero correlation indicates that the two estimators are statistically independent and the combined standard deviation should be significantly less than either. A negative correlation indicates that one estimator is overestimating k_{eff}, the other is underestimating it, and the combination of the two should give larger improvement in the confidence level. If one does not have enough confidence in the results, several independent runs with different random number sequences (see BDCN card) should be made and the distribution of the obtained k_{eff} vales and the associated variances be examined. A good result will give consistent values.

MCNP provides a net multiplication factor, M, in fixed source problems in which neutron-multiplicative reactions, e.g. fissionable or $(n, 2n)$, are encountered. This M factor is not k_{eff}. It is equal to unity plus the neutron gain from fission and non-fission multiplicative reactions.

Before running a criticality calculation, read carefully the excellent discussion on criticality in the various chapters of the code's manual.