3.3 Discretizing the energy or speed range

In order to calculate the neutron distribution in every detail one would need to consider the population of neutrons with a particular energy and direction of motion at every location and at every moment in time and be able to analyze their collisions, production, and capture. This is an enormous computational challenge particularly since the cross-sections for those interactions are all complicated functions of the neutron energy. The problem is further complicated by the fact that the mean free paths are comparable with the dimensions of the detailed interior structure of the reactor core (for example the fuel rod diameter or coolant channel width). The general approach to this problem is known as *neutron transport theory*. The details of the general theory, for which the reader is referred to other classic texts such as Glasstone and Sesonske (1981) or Duderstadt and Hamilton (1976), are beyond the scope of this monograph. In part, this is because most practical calculations are performed only after radical simplifications that are necessary to arrive at a practical computation of the neutron dynamics in a practical reactor.

Before further deliberation of neutron transport theory some of the approximations that will be made later in the analysis can be anticipated. As implied in the preceding section the neutron energies represented in a reactor cover a wide range of speeds and, since each speed may have different cross-sections to various reactions, it becomes extremely complicated to incorporate all of these intricate details. Fortunately, it is sufficient for many purposes to discretize the energy range in very crude ways. The crudest approach is to assume that all the neutrons have the same energy, a thermal energy in thermal reactors since most of the heat produced is generated by fission that is proportional to the thermal neutron flux. This approach is further pursued in section 3.6.3.

One of the first hurdles experienced in implementing a method with a very crude discretization of the energy spectrum is the need to find average cross-sections that are applicable to the assumed, uniform energy within each sub-range. This can be effected by using the *reduced thermal models* described in section 2.3.3. Thus a one-speed thermal neutron model could have a single neutron energy of $E = 0.0253\ eV$ and an absorption cross-section of $\hat{\sigma}$. If the corresponding thermal neutron flux (called the *flux reduced to* $0.0253\ eV$) is also denoted by a hat, or $\hat{\phi}$, then the rate of absorption would be given by $\mathcal{N}\hat{\phi}\hat{\sigma}$. Henceforth, this averaging will be adopted and, for the sake of simplicity, the $\hat{}$ will be omitted and σ and ϕ will be used to denote the averaged cross-section and the averaged neutron flux.