3.8.1 Steady state lattice calculations

The preceding sections addressed solutions for the distribution of the neutron flux in reactors that were assumed to be homogeneous. As previously described in section 3.6.1, the assumption of homogeneity was based on the fact that the typical mean free path of the neutrons is normally large compared with the small scale structure within the reactor (for example, the fuel rod diameter). On the other hand the neutron mean free path is somewhat smaller than the overall reactor geometry and this provides some qualitative validity for variations in the neutron flux within the reactor that are derived from analytical methods such as those based on the diffusion equation (or any other more detailed methodology).

As described in section 3.6.7 the typical mean free path in a LWR is of the order of centimeters and therefore comparable with the diameter of a fuel rod. Consequently the variation of the neutron flux within and around the fuel rod of a LWR may be substantial and is therefore important to take into account in the design of those components. In contrast, a fast breeder reactor has typical mean free paths of the order of tens of centimeters. With fuel rod dimensions similar to a LWR, it follows that the inhomogeneity is less important in a fast reactor. In either case, practical reactor analysis and design requires detailed calculation of the variations in the neutron flux at these smaller scales and this can be effected using numerical codes called *heterogeneous lattice cell calculations*.

Thus it is appropriate to consider analytical methods that might be used to determine the variations in the neutron flux associated with the finer structure within a reactor core, for example the variations around a fuel rod or a control rod. In this endeavor, it is convenient to take advantage of the fact that much of this finer structure occurs in lattices or *units* that are repeated over the cross-section of the reactor (or at least parts of that cross-section). For example, each of the fuel rods are surrounded by coolant channels and other fuel rods in patterns that are described in section 4.3.4. Thus a fuel rod plus an appropriately allocated fraction of the surrounding coolant constitute a *unit* and those units are repeated across the reactor cross-section. Moreover since the neutron mean free path is comparable to or larger than the dimensions of this unit, it may be adequate to adjust the geometry of the unit to facilitate the mathematical solution of the neutron flux. Thus, as shown in figure 1, the geometry of a fuel rod unit might be modeled by a central cylinder of fuel pellets of radius, R_1, surrounded by a cylinder of neutronically passive and moderating material, $R_1 < r < R_2$, where the ratio of the areas of the two regions is the same as the ratio of the cross-sectional area of fuel pellet to the cross-sectional area of allocated non-pellet material within the reactor.

Before outlining some typical examples of the calculation of the neutron flux variations within a lattice cell, it is necessary to consider the nature of the boundary conditions that might be applied at the interfaces and boundaries of a cell such as that of figure 1. As described in section 3.6.1, at an interface such as $r = R_1$ not only should the neutron fluxes in the two regions be the same but the one-way fluxes must also be the same. In the context of diffusion theory these imply that at the interface the conditions should be as given in equation

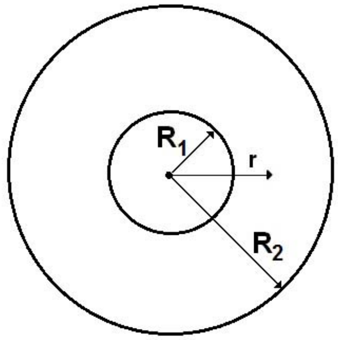

Figure 1: Generic circular lattice cell.

4, section 3.6.1. With the geometry of figure 1 these become:

$$\phi_1 = \phi_2 \qquad \text{and} \qquad D_1 \frac{\partial \phi_1}{\partial r} = D_2 \frac{\partial \phi_2}{\partial r} \tag{1}$$

Now consider the conditions on the outer boundary of the lattice cell ($r = R_2$ in figure 1). If the reactor is in a steady critical state each of the unit cells should be operating similarly with little or no net neutron exchange between them and therefore the condition on the outer boundary should be

$$\frac{\partial \phi_2}{\partial r} = 0 \quad \text{on} \quad r = R_2 \tag{2}$$

or the equivalent in more complex neutron flux models.

In the sections that follow diffusion theory solutions will be used to explore some of the features of these lattice cell models.