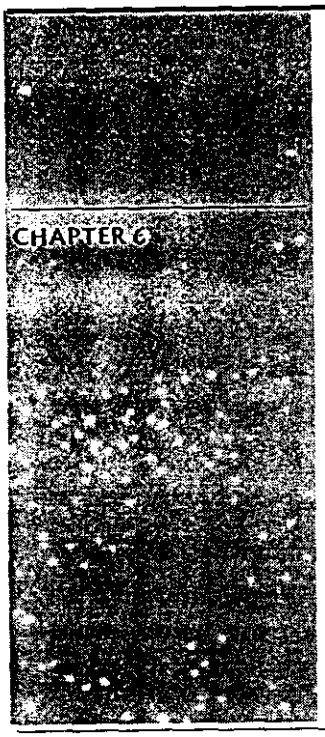


Spatial Mesh Considerations

A REACTOR CALCULATION will need at first a spatial domain partitioning in a three dimensional mesh, giving rise to a limited number of regions whose nuclear properties are constants. This process is examined in detail in this chapter. As an illustration, a simplified model of a CANDU-6 reactor in three dimensions is introduced.

Reactor Core

The core of most commercial nuclear reactors is made of a calandria, inside which is found a regular lattice of identicai cells from the geometry point of view. Mobile controllers and fixed structural materials are also present inside the calandria. A good neutronic model will take into account all of these.

Cell

The fuel cell of a CANDU-6 reactor type is illustrated on Figure 2, "Typical Cell of a CANDU-6 with 37 Elements", page 52. Note the presence of a 37 pin fuel bundles of 50 cm length approximately. The fuel pins contain natural Uranium in the form of Uranium oxide pellets. These pellets are surrounded by a Zircalloy sheath. Heavy water coolant circulates between the fuel pins, and is confined with the fuel bundles inside a cylindrical pressure tube, also made of a Zirconium alloy. A calandria tube (Zirconium alloy also) separates the pressure tube from the cold moderator. Between the calandria tube and the pressure tube CO_2 gas circulates. The moderator is used to slow down the fast neutrons emerging from the fission process to thermal energies, to increase the fission probability. The moderator is heavy water also, in order to minimize parasitic neutron captures.

Each cell contains all of these elements. The standard CANDU cell has 28,575 cm on the two sides, and about 50 cm axial length. There is a repetitive placement of this cell in the core:this is known as the cell lattice. A typical CANDU-6 has 380 fuel channels, containing 12 fuel bundles each, for a total of 4560 fuel bundles, and thus a total of 4560 cells.

In the current state of computer resources, it is still out of the question to solve even with the lowest order approximations the transport equations taking into account the complete geometry of the reactor.

Instead, the hypothesis of the separation of the flux into two components, the product of the microscopic flux by the macroscopic flux, is used.

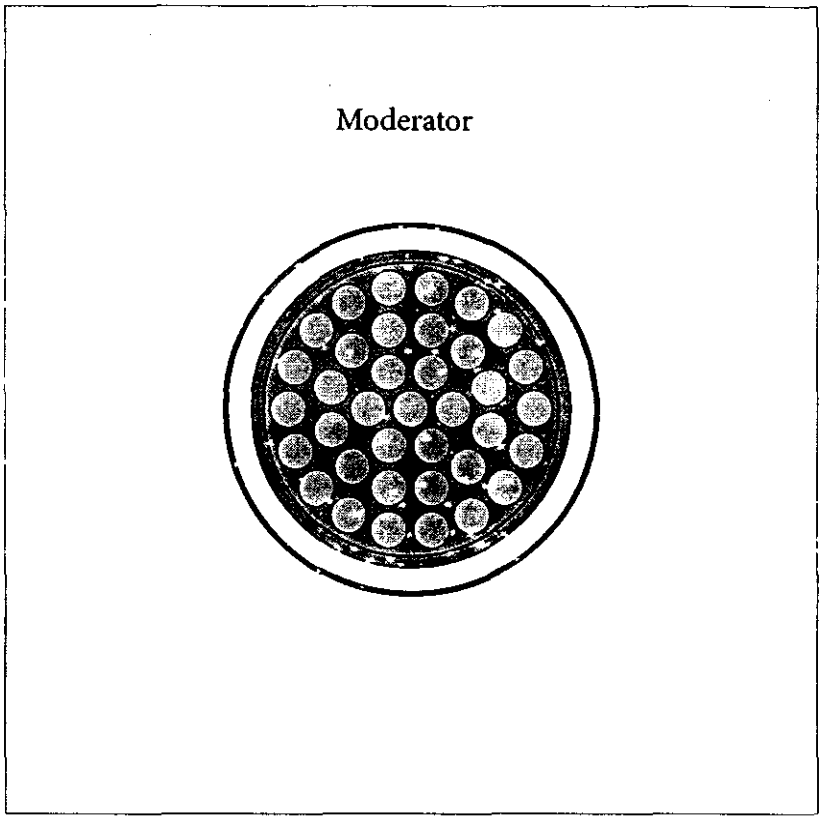
The microscopic flux is obtained from lattice cell calculations, in which the transport equation is solved, often by the use of first collision probabilities for example. A cell is supposed to be imbedded in an infinite lattice of identical cells, which permits using reflection boundary conditions on the external surfaces of the cell.

The cell calculation is followed by both a spatial homogenization and an energy condensation of nuclear cross-sections and diffusion coefficients. The spatial homogenization gets rid of the geometric description of the cell for subsequent diffusion theory calculations. The cell is effectively replaced by a homogeneous material whose cross-sections and diffusion coefficients are constant (space independent) on the complete cell volume. The energy condensation will generate these homogeneous properties on a small number of energy groups, generally two for thermal reactor analysis.

These cell calculations will provide the properties of cells containing fuel, and the properties of cells containing only moderator, for the reflector zones for example.

The diffusion calculations with homogenized cells will provide the macroscopic component of the flux and constitute the object of the different methods described in subsequent chapters.

FIGURE 2. **Typical Cell of a CANDU-6 with 37 Elements**

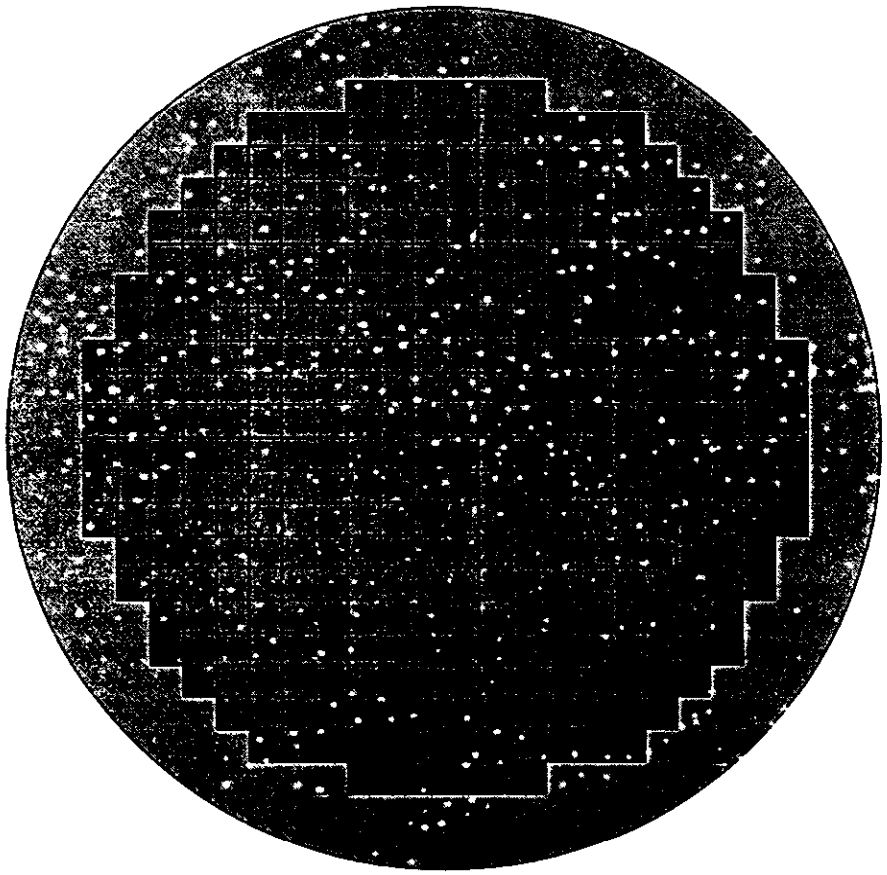


Raw Core

An adequate coordinate system will first be chosen, to permit accurate positioning of core elements. The first step consists in drawing with accuracy the reactor geometry, taking into account the exact cell positions, as illustrated on Figure 3, "Raw Geometry of a CANDU-6", page 53. Note the cells are arranged in such a fashion that they are aligned both vertically and horizontally, and also axially. This applies

in general to most commercial reactor types, such as CANDU, PWR, and BWR. This implies that cartesian geometry will be chosen for the spatial discretisation methods.

FIGURE 3. **Raw Geometry of a CANDU-6**

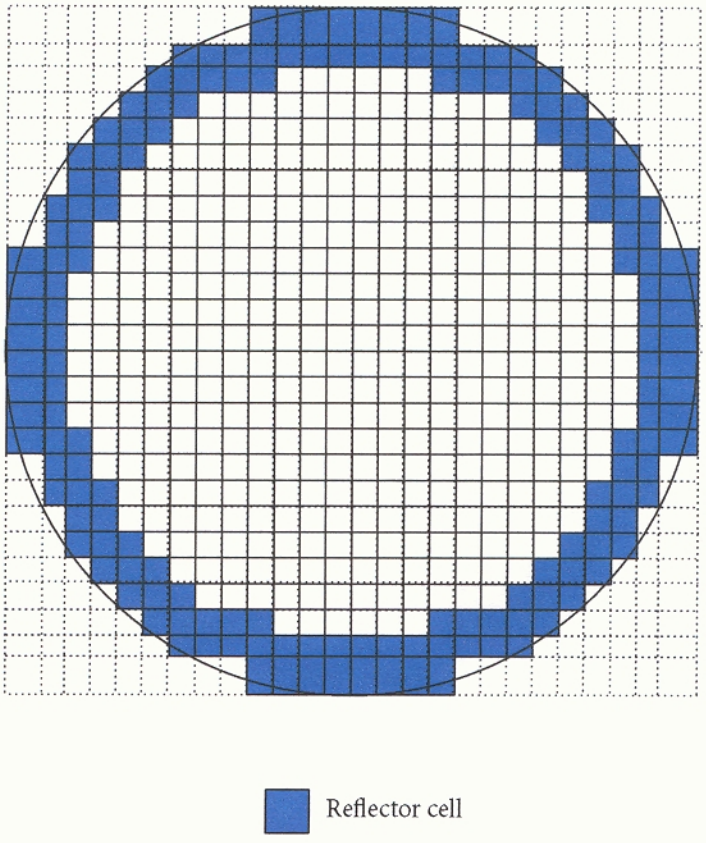


Such a drawing will be made for one plane at least; the CANDU-6 comprises 12 such planes. If there are planes with different geometries, the process will have to be repeated for each of them. It is generally the

case for the CANDU-6 because the planes at the axial extremities have a smaller thickness of reflector than the central planes. This is to save on the heavy water inventory of the core.

The cell lattice lines are then extended far enough to encompass the external boundaries of the reactor; these are the mesh lines. Special care must be taken at the external boundary, which is of cylindrical shape. The mesh lines will then have a staircase shape. Certain correction factors can also be implemented in discretisation techniques to take into account the rounded shape of the actual boundary.

FIGURE 4. **Mesh Lines for a CANDU-6**



Certain areas of the core are left empty. The final choice of which one should be empty is left to the analyst.

This preliminary step of the geometrical description of the core will be finished by the construction of a table that specifies either the coordinates of the mesh lines or the mesh widths (distances between cell boundaries) in each of the three dimensions, according to the specifications of the reactor code. For example, the DONJON computer code

uses the mesh coordinates for the core geometry. An example of this is shown on Table 1, "Cell Coordinates of the CANDU-6", page 56.

TABLE 1. Cell Coordinates of the CANDU-6

no	pos x	Δx	pos y	Δy	pos z	Δz
1	3.150	41.900	3.150	41.900	3.94	49.53
2	45.050	23.475	45.050	23.475	53.47	49.53
3	68.525	28.575	68.525	28.575	103.0	49.53
4	97.100	28.575	97.100	28.575	152.53	49.53
5	125.675	28.575	125.675	28.575	202.06	49.53
6	154.250	28.575	154.250	28.575	251.59	49.53
7	182.825	28.575	182.825	28.575	301.12	49.53
8	211.400	28.575	211.400	28.575	350.65	49.53
9	239.975	28.575	239.975	28.575	400.18	49.53
10	268.550	28.575	268.550	28.575	449.71	49.53
11	297.125	28.575	297.125	28.575	499.24	49.53
12	325.700	28.575	325.700	28.575	548.77	49.53
13	354.275	28.575	354.275	28.575	598.3	—
14	382.850	28.575	382.850	28.575		
15	411.425	28.575	411.425	28.575		
16	440.000	28.575	440.000	28.575		
17	468.575	28.575	468.575	28.575		
18	497.150	28.575	497.150	28.575		
19	525.725	28.575	525.725	28.575		
20	554.300	28.575	554.300	28.575		
21	582.875	28.575	582.875	28.575		
22	611.450	28.575	611.450	28.575		
23	640.025	28.575	640.025	28.575		
24	668.600	28.575	668.600	28.575		
25	697.175	23.475	697.175	23.475		
26	720.650	41.900	720.650	41.900		
27	762.550		762.550			

One can now identify on the drawing the reflector zones and the fuel zones, and by leaving empty the areas that do not contain any materials.

The mesh formed in this way will be regular. All lines start from one extremity of the core to the other. A large regular parallelepiped is thus constructed in three dimensions; we see one of these on Figure 5, "Reflector and Fuel Positions", page 58.

In practice, the fuel properties will vary from cell to cell in the reactor. Different neutronic models of varying complexity will introduce many different types of fuel. For example, the fuel burn-up can vary from one fuel bundle to the other, as well as local properties, such as temperatures and densities.

The usual practice is to assign a numerical index to each cell type whose properties are unique in the reactor model. The CANDU-6 model that we use as an example is a very simple one, in which only two fuel types and one reflector type are present.

This is illustrated on Figure 6, "Material Index", page 59, while the fuel properties are given on Table 2, page 59.

FIGURE 5. Reflector and Fuel Positions

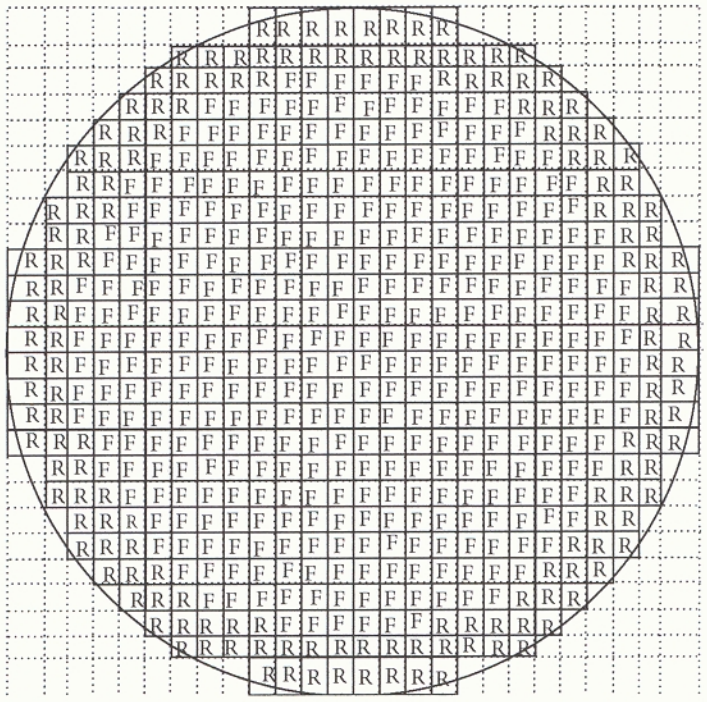


FIGURE 6. **Material Index**

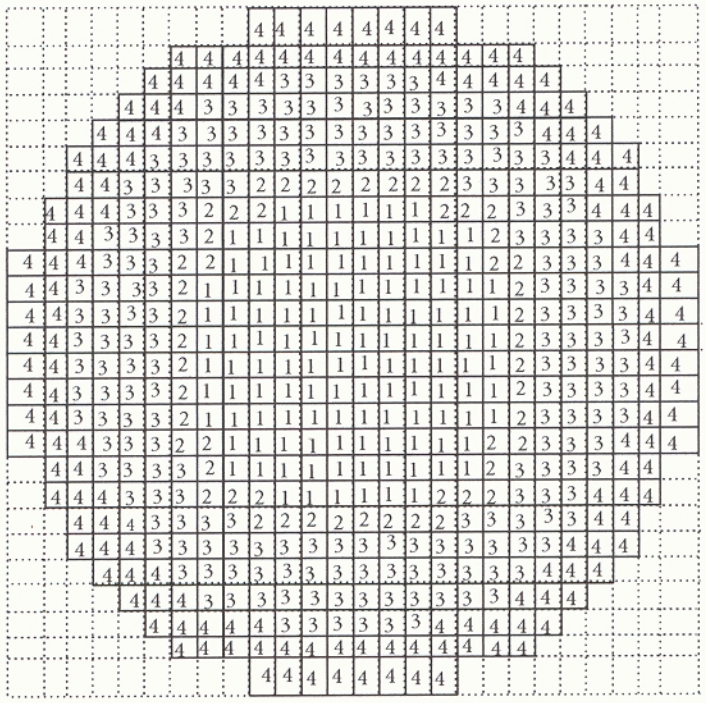


TABLE 2. Nuclear Properties

index	group	D	Σ_r	Σ_{sout}	$\nu\Sigma_f$
1	1	1.0693374	$1.1520207X10^{-2}$	$9.7073093X10^{-3}$	$9.1408670X10^{-4}$
—	2	0.79253197	$3.8237572X10^{-3}$	$1.9508027X10^{-6}$	$4.4642389X10^{-3}$
2	1	1.0693374	$1.1520207X10^{-2}$	$9.7073093X10^{-3}$	$9.1408670X10^{-4}$
—	2	0.79253197	$3.8237572X10^{-3}$	$1.9508027X10^{-6}$	$4.4642389X10^{-3}$
3	1	1.0692910	$1.1520118X10^{-2}$	$9.7108223X10^{-3}$	$9.1865181X10^{-4}$
—	2	0.79261804	3.7947595X10-3	$1.9387262X10^{-6}$	$4.4842800X10^{-3}$

TABLE 2. Nuclear Properties

index	group	D	Σ_r	Σ_{sout}	$\nu\Sigma_f$
4	1	1.3025166	1.2713954×10^{-2}	1.2713968×10^{-2}	0.0000
—	2	0.95088625	4.2647123×10^{-5}	1.0060013×10^{-6}	0.0000

These cross-sections are those of a very simple two region homogeneous "time-average" model. Fuel material of type 1 and type 2 have the same nuclear properties, and correspond to a central zone of high fuel burn-up. Material 3 corresponds to an external annulus of lower burn-up fuel. These properties, and those of the reflector, material 4, have been obtained with the lattice code DRAGON.

Boundary Conditions

The boundary conditions on the external surfaces of the core model have to be specified in order to complete the first basic model of the core. Many choices are possible, such as zero flux, zero flux at extrapolated boundary, or albedo.

Basic Model

Mesh layout and preliminary determination of the nuclear properties of fuel and reflector constitute the first phase in the process of setting up a realistic core model. The very crude model of the CANDU-6, with 12 identical planes already accounts for 6528 non empty regions.

Still to be considered is the presence of controllers and structural materials to have a complete reactor model. These new materials may imply a refinement of the basic mesh layout in order to represent them carefully in the core. Questions concerning spatial discretisation errors might also affect the final mesh choice.