CHAPTER 2

Parameter Definitions

Introduction

THE PARAMETERS USED in our derivation of the space-time kinetics are essentially the neutron flux and different cross-sections. These basic concepts will not be reviewed from a fundamental viewpoint here. We suppose that the reader is already familiar with these concepts which have been defined rigorously in a reactor physics course. Furthermore, we will presume in this course that the theory of neutron diffusion constitutes an adequate model for the description of the neutron population in a nuclear reactor.

This aspect of the question should always be examined for each new situation. It is the duty of the reactor analyst himself to determine if these approximations are acceptable for both the steady-state and the transient situations that he will have to simulate. The generalization to the neutron transport equations should not pause major difficulties in theory, if we exclude the problems related to obtaining the solution to the resulting equations...

Neutron Flux and Neutron Density

The neutron population in a nuclear reactor is described by the neutron density N, whose physical meaning is that $N(\vec{r}, \vec{\Omega}, E, t)$ represents the total number of neutrons present in the time interval dt at time t, in the element of volume dV surrounding the point $\vec{r}$ in space, whose energy is comprised between E and E+dE and travelling in a cone $d\Omega$ surrounding the direction $\vec{\Omega}$.

The neutron flux ϕ is used more often and provides the same information. The neutron flux is simply the neutron density multiplied by the speed of the neutrons. Since we will mostly be interested by diffusion theory, let us mention that after integration over the solid angles of the expansion in spherical harmonics of the $\vec{\Omega}$ terms, the flux and the neutron density then become $\phi(\vec{r}, E)$ and $N(\vec{r}, E)$.These are known as the total flux and total density, and are still densities of space and of energy.

Neutron Current

In the same way as for the neutron density, the neutron current can be defined in terms of the state variables. However, after passing to diffusion theory, we can say that the neutron current $\vec{J}(\vec{r}, E, t)$ when multiplied by a surface element $d\vec{S}$, by the energy interval dE and by the time interval dt represents the net number of neutrons whose energies are between E and $E + dE$ crossing the surface $d\vec{S}$ in the time interval dt.

Cross-sections

The microscopic neutron cross-sections of a nuclear material are related to the probabilities that a neutron will undergo an interaction of the type described by the microscopic cross-section. This probability is given by the expression $vn\sigma^{\alpha}dtdV$ where v is the speed of neutrons of energy E, n is the atomic density (in atoms per cc) of the material and σ^{α} is the microscopic cross-section for the interaction of type α for neutrons of energy E. Therefore the probable number of type α reactions with neutrons whose energy are between E and $E + dE$ in the volume element dV in the time interval dt becomes $N(\vec{r}, E, t)vn\sigma^{\alpha}dEdtdV$. We use the definition of the macroscopic cross-section $\Sigma^{\alpha} = n\sigma^{\alpha}$ and of the neutron flux $\phi = vN$ to obtain that this number of interactions is simply $\Sigma^{\alpha}\phi dEdVdt$.

We will also suppose that the neutron density is sufficiently large that the statistical fluctuations of this number of interactions are negligible. This permits a deterministic theory describing the distribution of the average number of neutrons in space and time, as well as for the average number of interactions of different types that these neutrons undergo.

Delayed Neutrons and Precursors

The fission of a fissile isotope generally produces the emission of a certain number of neutrons simultaneously with the production of two fission products. These fission products are often unstable isotopes which will eventually undergo β disintegration, some of which will also emit neutrons in the process.

The average number of neutrons emitted by fission, ν, takes into account all neutrons produced by the fission process, either simultaneously with the fission (prompt neutrons), or indirectly by the disintegration of some of the resulting fission products (delayed neutrons). Of all these neutrons, a fraction β are delayed neutrons; $(1 - \beta)$ then constitutes the prompt fraction. The total number of fission product atoms giving rise to delayed neutron emissions will depend on the neutron flux in the reactor.

The precise determination of the number of each fission product, of their disintegration constant and of the energy spectrum of the neutrons produced by these disintegrations, constitutes an important task of experimental nuclear physics. The total number of fission product types (about 800...) is usually not determined during space time kinetics calculations. In order to simplify the analyses, the fission products with similar disintegration constants are grouped into precursor families. The characteristics of these families are determined by nuclear physics experiments. The effective disintegration constant λ_i as well as the effective fraction β_i of the fission products belonging to each of these families are determined by statistical analysis of this data. In space time kinetics, six families of delayed neutron precursors are usually used, because of the excellent fit to the nuclear data obtained with this number.

However, it is sometimes necessary to include more than six families, depending on the type of calculation to be done. For example, the determination of the neutron population a long time after a shutdown will necessitate more precision concerning the precursors with long half lives. In the case of CANDU reactors, it will sometimes be necessary to include photo neutron emission in separate groups.

We introduce a concentration of delayed neutron precursors for the family i. It is denoted $C_i(\vec{r}, t)$ and represents the fictitious number of atoms per cc of this precursor at point $\vec{r}$ producing a fraction β_i of the total number of fission neutrons with a disintegration constant λ_i.

Let us note here the very important role that delayed neutrons play in a nuclear reactor. The total number of precursors is much greater than that of the neutrons. As the number of delayed neutrons changes with time constants that are long compared to the prompt neutron generation time, the neutron population will in part be governed by these slow time constants, and reactor control becomes possible.

Prompt and Delayed Neutron Spectra

Neutrons, whether prompt or delayed, are emitted with a certain energy distribution. We denote by $\chi^p(E)dE$ the probability that a prompt neutron is born with an energy E in the interval dE. Most of prompt neutrons appear with high energies (around 2 MeV). In the same way, delayed neutrons appear with an energy distribution that can vary (albeit slightly) according to the precursor family. Generally speaking, delayed neutrons also appear with high energies. The probability that a delayed neutron of the family i appears in the energy interval dE is given by $\chi_i^d(E)dE$.