

CHAPTER 3: ELEMENTARY PHYSICS OF REACTOR CONTROL

- in order to design and operate a control system for a process, the response of the process must be understood and described mathematically, or known from experimental data, and in some case by a combination of these methods
- the physics of the nuclear reactions taking place in a power reactor, including the response of the neutron flux to various internal and external disturbances, is well known: the characteristics that are key for the control of nuclear reactors are described in this chapter.

MODULE A: NEUTRON MULTIPLICATION AND REACTIVITY

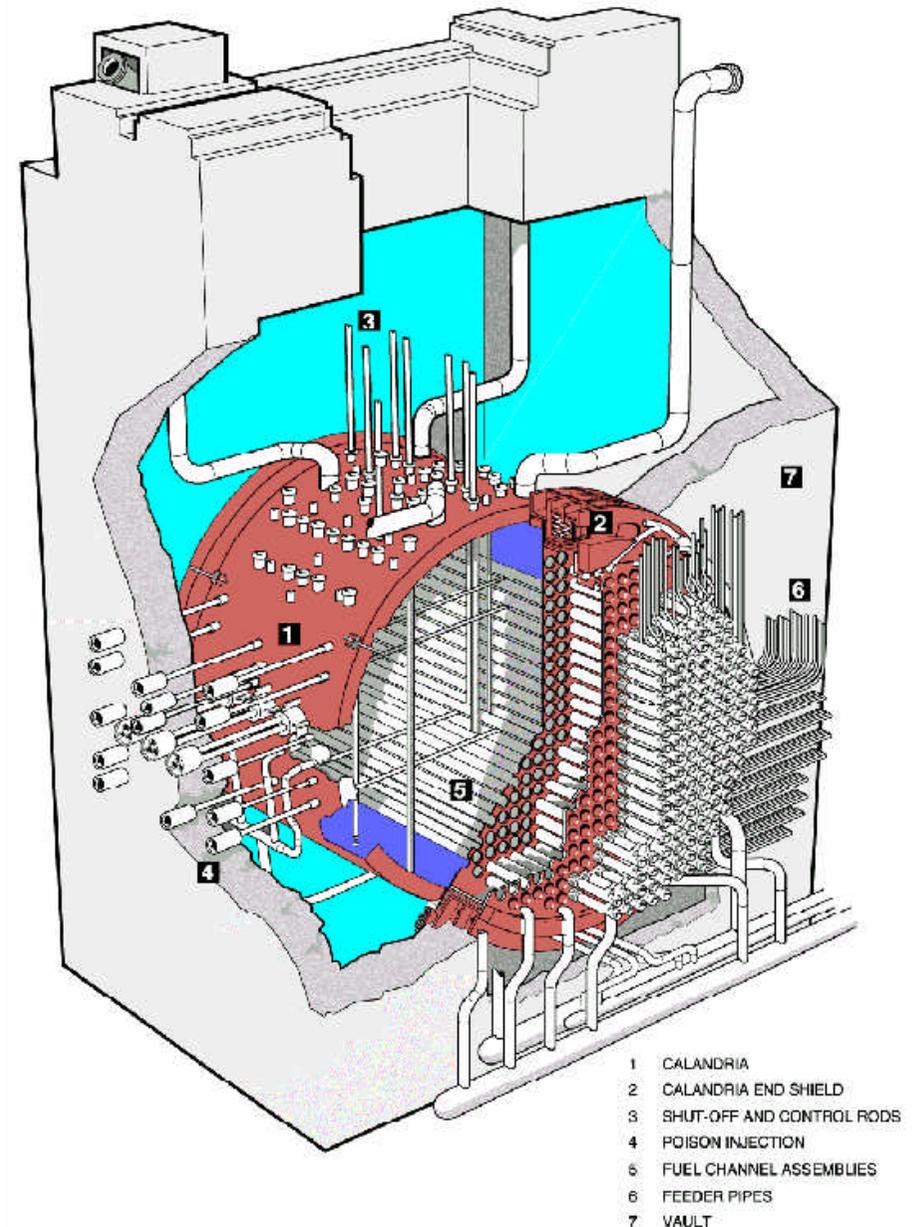
MODULE OBJECTIVES:

At the end of this module, you will be able to describe:

1. the meaning of the term 'multiplication factor'
2. the behaviour of a reactor that is subcritical, critical or supercritical
3. the neutron cycle in terms of the six factor formula, including
 - (a) Fast Fission Factor (ϵ)
 - (b) Resonance Escape Probability (p)
 - (c) Reproduction Factor (η)
 - (d) Thermal Utilization (f)
 - (e) Fast Non-leakage Probability (Λ_f)
 - (f) Thermal Non-leakage Probability (Λ_t)
4. Average Neutron Lifetime
5. Reactivity

1. INTRODUCTION

- the basis of energy production in a nuclear reactor is the conversion of mass to energy that occurs when a heavy nucleus such as U-235 or Pu-239 fissions, following the absorption of a thermal neutron
- in order for nuclear fission to continue in the form of a chain reaction, there must be a number of neutrons released in the fissioning process
- the amount of energy produced by the reactor is proportional to the number of fissions taking place, so the reactor's power level can be controlled by altering the number of neutrons between successive generations
- in this module we consider what happens to neutrons from the time they are 'born' in fission until they are absorbed in the many materials in the reactor, or leak out of the reactor
- an understanding of the 'neutron life cycle' will help us to understand the various ways that the chain reaction can be controlled



2. CONTROLLING THE CHAIN REACTION

The principle task of a reactor control system is to maintain control over the chain reaction, that is to control the number of neutrons in one generation relative to the number of neutrons in the previous generation. This ratio is defined as the multiplication factor (k):

$$k = \frac{\text{number of neutrons in one generation } (N_i)}{\text{number of neutrons in the previous generation } (N_{i-1})}$$

If the number of neutrons from one generation to the next are exactly equal, then $k = 1$, the reactor is said to be critical and the chain reaction will continue at a constant level. Note that a reactor can be critical at any power level.

If succeeding generations of neutrons are causing a decreasing number of fissions between generations, then $k < 1$, the reactor is said to be subcritical, and the power level of the reactor will be decreasing.

If succeeding generations of neutrons are resulting in an increase of the number of fissions between generations, then $k > 1$, the reactor is said to be supercritical, and the power level of the reactor will be increasing.

Note that the power level of a reactor can only change if the reactor is subcritical (power will decrease) or if the reactor is supercritical (power will increase).

Maintaining reactor power at a prescribed value (setpoint), or changing reactor power to a new value at a prescribed rate are functions that must be performed by the reactor control system. The reactor control system alters the value of 'k' by altering what happens to the neutrons during their 'life time' i.e. from the time they are 'born' until they either escape from the reactor (leakage) or are absorbed. For a critical reactor for each neutron born exactly one neutron must cause fission in the next generation.

There are six factors that govern the production, leakage and absorption neutrons, and enable the quantitative description of the components that govern the multiplication factor in the neutron cycle.

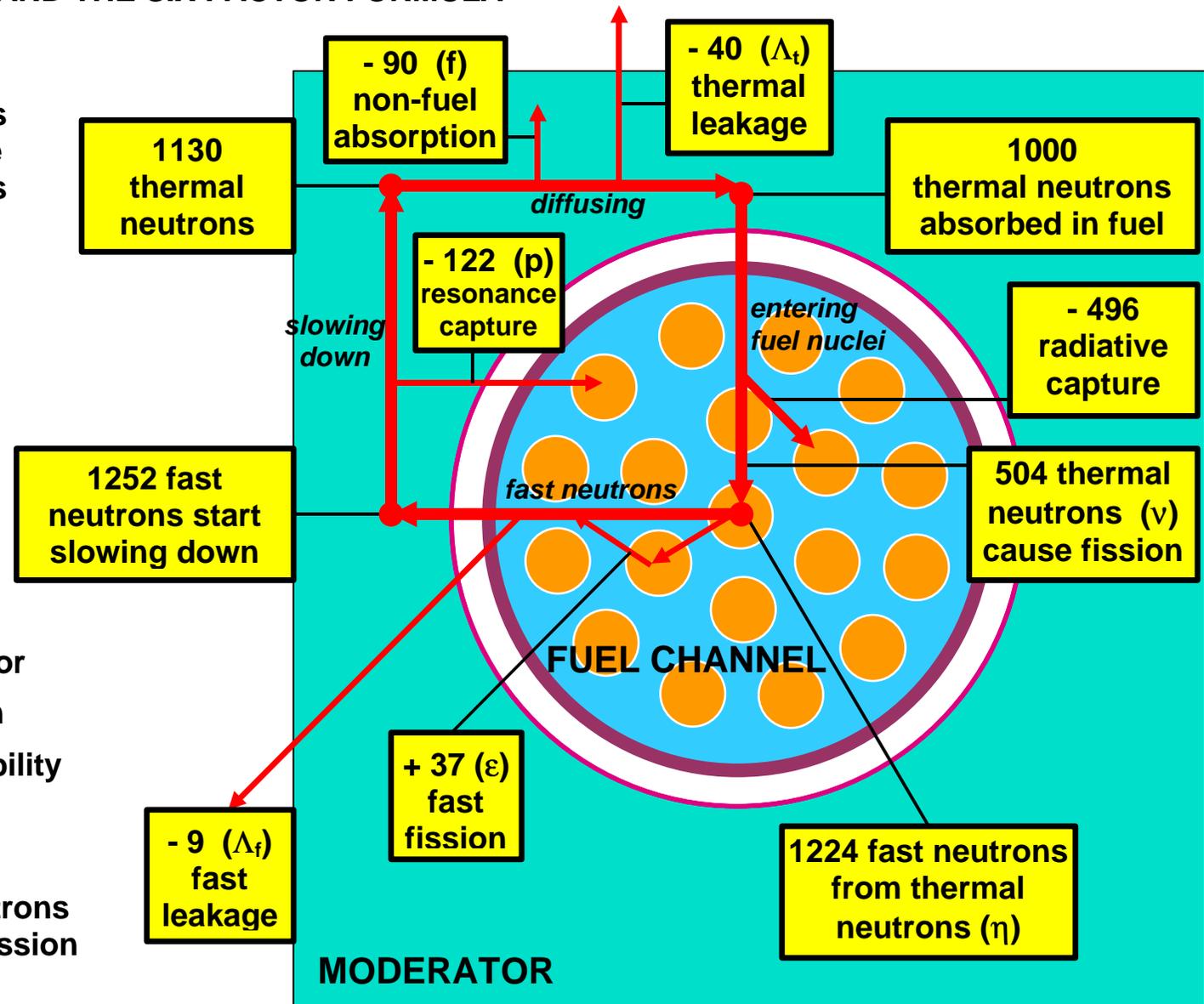
3. THE NEUTRON CYCLE AND THE SIX FACTOR FORMULA

The multiplication factor (k) expresses the ratio of the number of thermal neutrons in one generation (N_i) to the number of thermal neutrons in the previous generation (N_{i-1}) by six factors.

$$k = \frac{N_i}{N_{i-1}} = \epsilon p \eta f \Lambda_f \Lambda_t$$

where:

- ' ϵ ' is the fast fission factor
- ' p ' is the resonance escape probability
- ' η ' is the reproduction factor
- ' f ' is the thermal utilization
- ' Λ_f ' fast non-leakage probability
- ' Λ_t ' thermal non-leakage probability
- ' ν ' average number of neutrons produced per thermal fission



3.1 Fast Fission Factor (ϵ)

The factor by which the fast neutron population increases due to fast fission in U-238. For natural uranium a typical value is 1.03.

$$\epsilon = \frac{\text{number of neutrons from thermal fission} + \text{number of neutrons from fast fission}}{\text{number of neutrons from thermal fission}}$$

3.2 Resonance Escape Probability (p)

The probability that a neutron will not undergo resonance capture in U-238 while slowing down from fast to thermal energy levels. For natural uranium a typical value is 0.90.

$$p = \frac{\text{number of neutrons leaving resonance energy range}}{\text{number of neutrons entering resonance energy range}}$$

3.3 Reproduction Factor (η)

The number of fission neutrons produced per thermal neutron absorbed in the fuel. Note that this definition assumes that the term 'fuel' applies to the full content of the fuel pellets, i.e. all isotopes of uranium, plutonium and all fission products. For natural uranium a typical value for η is about 1.2.

$$\eta = v \frac{\sum_f (\text{fuel})}{\sum_a (\text{fuel})} = v \frac{\sum_f (\text{fuel})}{\sum_f (\text{fuel}) + \sum_{n,\gamma} (\text{fuel})}$$

where: \sum_a is the macroscopic absorption cross-section and \sum_f macroscopic fission cross-section,
 v is the average number of neutrons produced per thermal fission (typical value for v in natural uranium is 2.5).

Note that η is the number of fission neutrons per thermal neutrons absorbed in all the fuel pellet materials, while v is the average number of neutrons produced per thermal fission.

3.4 Thermal Utilization (f)

The fraction of thermal neutrons absorbed by the fuel (as defined above) compared to the total number of thermal neutrons absorbed in the whole reactor. For natural uranium a typical value is about 0.94.

$$f = \frac{\sum_a(\text{fuel}) \phi(\text{fuel})}{\sum_a(\text{total reactor}) \phi(\text{total reactor})}$$

where $\phi(\text{fuel})$ and $\phi(\text{total reactor})$ are the average thermal fluxes in the fuel and total reactor, respectively.

3.5 Fast non-leakage probability (Λ_f)

The probability that a fast neutron will not leak out of the reactor. Typical value for a CANDU core is 0.995.

3.6 Thermal non-leakage probability (Λ_t)

The probability that a thermal neutron will not leak out of the reactor. Typical value for a CANDU core is 0.98.

4. MULTIPLICATION FACTOR IN THE NEUTRON CYCLE

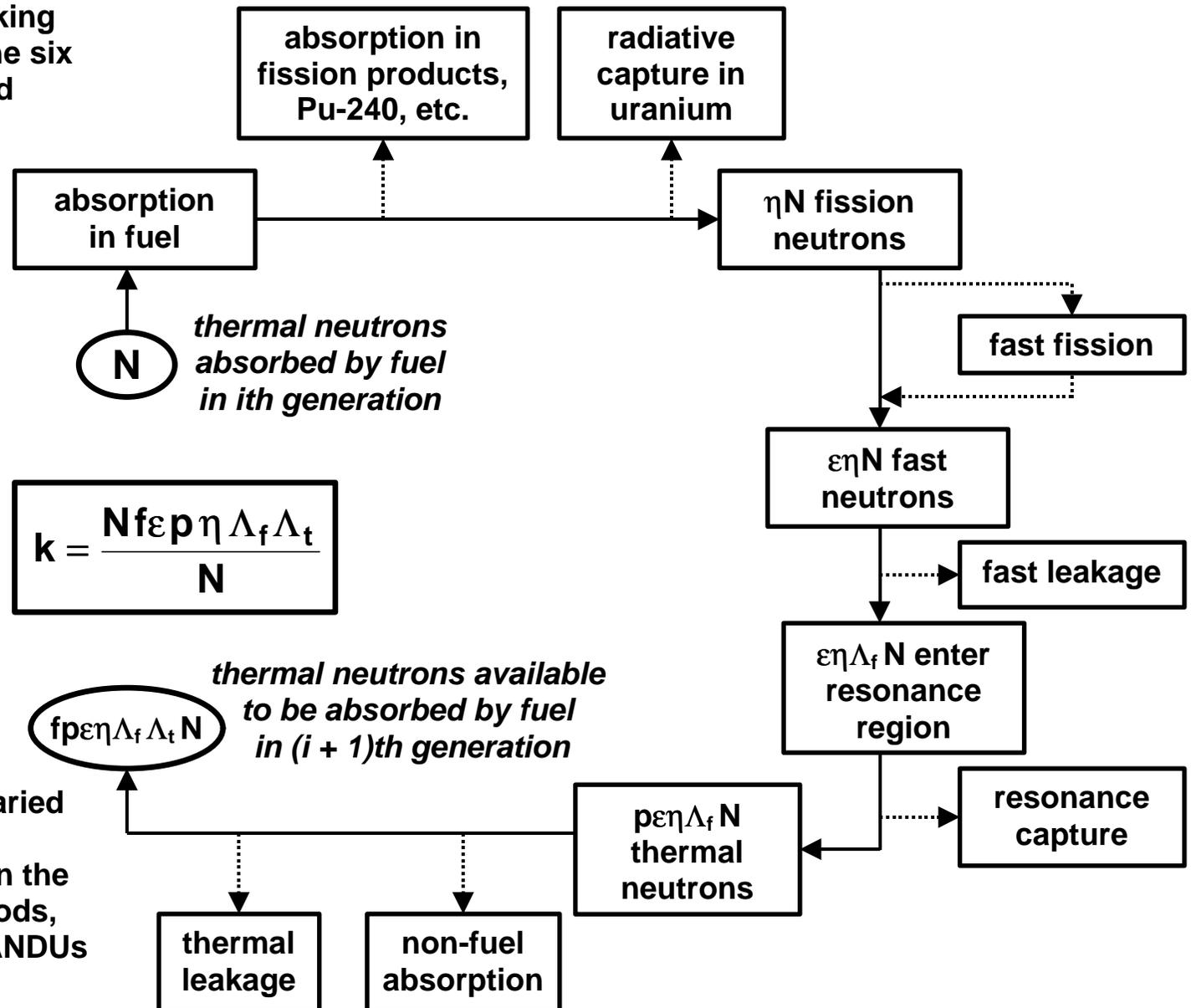
Reactor control involves making changes to one or more of the six factors to achieve the desired value of multiplication.

ϵ and p are fixed by the design of the core;

η can be changed in the long term by replacing spent fuel with fresh fuel, some early reactors used highly enriched 'booster' rods for making short term changes in η ;

Λ_f and Λ_t could be changed in early CANDU designs by altering the level of the moderator in the calandria;

f is the parameter that is changed for short term reactivity control in most thermal reactors: it can be varied by changing the amount of neutron absorbing material in the core, usually in the form of rods, or dissolved 'poisons'; in CANDUs light water can also be used.



5. REACTIVITY

Under normal operating conditions a reactor is operating at or very near criticality, i.e. k is nearly equal to 1. Small deviations of k above or below 1 will result in significant changes in reactor power. Under these circumstances it is more practical to work in terms of the amount by which k deviates from unity, instead of quoting several zeros before we reach any significant digits after 1. The term that has been found useful in reactor analysis to describe the behaviour of the reactor when k deviates from 1 is called 'reactivity'. It is a mathematically defined quantity that cannot be measured directly in practice:

$$\text{reactivity} = \frac{k - 1}{k}$$

As mentioned above, normal operating conditions involve values of k that are near 1, so the above expression can be approximated under these conditions as $k - 1$, and in our work we will use the following:

$$\text{reactivity} = k - 1 = \delta k$$

Because the reactivity changes involved in normal operations are small, the unit of reactivity that we find most useful is the milli- k , which is abbreviated as 'mk'.

For example a multiplication factor of $k = 1.003$ gives a $\delta k = 0.003$.

Note that for a typical CANDU reactor the primary means of power level control is by varying the level of light water in the zone control compartments. The reactivity change that can be achieved by emptying or filling all the compartments is ± 3 mk.