Aula 5 – Cinética de reatores nucleares

Reações principais

$$^{235}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \rightarrow \left({}^{236}_{92}\mathrm{U}\right)$$

$$^{235}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{A_1}_{Z_1}F_1 + {}^{A_2}_{Z_2}F_2 + \nu\ {}^{1}_{0}\mathrm{n} + \mathrm{energy}\ .$$

Exemplo:

$$^{235}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{90}_{36}\mathrm{Kr} + {}^{144}_{56}\mathrm{Ba} + 2{}^{1}_{0}\mathrm{n} + E$$

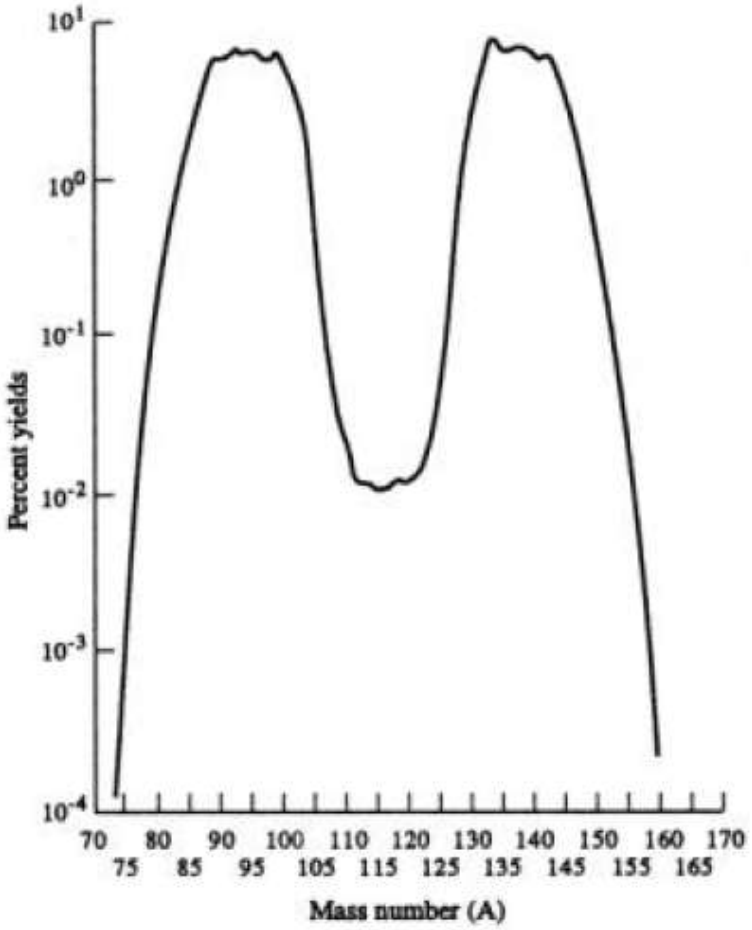


FIG. 6.3 Yield of fission products according to mass number (Courtesy of T. R. England of Los Alamos National Laboratory).

Reações secundárias

Produtos de fissão nêutrons atrasados (delayed)

Origin of ~55 sec. Delayed Neutron

$_0n^1 + {}_{92}U^{235} \rightarrow$ fission — $_{35}Br^{87}$ is a fission product

$_{35}Br^{87} \rightarrow {}_{36}Kr^{87} + {}_0\beta^{-1} + v$ — β-decay (neutron decays to proton)

$_{35}Br^{87} \rightarrow {}_{35}Br^{86} + {}_0n^1$ — neutron emission

Precursor	Precursor half-life (sec) and group assignment	
Br^{87}	54.5	Group 1
I^{137}	24.4	Group 2
Br^{88}	16.3	
I^{138}	6.3	Group 3
$Br^{(89)}$	4.4	
$Rb^{(93,\ 94)}$	~6	
I^{139}	2.0	Group 4
(Cs, Sb or Te)	(1.6–2.4)	
$Br^{(90,\ 92)}$	1.6	
$Kr^{(93)}$	~1.5	
(I^{140} + Kr?)	0.5	Group 5
(Br, Rb, As + ?)	0.2	Group 6

* From G. R. Keepin, *Physics of Nuclear Kinetics*, Reading, Mass.: Addison-Wesley, 1965.

Group	Half-life (seconds)	Mean life, t_{mi} (seconds)	Decay constant λ_i (second^{-1})	Fraction of total fission neutrons β_i
1	55·7	80·2	0·0124	0·000 215
2	22·7	32·7	0·0305	0·001 424
3	6·2	8·9	0·111	0·001 274
4	2·3	3·3	0·301	0·002 568
5	0·61	0·88	1·14	0·000 748
6	0·23	0·33	3·01	0·000 273

The total fraction of neutrons which are delayed β is:

$$\beta = \sum_{i=1}^{6} \beta_i$$

and its value for thermal fission in ^{235}U is 0·0065

$$^{135}Te \xrightarrow[19\ s]{\beta^-} {}^{135}I \xrightarrow[6.7\ h]{\beta^-} {}^{135}Xe \xrightarrow[9.2\ h]{\beta^-} {}^{135}Cs \xrightarrow[2.6\ My]{\beta^-} {}^{135}Ba \text{ (stable)}$$

Modelagem cinética de reatores nucleares

O termo que se adotou como bastante prático na análise do reator para descrever seu comportamento quando k desvia de 1 é chamado de reatividade, e é o parâmetro mais importante na operação do reator nuclear

$$K = \frac{\text{Número de fissões da geração atual } (N_i)}{\text{Número de fissões da geração precedente } (N_{i-1})}$$

O valor de k do sistema pode ser classificado como:

$k < 1$, sistema subcrítico⇒ não há reação em cadeia auto-sustentada.

$k = 1$, sistema crítico⇒ reação em cadeia no estado estacionário.

$k > 1$, sistema supercrítico⇒ reação em cadeia crescente.

$$k_\infty = \frac{\text{neutron production from fission in one generation}}{\text{neutron absorption in the preceding generation}}$$

Reatividade

Reatividade é a medida do afastamento da posição de criticalidade em função do fator de multiplicação

$$\rho = \frac{k_\infty - 1}{k_\infty}$$

Quando a reatividade é positiva, o reator é supercrítico; quando é zero, ele está critico, e, quando negativa, o reator está subcrítico.

A unidade para a reatividade de uso comum nos reatores de potência é o pcm (“por cem mil”), que é igual a um valor de ρ de 10-5 (= 1 pcm)

Example:

Calculate the reactivity in the reactor when k_{eff} is equal to 1.002 and 0.998.

Solution:

The reactivity for each case is determined by substituting the value of k_{eff} into Equation (3-5).

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \qquad \rho = \frac{k_{eff} - 1}{k_{eff}}$$

$$= \frac{1.002 - 1}{1.002} \qquad = \frac{0.998 - 1}{0.998}$$

$$= 0.001996 \qquad = -0.0020$$

Another important parameter in the study of reactor kinetics is the prompt neutron lifetime, l_p. In an infinite reactor the prompt neutron lifetime is the average time between the birth of prompt neutrons by fission and their final absorption in the reactor. In a thermal reactor this time is the sum of the average neutron slowing-down time (during which neutrons are slowing down from fission to thermal energy), and the average diffusion time (during which neutrons are diffusing at thermal energy up to their point of absorption). In all thermal reactors the diffusion time is much greater than the slowing-down time, typical values being about 10^{-3} seconds and 10^{-5} seconds respectively, so that the prompt neutron lifetime is very nearly equal to the diffusion time. The average diffusion time t_d for thermal neutrons in a reactor is:

$$t_d = \frac{\text{Absorption mean free path of thermal neutrons in the reactor}}{\text{Average speed of thermal neutrons}}$$

$$t_d = l_p = 1{\cdot}18 \times 10^{-3} \text{ seconds}$$

This value, namely about 0·001 seconds, is typical of graphite-moderated reactors. Water-moderated, enriched uranium reactors have typical prompt neutron lifetimes of about 0·0001 seconds, and fast reactors, in which neutrons do not become thermalized, have prompt neutron lifetimes of about 10^{-7} seconds.

Modelagem cinética de reatores nucleares

$$\begin{pmatrix}\text{The rate of change}\\ \text{of neutron density}\end{pmatrix} = \begin{pmatrix}\text{The rate of production}\\ \text{of prompt neutrons}\end{pmatrix} + \begin{pmatrix}\text{The rate of decay of all}\\ \text{delayed neutron precursors}\end{pmatrix} - \begin{pmatrix}\text{The rate of}\\ \text{absorption of neutrons}\end{pmatrix}$$

$$\begin{pmatrix}\text{The rate of change of the}\\ \text{concentration of the } i\text{th group}\\ \text{of delayed neutron precursors}\end{pmatrix} = \begin{pmatrix}\text{The rate of formation of}\\ \text{the } i\text{th group of precursors}\end{pmatrix} - \begin{pmatrix}\text{The rate of decay of the}\\ i\text{th group of precursors}\end{pmatrix}$$

$$\frac{dN(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda} N(t) + \sum_{i=1}^{6} \lambda_i C_i(t)$$

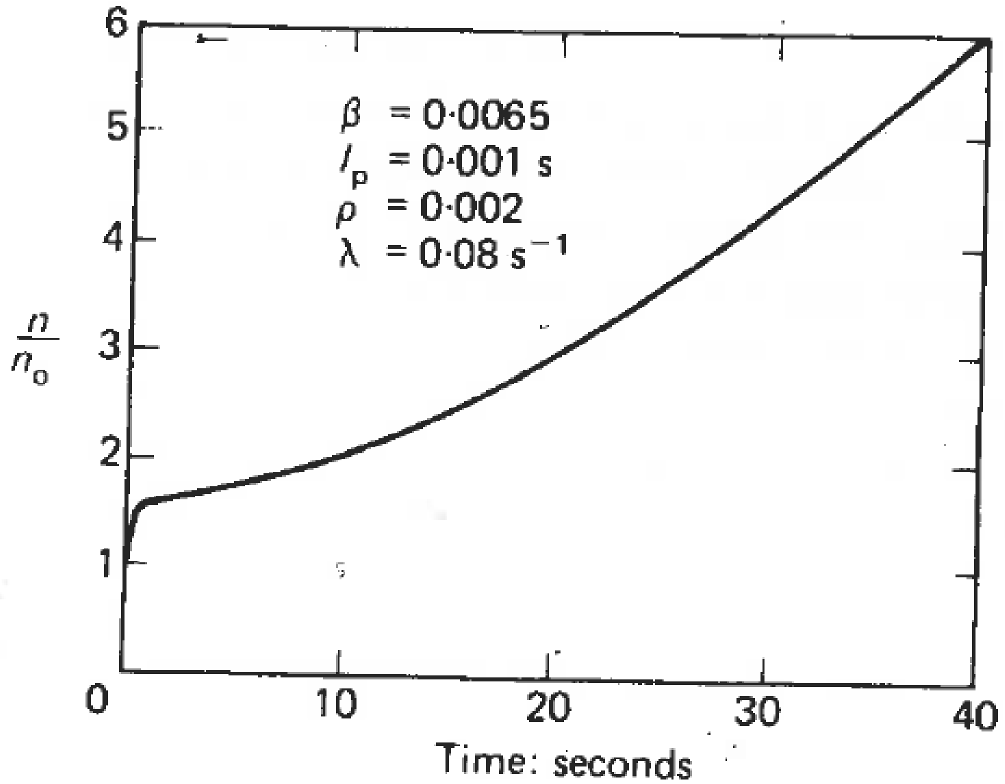
Duderstadt and Hamilton, 1976

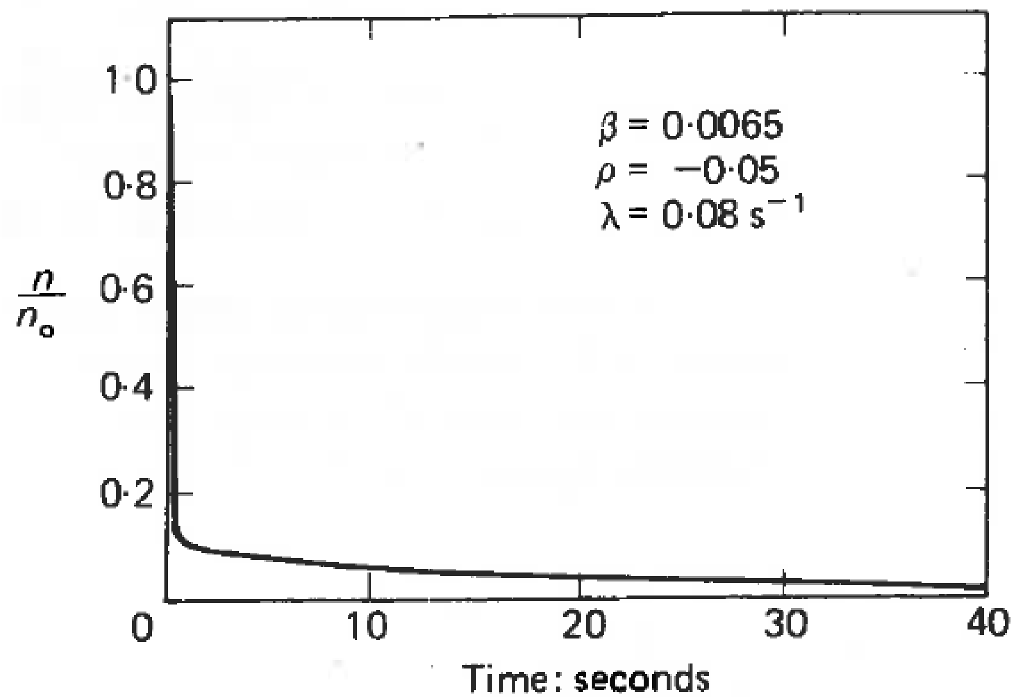
$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} N(t) - \lambda_i C_i(t)$$

The value of the decay constant λ of the single group of delayed neutron precursors is given by:

$$\lambda = \frac{1}{\bar{t}_m}$$

where $\bar{t}_m$ is given by equation (8.11). The value of λ for a ^{235}U fuelled reactor is about 0·08 second^{-1}.





1·0
0·8
0·6
0·4
0·2
$\frac{n}{n_o}$
β = 0·0065
ρ = −0·05
λ = 0·08 s^{-1}
0
10
20
30
40
Time: seconds

In a reactor containing a large amount of fertile material, for example a natural or slightly enriched uranium reactor, the burnup of the original fissile isotope is offset to some extent by the production of new fissile material. In the case of a uranium fuelled reactor the important processes are:

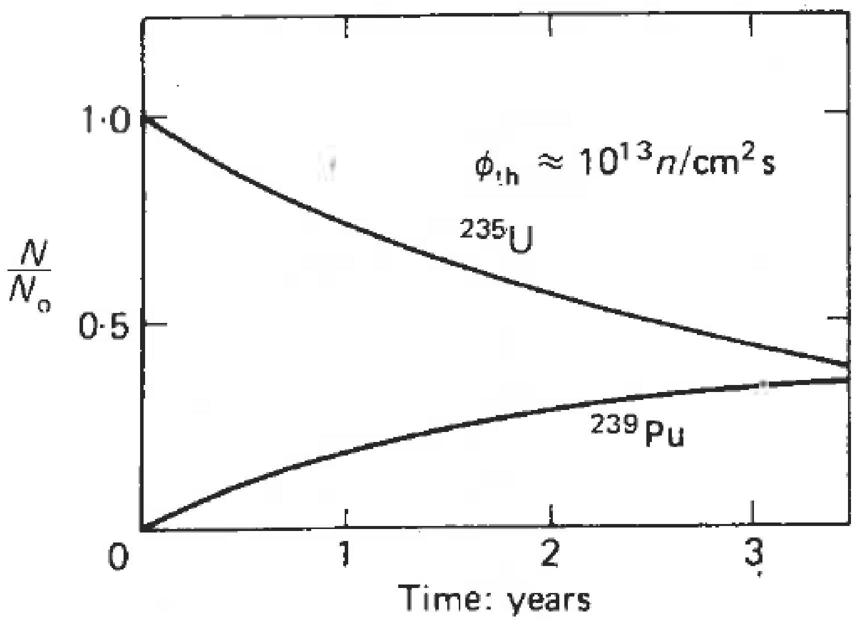
$$^{235}U + n \begin{cases} \nearrow\ ^{236}U \\ \searrow\ \text{fission} \end{cases}$$

$$^{238}U + n \rightarrow\ ^{239}U \rightarrow\ ^{239}Np \rightarrow\ ^{239}Pu$$

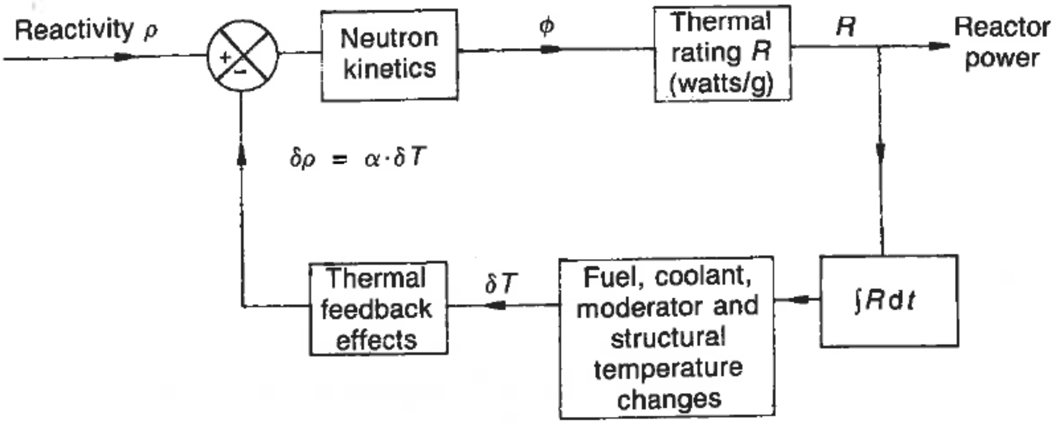
$$^{239}Pu + n \begin{cases} \nearrow\ ^{240}Pu \\ \searrow\ \text{fission} \end{cases}$$

$$^{240}Pu + n \rightarrow\ ^{241}Pu$$

$$^{241}Pu + n \begin{cases} \nearrow\ ^{242}Pu \\ \searrow\ \text{fission} \end{cases}$$



The concentration of ^{235}U and ^{239}Pu in an operating reactor



Thermal feedback in nuclear reactors

Exercício – Aula nuclear 5 – Prof . Juan

1) Simular a cinética de um reator nuclear sem moderadores e com moderadores obtendo o fluxo de nêutrons no reator (N/No) em função do tempo e as concentrações dos 6 compostos que geram nêutrons atrasados em função do tempo.

Sistema de EDos sem moderadores

$$\frac{dN(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda} N(t) + \sum_{i=1}^{6} \lambda_i C_i(t) \qquad (1)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} N(t) - \lambda_i C_i(t) \qquad (2)$$

Duderstadt and Hamilton, 1976

Equações de EDos com moderadores

$$\frac{dN(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda} N(t) \left(1 - \frac{N}{N_{\text{MODERADO}}}\right) + \sum_{i=1}^{6} \lambda_i C_i(t)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} N(t) - \lambda_i C_i(t)$$

Dados:

λ_i = 0.0127, 0.0317, 0.155, 0.311, 1.4, 3.87

β_i = 0.000266, 0.001491, 0.001316, 0.002849, 0.000896, 0.000182

Λ = 0.00002

$\boldsymbol{\rho}=0.0099;\boldsymbol{\beta}=0.007;\ \Lambda = l_p = 0.00002;$ Nmod=1.01