Chemical Engineering 412

Introductory Nuclear Engineering

Lecture 8

Radiation Interactions with Matter

Exam Review

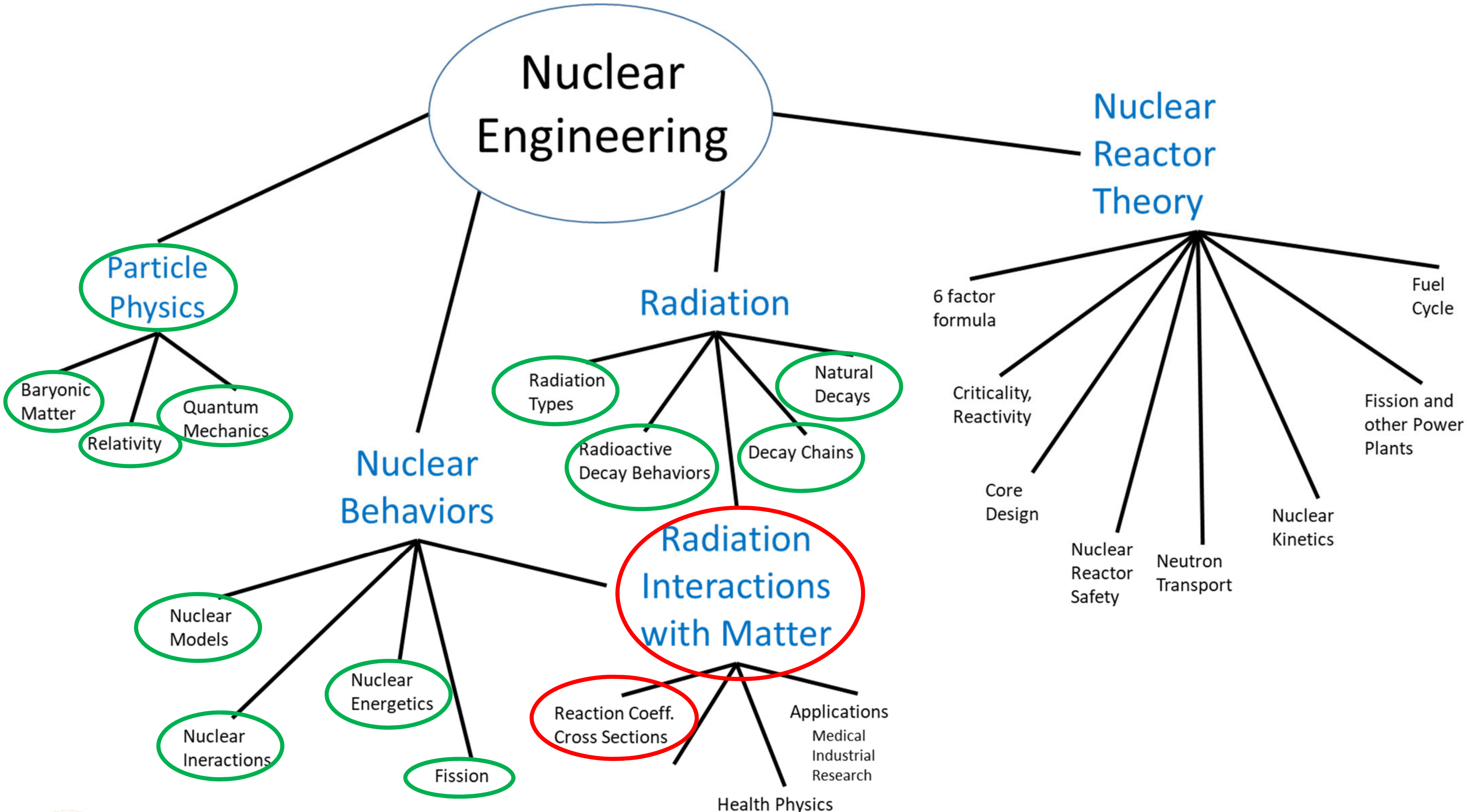
Spiritual Thought

"Most people miss opportunity when it knocks because it comes to the door dressed in overalls and looks like work."

Thomas S. Monson

(quoting Thomas Edison)

Roadmap



Nuclear Engineering
Particle Physics
Baryonic Matter
Relativity
Quantum Mechanics
Nuclear Behaviors
Nuclear Models
Nuclear Ineractions
Nuclear Energetics
Fission
Radiation
Radiation Types
Radioactive Decay Behaviors
Natural Decays
Decay Chains
Radiation Interactions with Matter
Reaction Coeff. Cross Sections
Health Physics
Applications
Medical
Industrial
Research
Nuclear Reactor Theory
6 factor formula
Criticality, Reactivity
Core Design
Nuclear Reactor Safety
Neutron Transport
Nuclear Kinetics
Fission and other Power Plants
Fuel Cycle

Objectives

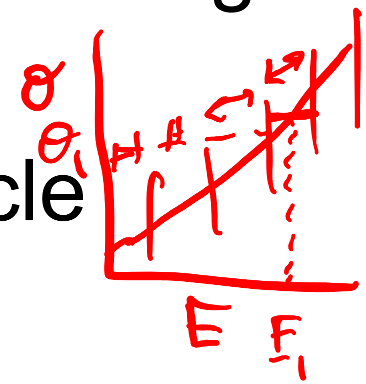
- Be able to calculate probabilities of interaction and radiation field intensities
- Understand both linear interaction coefficients and cross-sections
- Be able to calculate or find μ, σ, and Σ
- Be able to read, understand and take values from cross section libraries: plots or tables
- Know how to calculate reaction rates!!!

Microscopic cross section

- The microscopic cross section
  - Independent of atomic density
  - Based strongly and complexly on particle kinetic energy
  - Play vital roles in nuclear engineering
- Behaviors are empirical!
  - (can be conceptually explained but not always quantitatively predicted by theoretical means)
- Typical unit is barns (1 barn = 1×10^{-24} cm^2)
- 1 barn is approximate physical cross section of a uranium nucleus.

Flux and Reaction Rate

- If σ is probability of one particle interacting with one nucleus



- And Σ is the probability of one particle interacting with many nuclei
- How do we evaluate many particles with many nuclei?
- FLUX- Essentially particle density per time
- Reaction Rate (number of reactions per volume per time) $\phi_t(\sigma_1+\sigma_2)N = \hat{R}$ $R_t = \sigma_1 N_1 \phi + \sigma_2 N_2 \phi$

$$\hat{R}_i = \emptyset \sum\nolimits_i = \emptyset N \sigma_i = \emptyset \sigma_i \frac{\rho N_a}{A}$$

Example

- What is the power generation in a 1cm3 section of U^{235} fuel, assuming a thermal neutron flux of 1×10^{22} neutrons/cm^2-s?

From book: $\sigma_f = 587\,b$ $N = \frac{\rho N_a}{M}$ $M = 235\ g/mol$

Appendix C $\rho = 19.1\ g/cm^3$ $N = 4.89 \times 10^{22}\ \frac{atoms}{cm^3}$

$E_r \cong 200\ MeV$

$\Sigma_f = \sigma \cdot N =$

$Power = \Sigma_f \cdot \phi \cdot E_r = \underline{9.206\ MW}$

Mass Interaction Coefficient

- Photons – mass interaction coefficient
  - Interaction coefficient (macroscopic) divided by density
  - which depends only weakly on the properties of the medium (for photons)

$$\frac{\mu_i}{\rho} = \frac{\sigma_i N}{\rho} = \frac{N_a}{A}\sigma_i$$

- Homogeneous mixture properties can be determined from

$$\mu_i = \sum_j \mu_{i,j} = \sum_j N_j \sigma_{i,j} \qquad \frac{\mu_i}{\rho} = \sum_j w_j \left(\frac{\mu_i}{\rho}\right)_j$$

Cross sections for each interaction

$$\sigma_t = \underline{\sigma_e + \sigma_i + \sigma_\gamma + \sigma_f + \ldots}$$

total cross section

$$\sigma_a = \underline{\sigma_\gamma + \sigma_f + \sigma_\alpha + \sigma_p + \ldots}$$

absorption cross section

$$\sigma_s = \underline{\sigma_e + \sigma_i}$$

scattering cross section

$$\sigma_t = \underline{\sigma_s + \sigma_a}$$

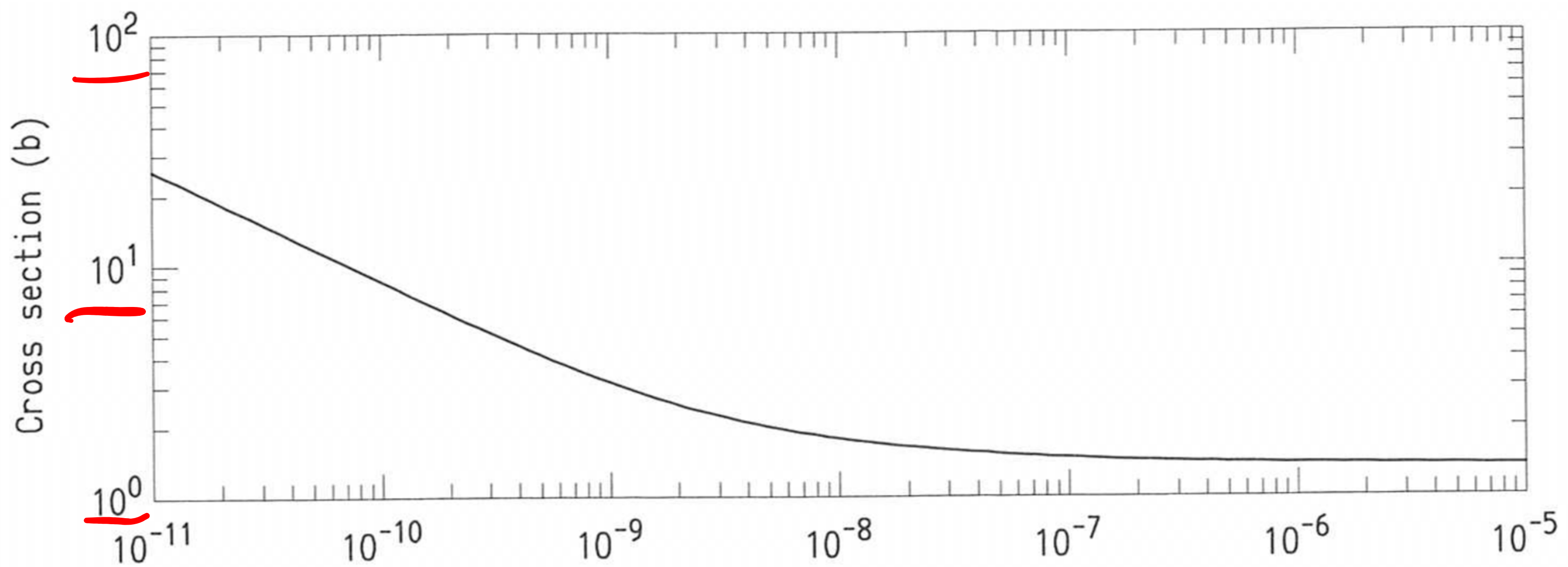
total cross section

t = total
e = elastic scattering
i = inelastic scattering
γ = radiative capture
f = fission
α= alpha (charged) particle
p = proton (charged) particle

Cross Section Trends

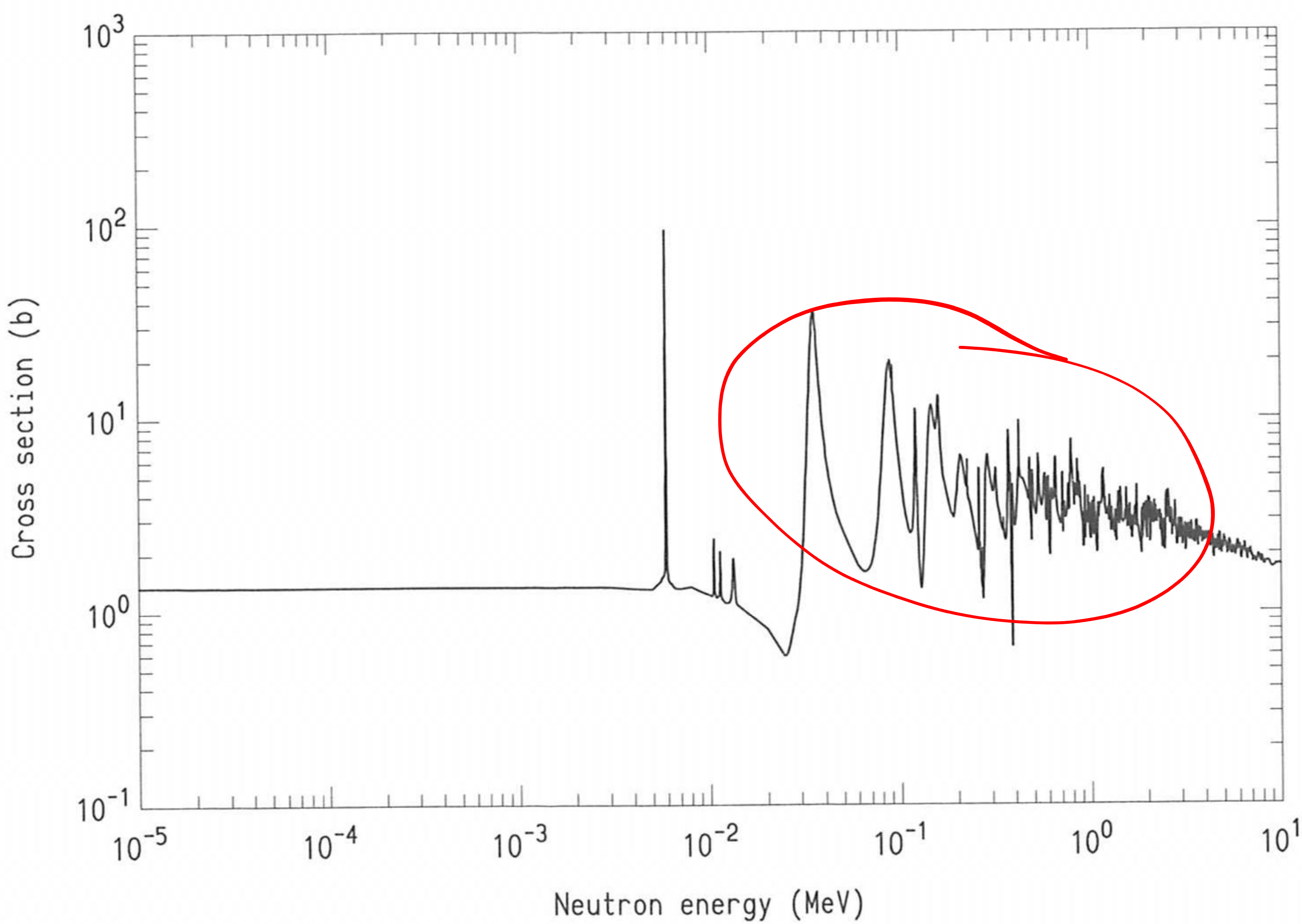
- Most Isotopes
  - Cross sections rise as neutron energy decreases.
  - Resonance regions with narrow and rapidly varying interactions that eventually are not resolvable
- Light isotopes (A < 25)
- Heavy isotopes (A > 150)
- Intermediate

Al Total Neutron Cross Section

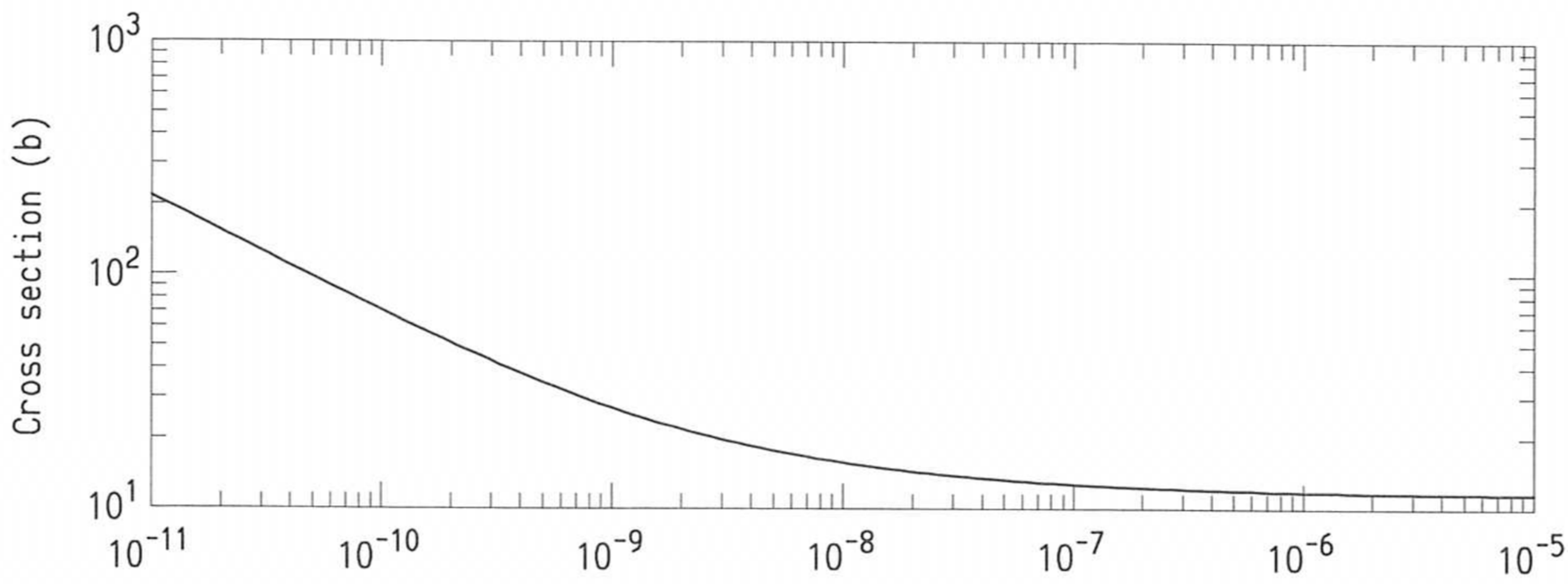


Neutron Energy (MeV)

Al Total Neutron Cross Section

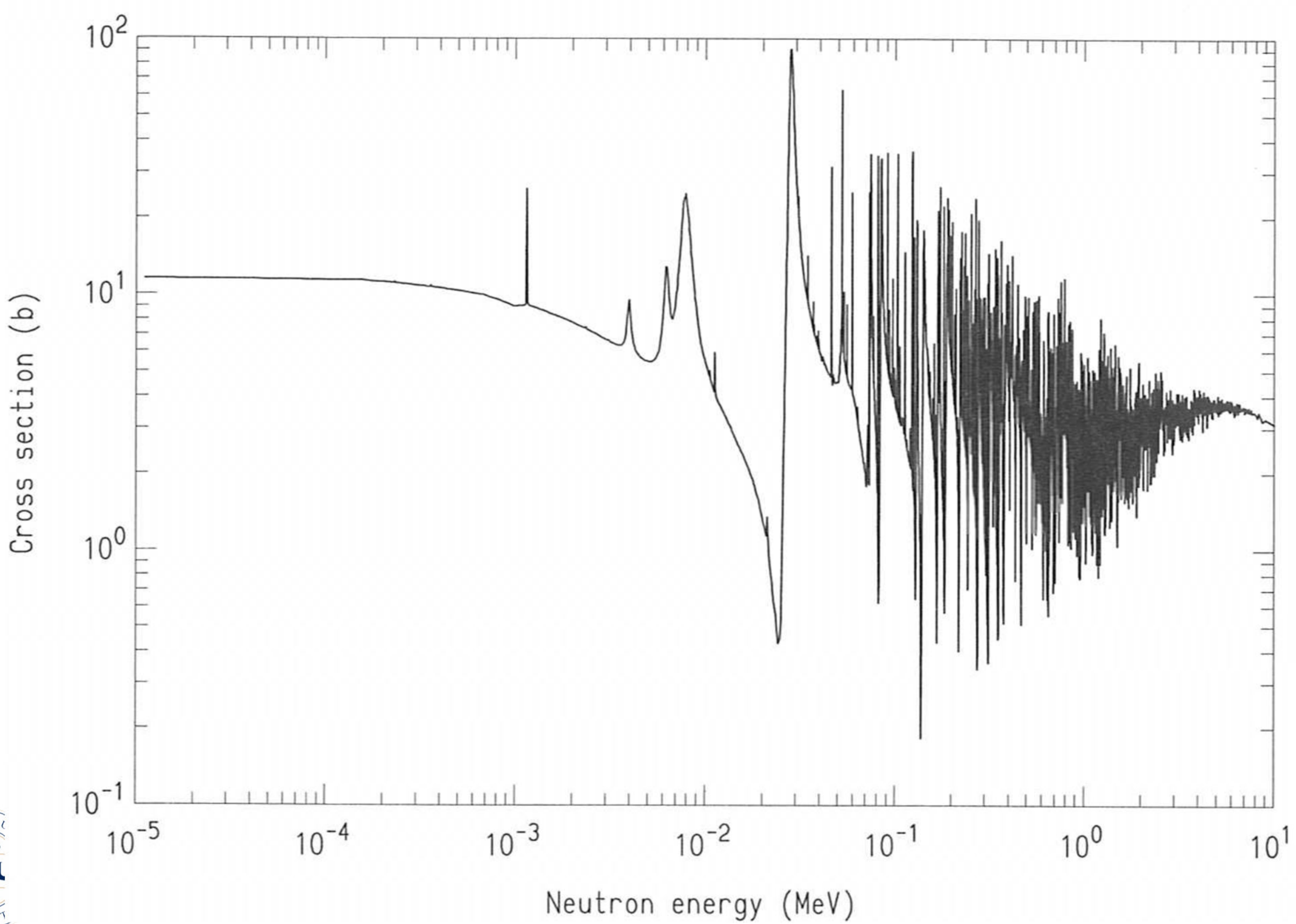


Fe total neutron cross section

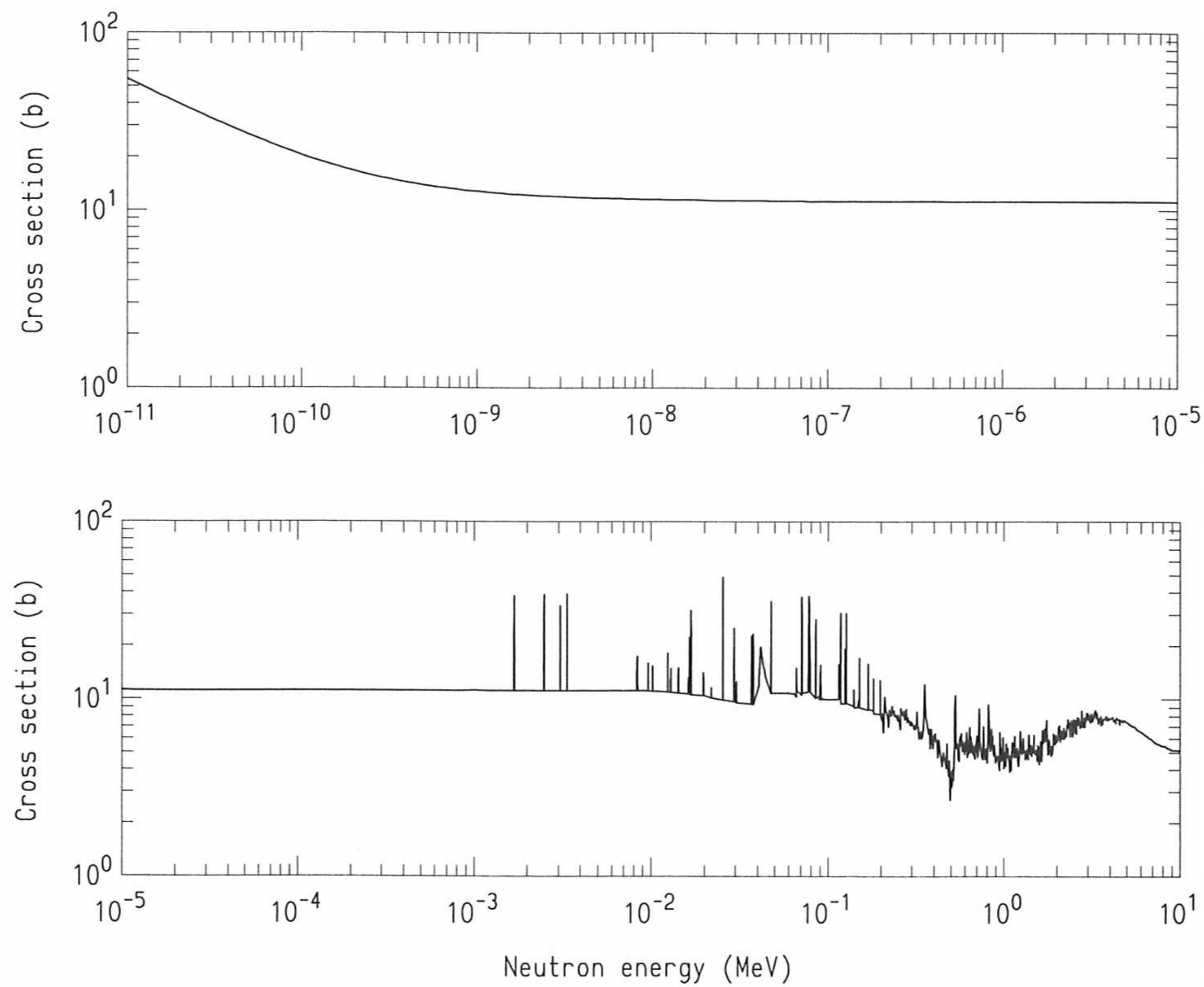


Neutron Energy (MeV)

Fe total neutron cross section

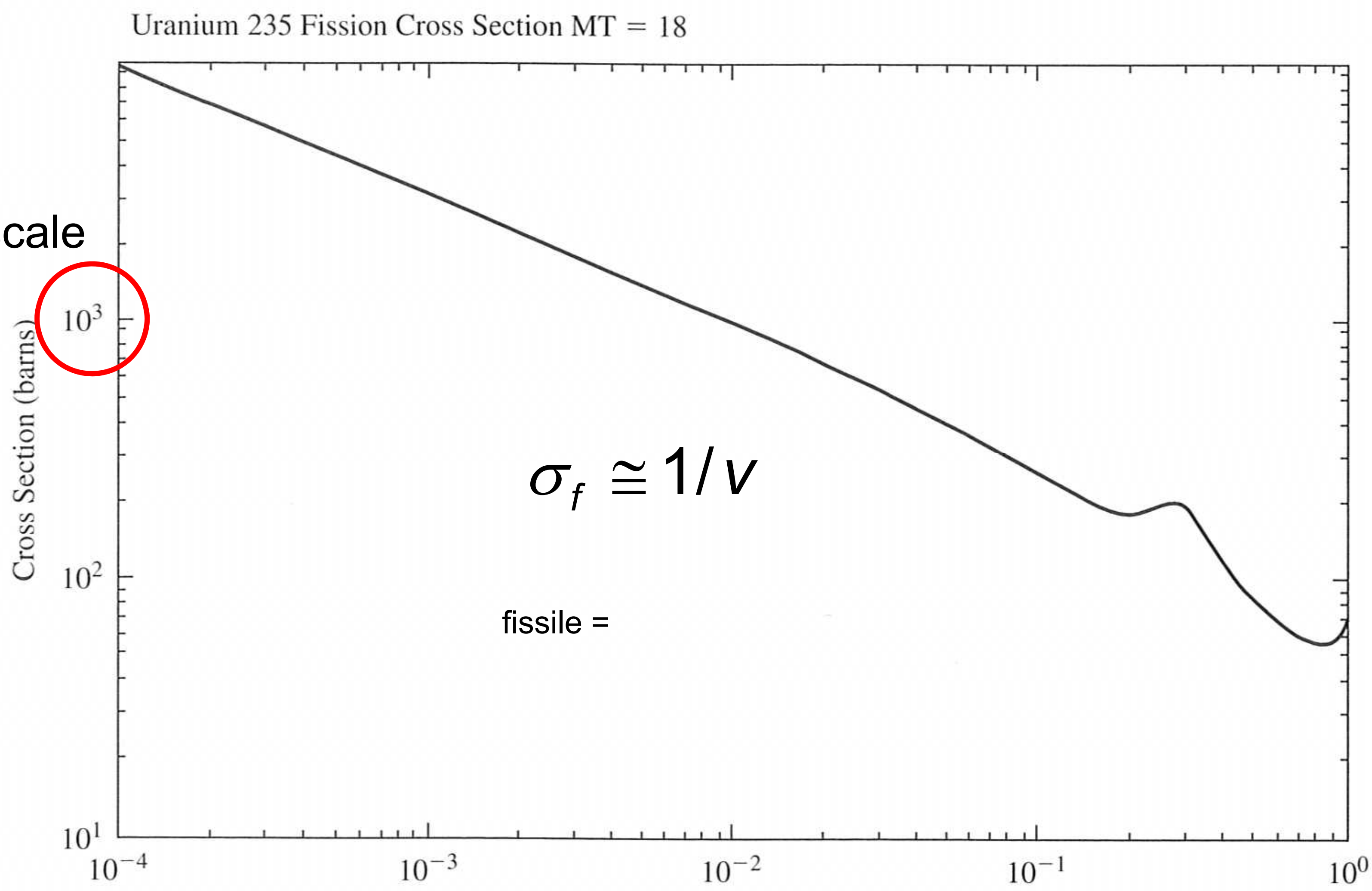


Lead Total Neutron Cross Section



Fission Cross Sections ^{235}U

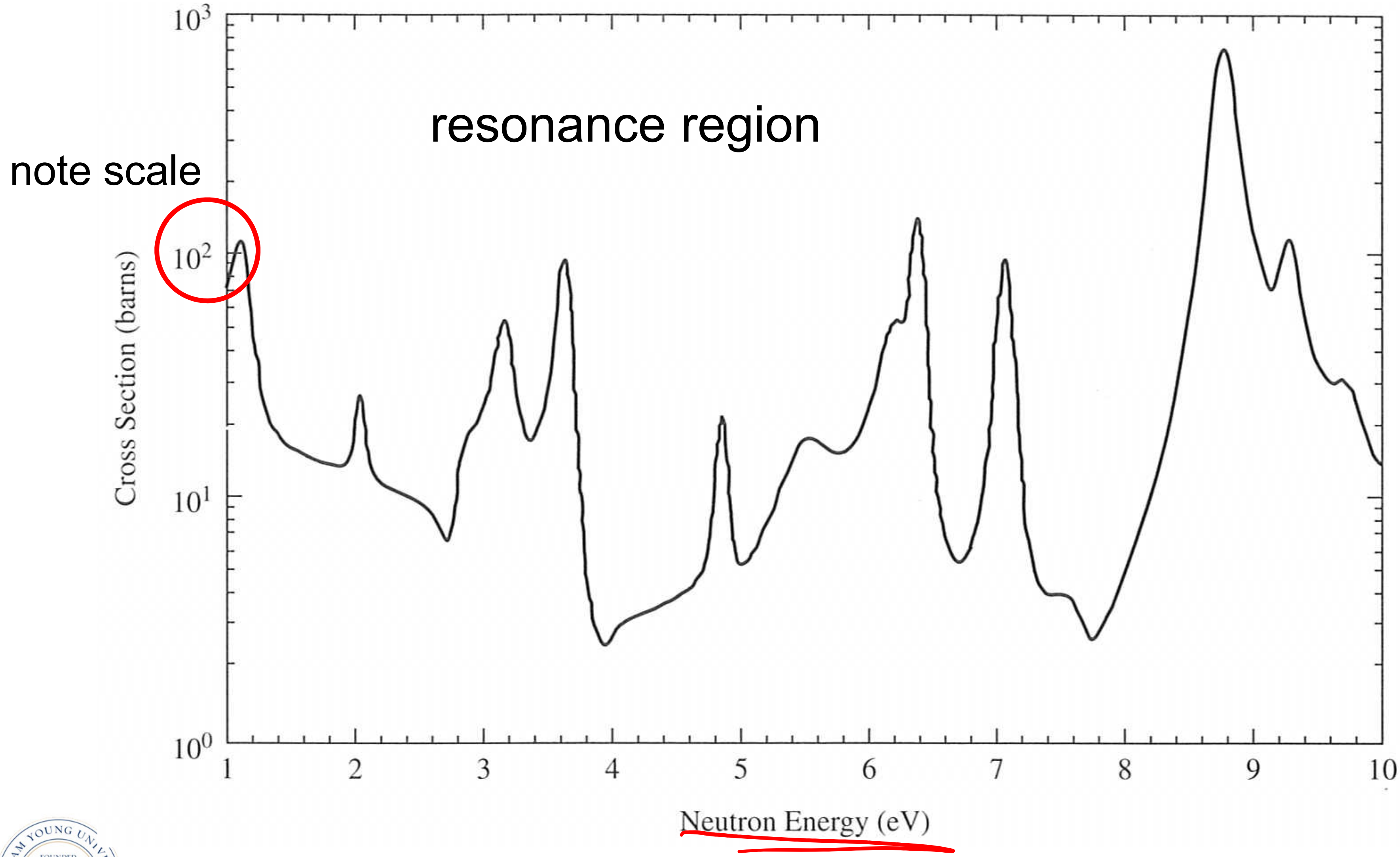
note scale



$$\sigma_f \cong 1/v$$

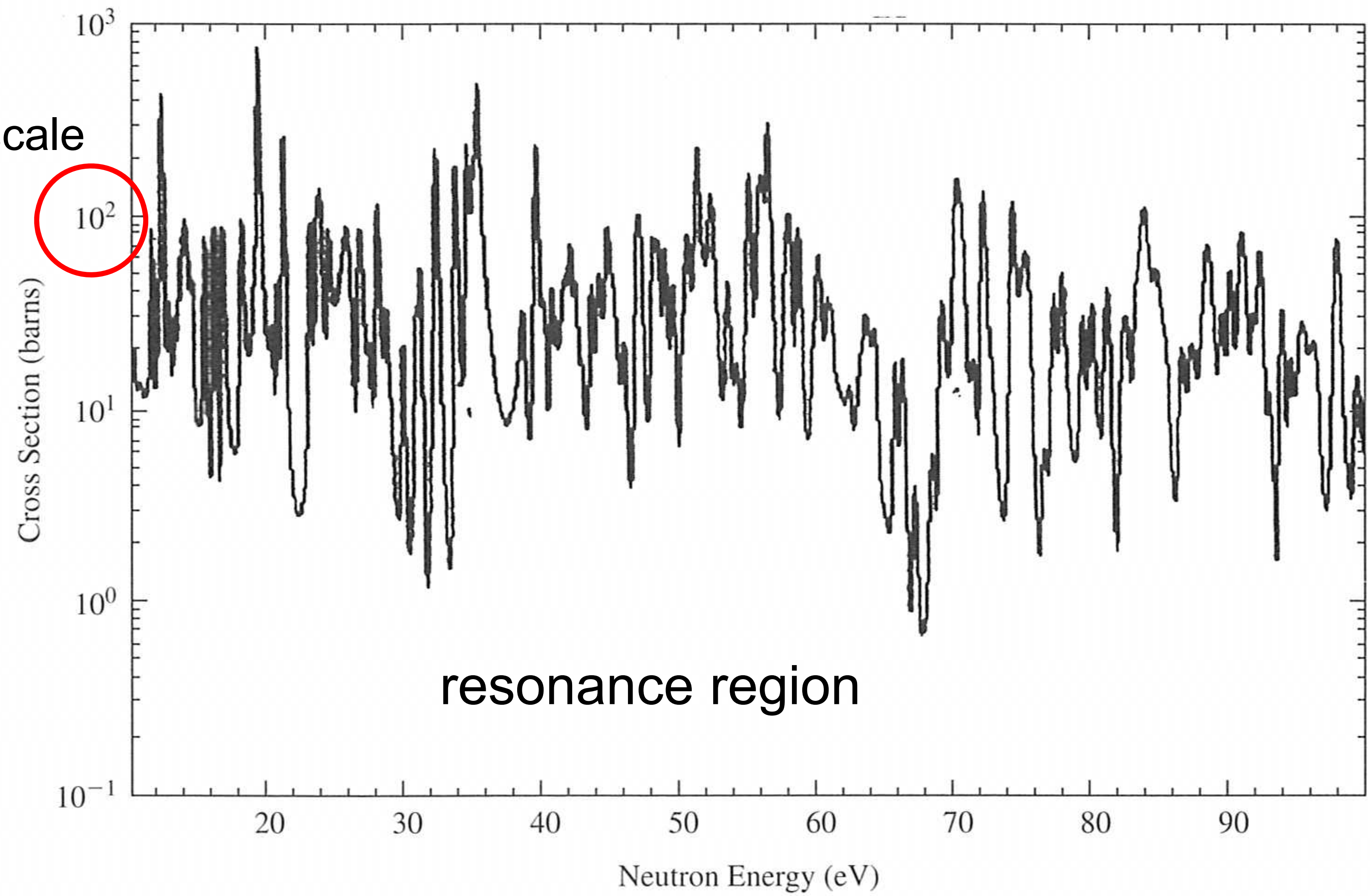
fissile =

Fission Cross Sections ^{235}U



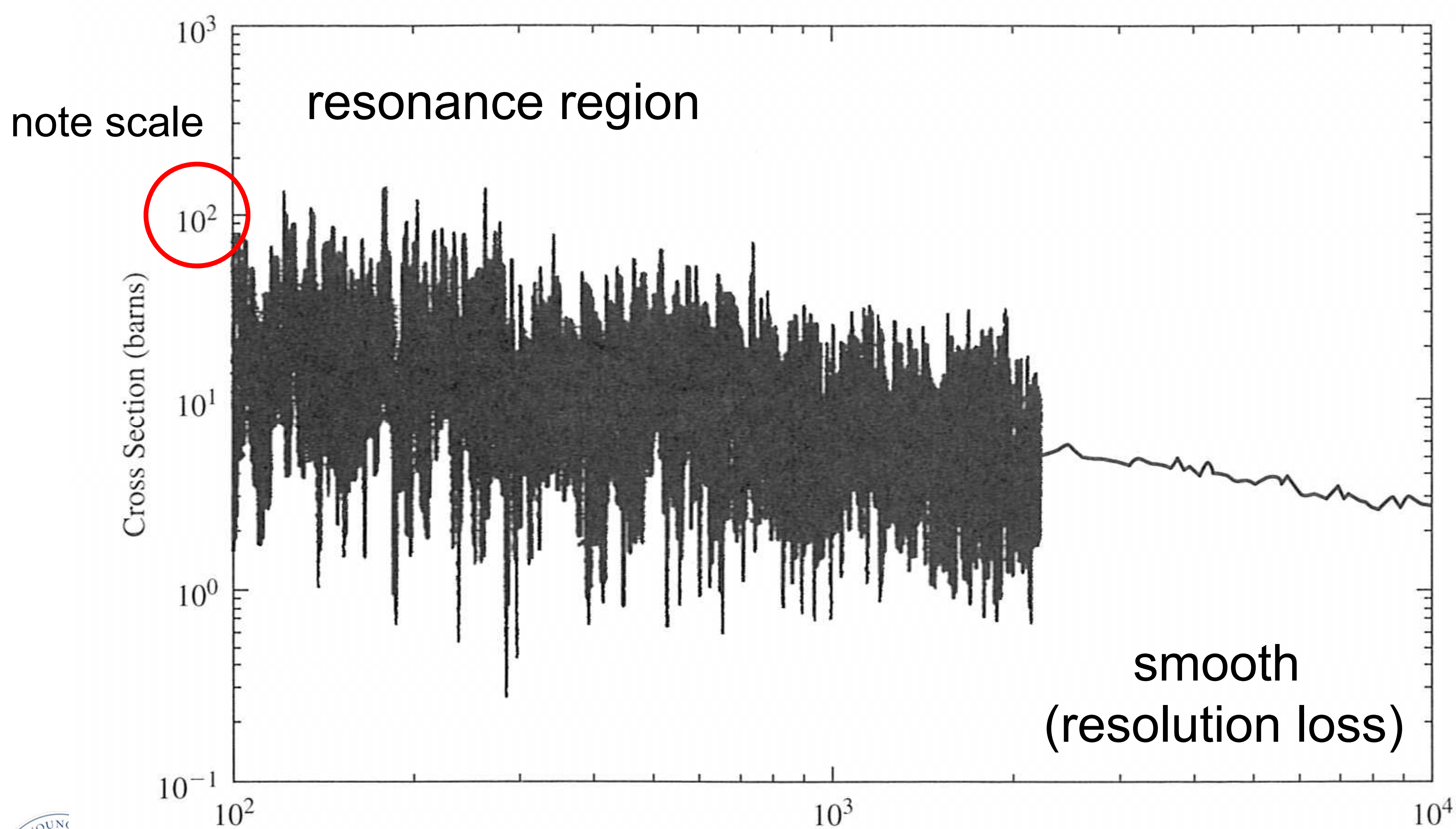
Fission Cross Sections ^{235}U

note scale



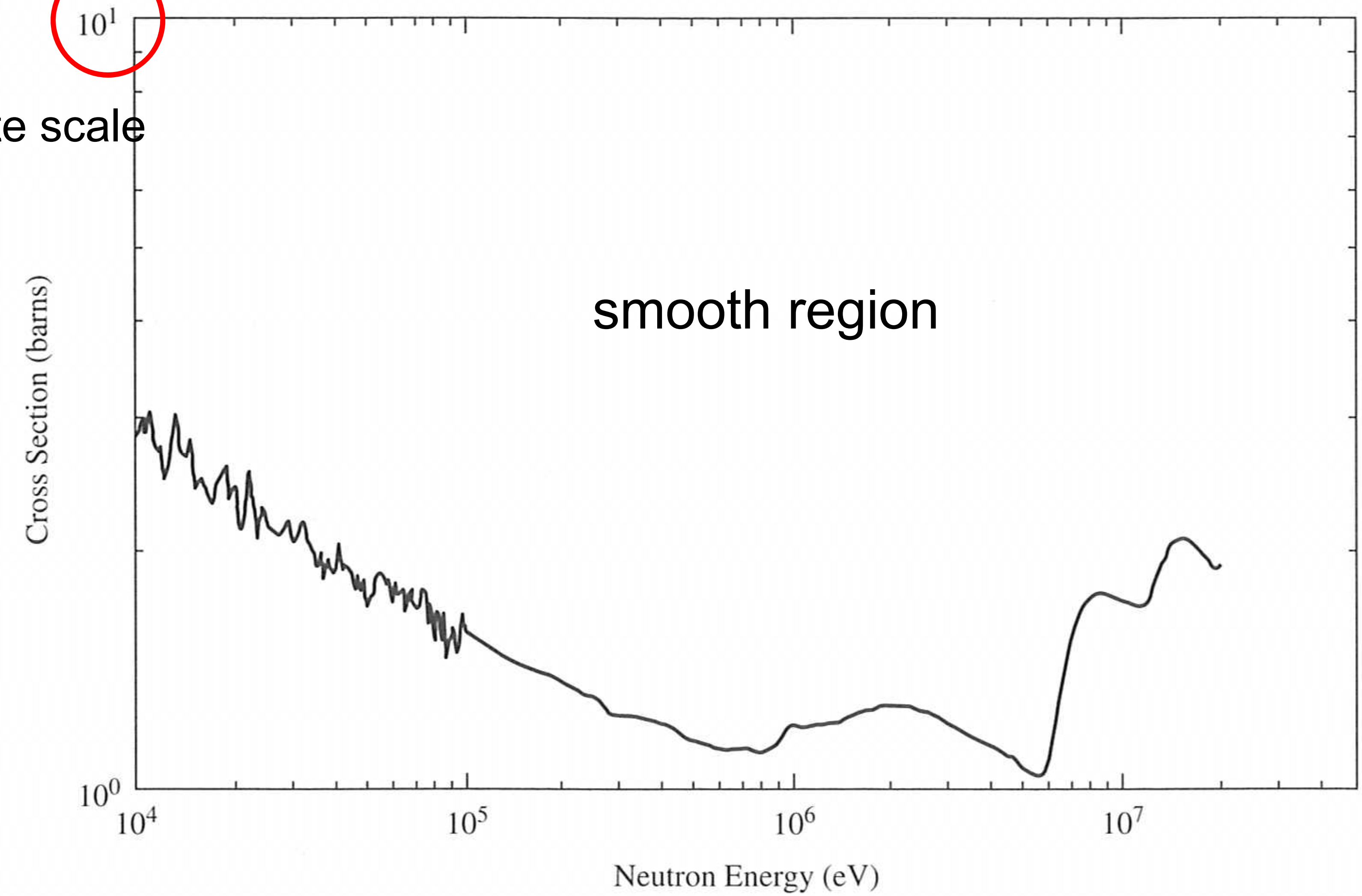
resonance region

Fission Cross Sections ^{235}U

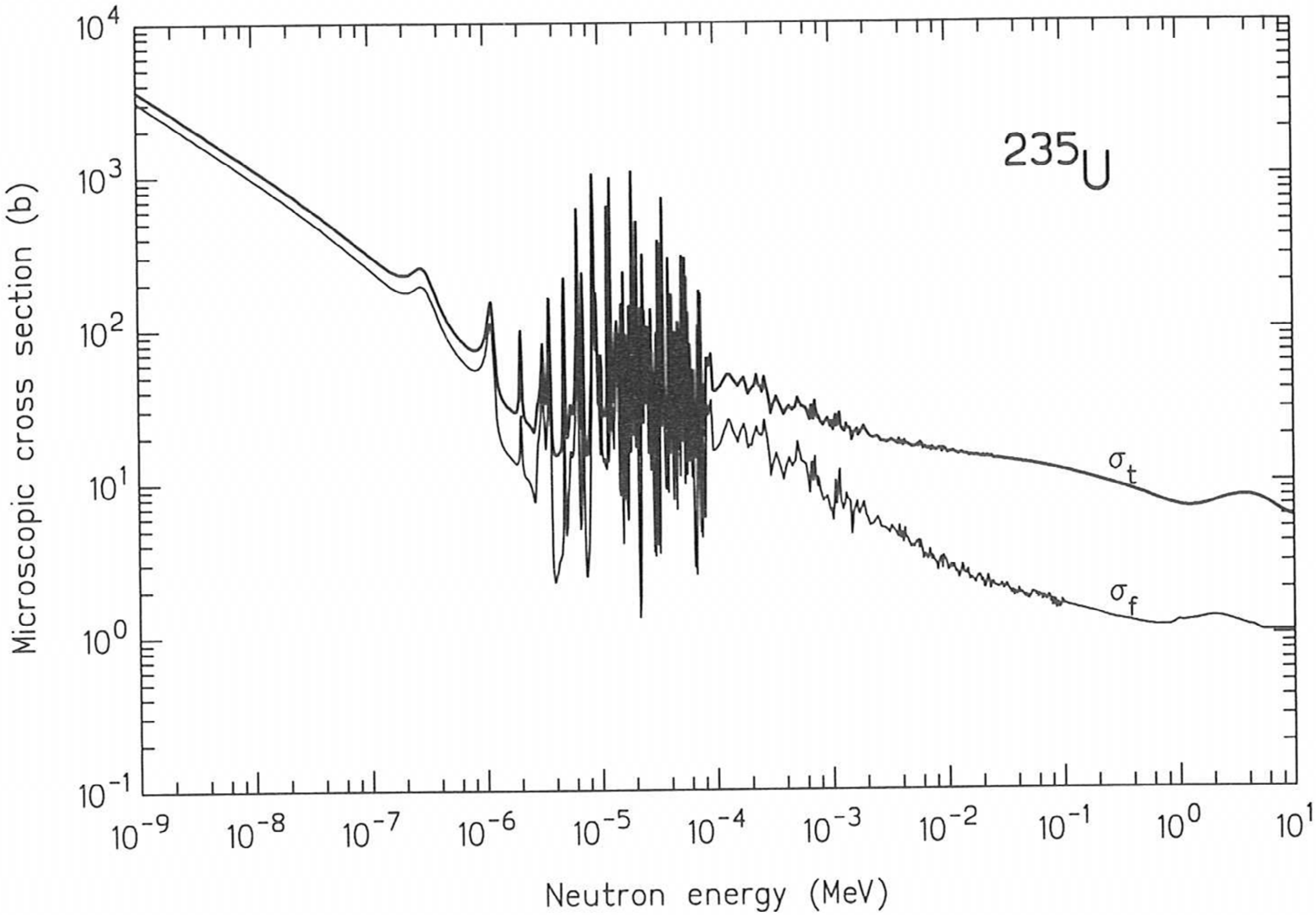


Fission Cross Sections ^{235}U

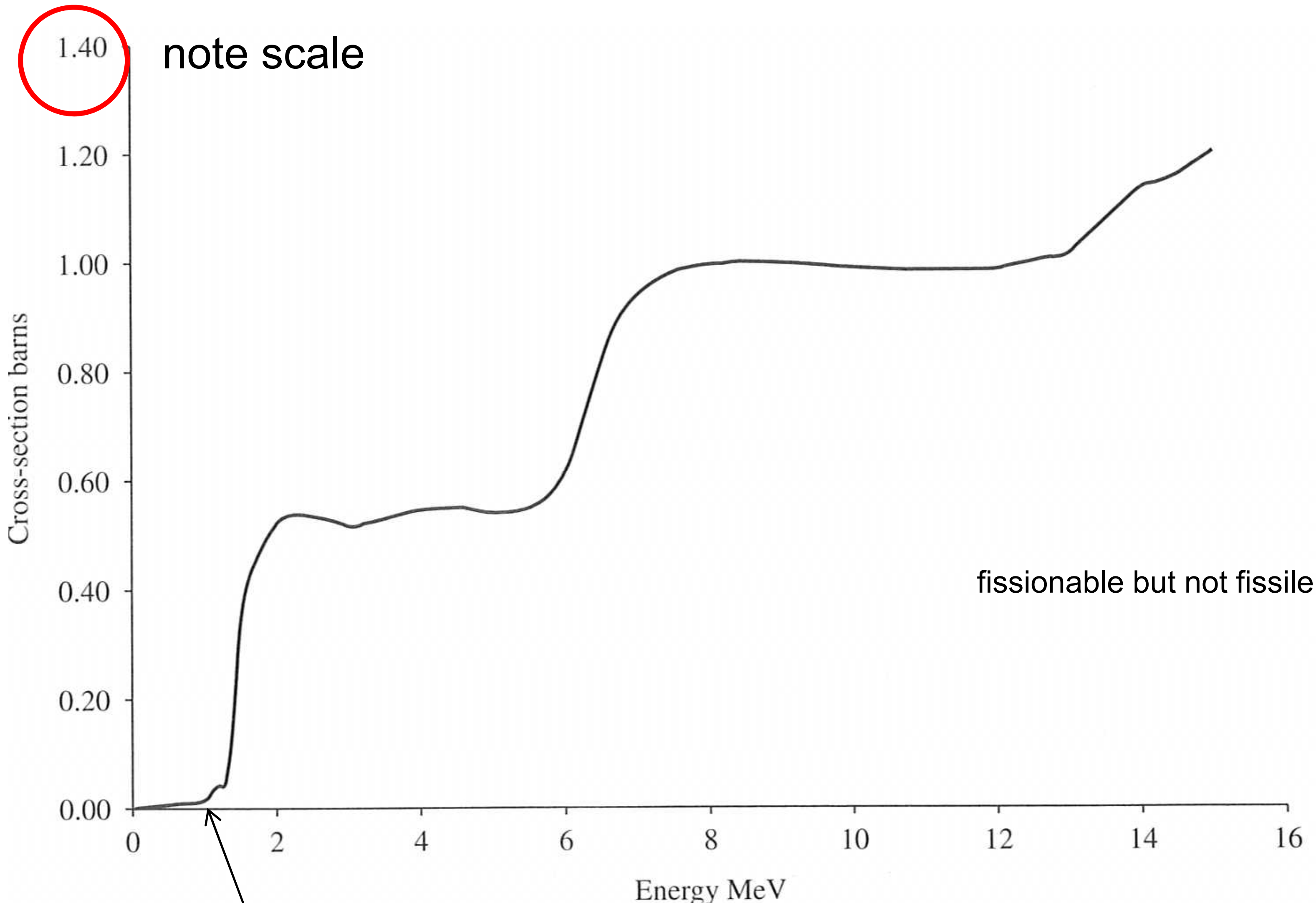
note scale



Cross section over entire range

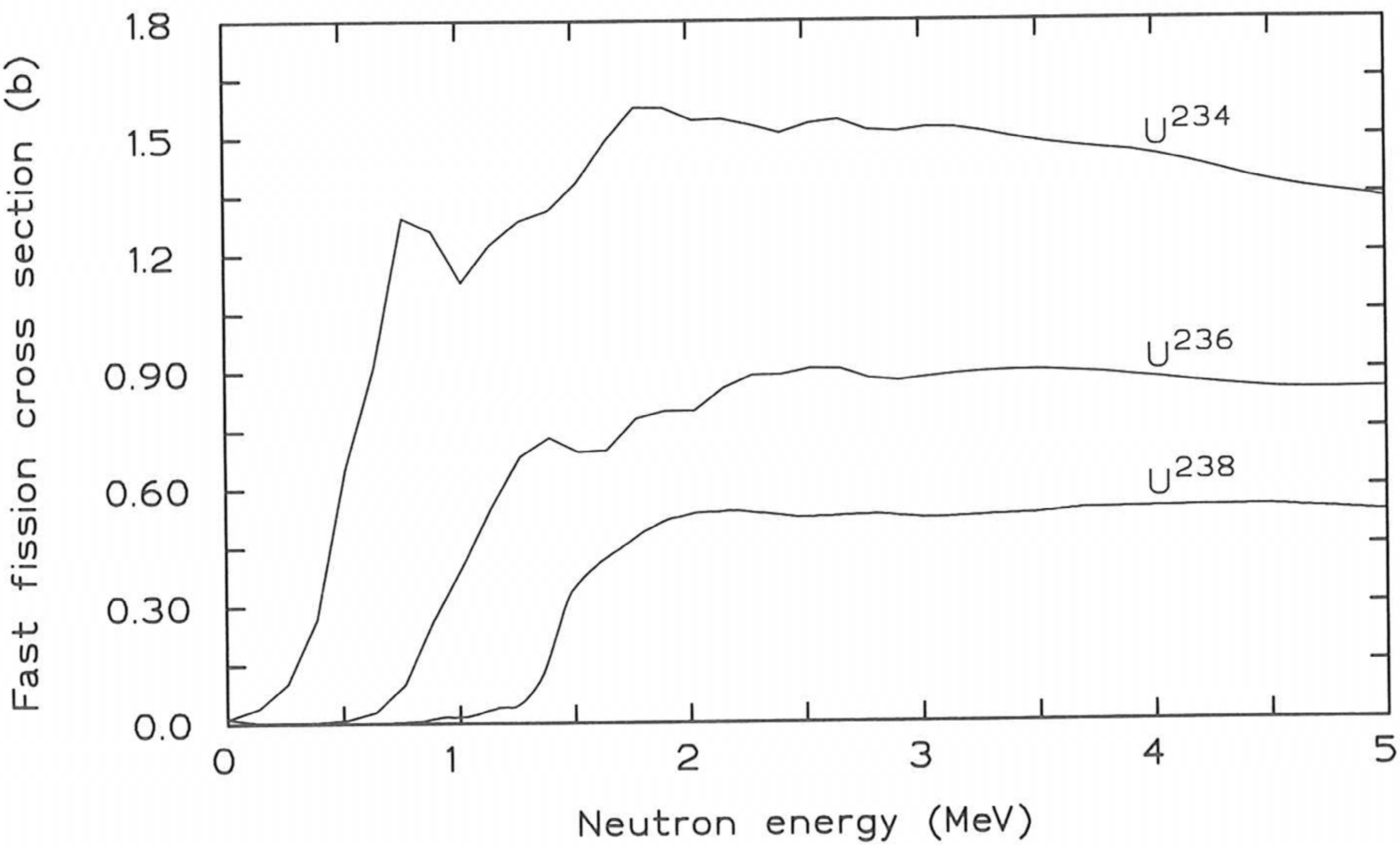


Fission Cross Section of ^{238}U



threshold energy (> resonance region energy)

Fissionable Cross Sections



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Introductory Nuclear Engineering

Exam 1 Review

$$\hat{R}_{ft} = \hat{R}_1 + \hat{R}_2 + \hat{R}_3$$

$$= \phi_1 \sigma_1 (N_1 + N_2 + N_3) + \phi_2 \sigma_2 (N_1 + N_2 + N_3) \ldots$$

Chapter 1 - Fundamentals

- Nuclear units
- Elementary particles/particle physics
- Isotopic nomenclature
- Atomic weight/number density
- Chart of nuclides
- Mass energy equivalency

Chapter 2 – Quantum Mechanics

- Special Relativity – time, length, mass changes
- Relativistic mass/momentum/energy relations
- Particle-wave duality
- Schrödinger’s wave equation
- Heisenberg’s uncertainty principle

Chapter 3 – Nuclear Models

- Nuclear energy states
- Liquid Drop Model
- Nuclear mass equation
- Shell Model
- Nuclear stability
- Binding energy/mass excess
- Modern Nucleus concepts

Chapter 4 – Nuclear Energetics

- Terminology
- Mass defect/BE
- Nuclear reactions
- Conserved quantities for various situations (not all the same!)
- **<u>****Q-Value**** (know how to calculate for ALL reactions)</u>**
  - Know how to deal with charge
  - Know how to deal with excited nuclei
  - Know how to deal with electrons/binding energy of electrons

Chapter 5 – Nuclear Decay

- Conservations
- Decay mechanisms – distinguishing features, Q values, energy/momentum balances
- ***Energy Diagrams***
- Alpha/Beta particle energy distribution
- Decay Constant
- Half-Life
- Activity

Chapter 5 – Nuclear Decay (cont)

- Parallel/Series Decay Routes
- Decay Chains
- Solutions to decay chain equations
  Secular Equilibrium
- Radionuclides in nature
- Carbon 14 dating
- Other isotopic dating methods
- Three component decays
- Isobars and most stable masses

Chapter 6 – Binary Nuclear Reactions

- Definitions
- Types of binary reactions
- Reaction Mechanisms
- Kinematics (scattering example)
- Threshold Energy
- Neutron Reactions
- Neutron Scattering/slowing
- Neutron Energy Spectrums
- Lethargy

Chapter 6 – Binary Nuclear Reactions (cont.)

- Neutron capture vs. slowing
- Fission reactions
- Emitted/recoverable fission energy
- Critical energies for fission
- Fertile vs. fissile vs. fissionable
- Fission product distribution
- Prompt vs. delayed neutrons
- Fission steps/timeline

Chapter 7 – Radiation Interactions with matter

- Linear Interaction Coefficient (micro vs. macro)
- Cross section (micro vs. macro)
- Attenuation in Material
- Derivation of material interaction
- Buildup factor
- Mass Attenuation Coefficient
- Energy dependence of cross sections
- Cross section Trends

Chapter 7 – Radiation Interactions with matter (cont.)

- Cross Section of mixture
- Total intensity/flux
- Neutron flux
- Time/space/position dependence of flux
- Fluence
- Uncollided flux transmission
- Thermal vs. fast neutrons
- Photon Interactions – types, trends, energies, cross sections
- Charged Particle interactions
- Stopping Power (collision vs. radiative)
- Range

Example 1

- The Radionuclide ^{41}Ar decays by β^- emission to an excited level of ^{41}K that is 1.293 MeV above the ground state. What is the maximum kinetic energy of the emitted β^- particle?
- What makes this the maximum energy?

Solution

Q Equation

$Q_{\beta-}$ = {M($^{41}_{18}Ar$) - [M($^{41}_{19}K$)+E*/c^2]}c^2

[40.9645008 – 40.9618259]*931.5 – 1.293 MeV

=1.199 MeV

B) Because an antineutrino is also released, which carries away some energy – this maximum is when the antineutrino has zero energy

Example 2

Assume a fuel rod has a diameter of 1 cm and a length of 5 m. Assuming an enrichment of 5% ^{235}U and a thermal flux of 2×10^{13} neutrons, what is the reaction rate in the fuel rod for:

a) scattering?

b) fission?

1) $R_s = \phi \cdot N \cdot \sigma_s \Rightarrow$ Energy? → thermal, $E = 0.025\,eV$

$R_f = \phi \cdot N \cdot \sigma_f \Rightarrow E = 0.025\,eV$

$\sigma_{s_{th}} = 2$ parts! ^{235}U & ^{238}U!

2) $\sigma_{s-235} = \underset{\text{total}}{700b} - \underset{f}{587b} - \underset{\gamma}{103b} = 10b$ $\quad \sigma_{f-235} = 587b$

$\sigma_{s-238} = 12.2b - .0000118b - 2.73b = {\sim}9.47b$ $\quad \sigma_{f-238} = .0000118b$

3) $N_{235} = \frac{\rho N_a}{A} x = \frac{(19.1 \, g/cm^3)(6.022 \times 10^{23} \, \frac{atoms}{mol})}{(235 \, g/mol)} (5\%) = \underline{2.447 \times 10^{21} \, \frac{atoms}{cm^3}}$

$N_{238} = \frac{\rho N_a}{A} x = \frac{(19.1 \, g/cm^3)(6.022 \times 10^{23} \, \frac{atoms}{mol})}{(238 \, g/mol)} (95\%) = \underline{4.591 \times 10^{22} \, \frac{atoms}{cm^3}}$

4) ⓐ $R_{s-235} + R_{s-238} = \phi N_{235} \sigma_{s-235} = (2 \times 10^{13} \frac{neut}{cm^2 \cdot s})(2.447 \times 10^{21} \frac{atoms}{cm^3})(10b)(\frac{1 \, scatter}{atom}) + \phi N_{238} \sigma_{s-238}$

$= (9.185 \times 10^{12} \, \frac{scatters}{cm^3 \cdot s})(\pi \frac{(1cm)^2}{4})(5m) = \boxed{3.607 \times 10^{15} \, \frac{scatters}{atom}}$

$\boxed{ⓑ = 1.1283 \times 10^{16} \, \frac{fissions}{s}}$

Example 3

- What is the probability of producing ^{91}Br in a fission reaction?

- Use fission product mass distribution chart:

- ~8.5%

Example 4

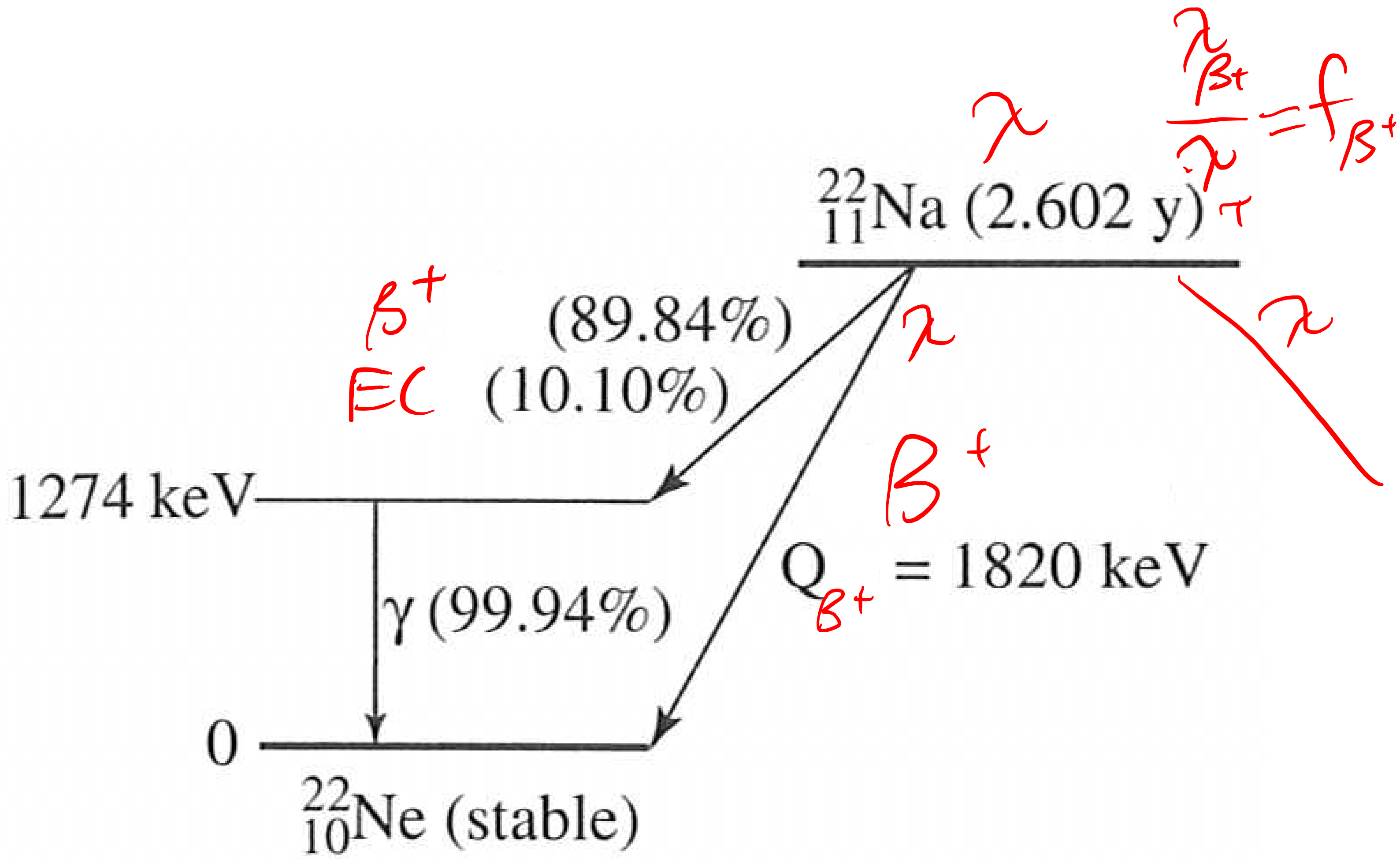
- What is the amount of thermal neutrons that are absorbed in water per cm^3 over 1 hour in a fission reactor if the thermal flux is 2.2*10^{16} neutrons/cm^2/s? (Σ_a = 0.0197 cm^{-1})

- 1.56*10^{18} absorptions per cm^3

$\phi \cdot t = \Phi$

$\hat{R}_a = \phi \cdot \Sigma_a$ or $\Phi \cdot \Sigma_a$

Example 5


λ
$\frac{\lambda_{\beta^+}}{\lambda} = f_{\beta^+}$
$^{22}_{11}$Na (2.602 y) τ
β^+ (89.84%)
EC (10.10%)
λ
λ
1274 keV
β^+
γ (99.94%)
Q_{β^+} = 1820 keV
0
$^{22}_{10}$Ne (stable)