Nuclear Power Plants (NPPs)

- **Weeks 1 & 2:** Introduction, nuclear physics basics, fission, nuclear reactors
  - Critical size, nuclear fuel cycles, NPPs *(CROCUS visit)*

- **Week 3:** Neutronics (reactor physics design) + Reactor heat transfer (fuel rod)

- **Week 4:** Reactor heat transfer (cladding, coolant) + Time-dependent reactor behaviour

- **Week 5:** Long-term reactivity changes + Principal types of nuclear power plants

- **Week 6:** Environmental aspects, nuclear safety, advanced systems
Summary, Week 1

- Nucleus: protons + neutrons (atomic number $Z$, atomic mass $A$)
- Radioactivity, specific type of nuclear reaction (spontaneous disintegration)
- Energy in a nuclear reaction: linked to binding energies (mass defects) of reactants
  - Fission, fusion: “movement” towards the large maximum of the BE-curve
- Different types of reactions: absorption (fission, capture,…), scattering
- Reaction rate = Flux $\times$ Cross-section (microscopic, macroscopic)
- Fission discovered relatively soon after discovery of neutron
- On average, $\bar{\nu}$ (2 to 3) n’s emitted per fission… chain reaction rendered possible
- Small fraction of neutrons “delayed”: crucial for reactor control
- Most of fission energy deposited in fuel (as heat)
Nuclear Fuels

- U, Th: the 2 "natural" nuclear fuels
- Only U contains *fissile* material (can be fissioned by slow neutrons)
  - $^{235}\text{U} \ldots \sim 0.7\ % \ U_{\text{nat}}$
- Rest of $U_{\text{nat}}$ ($\sim 99.3\ % \ldots \ U^{238}$), as also Th (100%… $\text{Th}^{232}$), are *fertile*
  - Give rise, via neutron capture, to the "artificial" fissile isotopes: $\text{Pu}^{239}, U^{233}$

\[
_{92}U^{238} + _0n^1 \rightarrow_{92}U^{239} \beta \rightarrow_{93}Np^{239} \beta \rightarrow_{94}Pu^{239}
\]

\[
_{90}Th^{232} + _0n^1 \rightarrow_{90}Th^{233} \beta \rightarrow_{91}Pa^{233} \beta \rightarrow_{92}U^{233}
\]

- Like $U^{235}, Pu^{239}$ and $U^{233}$ have high $\sigma_f$ values for slow (thermal) neutrons
- They are radioactive ($\alpha$): $T_{1/2}'s \sim 2.4 \times 10^4\ y (Pu^{239}), 1.6 \times 10^5\ y (U^{233})$
Neutron Cross-sections... \( \text{U}^{235} \sigma_f, \sigma_t \)
Neutron Cross-sections... $\sigma_f$ for fast n’s

- Low values for fissile nuclides
- Non-zero values for fertiles, above nuclide-specific threshold energies
  - $\sim 1$ MeV for $^{238}U$, $\sim 2$ MeV for $^{232}Th$
Comparison of Fissiles, Fertiles

- Fissiles… $\sigma_f$ goes up for low neutron energies
- Fertiles… only fissionable with neutrons of $E > 1$ MeV
- Captures “parasitic” for fissiles, useful for fertiles, e.g.

$$^{92}\text{U}^{235} + _0\text{n}^1 \rightarrow ^{92}\text{U}^{236} \text{ (product, “useless”)}$$

<table>
<thead>
<tr>
<th></th>
<th>Fissiles</th>
<th>Fertiles</th>
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<tbody>
<tr>
<td>U$^{235}$, Pu$^{239}$, U$^{233}$</td>
<td></td>
<td>U$^{238}$, Th$^{232}$</td>
</tr>
<tr>
<td>Thermal fissions</td>
<td>strong</td>
<td>zero</td>
</tr>
<tr>
<td>Fast fissions</td>
<td>weak</td>
<td>weak</td>
</tr>
<tr>
<td>Captures (th., epith.)</td>
<td>parasitic</td>
<td>useful (new fissiles produced)</td>
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$\eta_c$ : Multiplication Factor for a Nuclear Fuel (fissile, fertile)

- In general, fuel: mixture of fissile (1) and fertile (2) atoms

- $\eta_c$ : multiplication factor… number of n’s produced per n absorbed in fuel, i.e.

$$
\eta_c = \frac{\Sigma f}{\Sigma a} = \frac{\Sigma}{\Sigma_a} \cdot \left[ \frac{N_1 \cdot \sigma_{f1} + N_2 \cdot \sigma_{f2}}{N_1 \cdot \sigma_{a1} + N_2 \cdot \sigma_{a2}} \right]^{\infty} \\
= \Sigma \cdot \left( \frac{\Sigma f}{\Sigma a} \right)_e = \frac{\Sigma}{\Sigma_a} \cdot \frac{x \sigma_{f1}}{x \sigma_{a1} + (1-x) \sigma_{a2}}
$$

with $x = \frac{N_1}{N_1 + N_2}$
If fissile, fertile are isotopes of the same element... $x$ : enrichment

For $U$ of enr. $x$, and for thermal n’s ($E \approx 0.025$ eV )

\[
\begin{align*}
\sigma_f &= 582.6, \\
\sigma_t &= 681.6, \\
\sigma_r &= 2.76, \\
\bar{\nu} &= 2.42
\end{align*}
\]

\[
\begin{array}{cccccc}
x & 0.0672 & 0.015 & 0.030 & 0.20 & 0.93 & 1.0 \\
\text{(nat)} & & & & & & \\
\eta_c & 1.53 & 1.64 & 1.83 & 2.04 & 2.07 & 2.07
\end{array}
\]

$\eta_c$ increases sharply at first, then “saturates”...
Infinite Multiplying Medium  (fuel + moderator + structure)

Consider the sequence of events  

- \( f \) : fraction absorbed in the fuel  
  \[ f = \frac{\sum a_{\text{fuel}}}{\sum a_{\text{total}}} \]  ... thermal utilisation factor  
  \( \eta_c f \) : number of fission neutrons (fast) produced

- \( \varepsilon \) : factor (>1) which accounts for fast fissions in the fertile material  
  \( \eta_c f \varepsilon \) : number of fast neutrons which begin slowing down

- \( p \) : fraction which escape absorption while slowing down (fertile captures)  
  ... resonance escape probability  
  \( \eta_c f \varepsilon p \) : number of thermal neutrons

Neutron Multiplication Factor (betw. 2 consecutive generations) :  
\[ k_\infty = \eta_c f \varepsilon p \]  
(Four-factor formula)
Schematic of the Four-Factor Formula

\[ \eta_c f \epsilon \text{ produced by thermal fission} \]

\[ f \text{ absorbed in the fuel} \]

\[ \eta_c \text{ fast fissions} \]

\[ \epsilon \text{ neutrons which start slowing down} \]

\[ p \text{ (escape capture in fertile)} \]

\[ \eta_c f \epsilon p \text{ thermal neutrons} \]

\[ k_\infty = \eta_c f \epsilon p \]

... contd.
A reactor core has the following volumetric composition:

\[
U \ldots 32\%, \quad Zr \ldots 8\%, \quad Fe \ldots 2\%, \quad H_2O \ldots 58\%
\]

The enrichment of the uranium is 3%. The macroscopic thermal cross-sections (\(\Sigma_a\)) are:

\[
U \ldots 0.5553 \text{ cm}^{-1}, \quad Zr \ldots 0.0077 \text{ cm}^{-1}, \quad Fe \ldots 0.2145 \text{ cm}^{-1}, \quad H_2O \ldots 0.0176 \text{ cm}^{-1}
\]

The ratio of neutrons produced by fission in \(^{238}\text{U}\) (fast fissions), relative to those produced by \(^{235}\text{U}\) fission, is 2.4%.

Another measurement reveals that the fraction of neutrons slowing down, which are captured in the resonances of \(^{238}\text{U}\), is 31%.

What is the \(k_\infty\) value for the reactor core?
Ex. 3... Solution

\[ \eta_c \ (3\% \ enr.\ U) = 1.83 \]

\[ f = \frac{(\Sigma_a)u}{(\Sigma_a)_{tot}} = \frac{(0.5553 \cdot 0.32)}{(0.5553 \cdot 0.32) + (0.0077 \cdot 0.08) + (0.2145 \cdot 0.02) + (0.0176 \cdot 0.58)} \]

\[ f = \frac{0.1777}{0.1777 + 0.0006 + 0.0043 + 0.0102} = \frac{0.1777}{0.1928} = 0.922 \]

\[ \varepsilon = \frac{1 + 0.024}{1} = 1.024 \]

\[ p = 1 - 0.31 = 0.69 \]

\[ k_\infty = \eta_c \cdot f \cdot \varepsilon \cdot p = 1.83 \cdot 0.922 \cdot 1.024 \cdot 0.69 = 1.192 \]
Effective Multiplication Factor

- For an infinite medium: \( k_\infty = \eta_c f \varepsilon p \)
- For a reactor (finite size), one needs to consider the fraction which “escape”
  - *Non-leakage probabilities*… \( P_f \) (for fast neutrons), \( P_{th} \) (for thermal)
- Effective multiplication factor: \( k_{eff} = k_\infty P_f P_{th} = \eta_c f \varepsilon p P_f P_{th} \) (*Six-factor formula*)
- Criticality condition: \( k_{eff} = 1 \)
- \( P = P_f P_{th} \) (*non-leakage probability*) \( \approx \frac{1}{1 + B^2 M^2} \)
  (simplified result from detailed “neutronics” considerations to be made later)
  - \( B^2 \) depends inversely on system dimensions, e.g. for a spherical reactor, \( B^2 = \left( \frac{\pi}{R} \right)^2 \)
  - \( M^2 \) depends on distance the neutrons travel across the medium
    – determined largely by the moderator
Critical Size

- Considering the criticality condition: \( k_{\text{eff}} = k_\infty \cdot P = \frac{k_\infty}{1 + B^2 M^2} = 1 \)

\[ \Rightarrow B^2 = \frac{k_\infty - 1}{M^2} \]

- On the right, the expression depends purely on material characteristics
- On the left, \( B^2 \) depends only on the system dimensions

- The equation determines the *critical size*
  - Small \( B^2 \) value corresponds to large system size, and vice versa

- If \( k_\infty = 1 \), the system has to be infinitely large for criticality

- If \( k_\infty \) is large, the required (critical) size is small
  - Realisation of a high reactor power is more difficult (heat transfer limitations)
  - Advantageous in other situations, e.g. submarine, space applications
Ex. 4

For a cubic reactor of side $a$, $B^2 \approx 3.(\pi/a)^2$.

Consider such a geometry, and with the data provided for the core material in Ex. 25, calculate (a) the critical size of the reactor (take $M^2 = 60 \text{ cm}^2$), (b) the critical mass of uranium and of $U^{235}$ for the system ($\rho_U = 19 \text{ g/cm}^3$),
Ex. 4… Solution

(a) \[ B^2 = 3 \cdot \left( \frac{\pi}{a} \right)^2 = \frac{k_\infty - 1}{\text{M}^2} = \frac{1.192 - 1}{60} = \frac{0.192}{60} = 0.0032 \text{ cm}^{-2} \]

\[ \left( \frac{\pi}{a} \right)^2 = 0.001067 \rightarrow a = \frac{\pi}{\sqrt{0.001067}} = \frac{\pi}{0.0327} = 96.2 \text{ cm} \]

(b) Critical mass of U = Vol. of reactor. 32\%.19 = (a^3 \cdot 0.32.19) g = 5.4 \times 10^6 g = 5.4 \text{ t}

(c) Critical mass of U^{235} = 3\% \times 5.4 \text{ t} = 162 \text{ kg}
Fuel Cycles

- 4 “cycles” possible…
- Each one corresponds to a specific fertile / fissile “combination”
- Nearly all nuclear power plants (NPPs) today use (1)... $^{238}U / ^{235}U$

<table>
<thead>
<tr>
<th>NPP</th>
<th>Fuel</th>
<th>Moderator</th>
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<tbody>
<tr>
<td>LWR</td>
<td>3-4% enr. U</td>
<td>$H_2O$</td>
</tr>
<tr>
<td>Candu</td>
<td>Natural U</td>
<td>$D_2O$</td>
</tr>
<tr>
<td>AGR</td>
<td>2-3% enr. U</td>
<td>Graphite</td>
</tr>
</tbody>
</table>
U^{238} / U^{235} Cycle

Current-day situation in most countries (without recycle; “once-through”
$^{238}\text{U} / ^{235}\text{U}$ Cycle (with recycling)
Classification of Reactors (according to power) ...

Zero Power Reactors

- Critical assemblies, low neutron flux ($10^6$ to $10^8$ n/cm$^2$-s)
  - Power only of several watts, natural convection sufficient for cooling

- Allow detailed reactor physics (neutronics) studies to be carried out for advanced systems, e.g. PROTEUS reactor at PSI
  - Used for different experimental programmes (GCFR, HTR, LWR,...)

- Main purpose is to generate experimental (integral) data for verification and validation of reactor physics calculational codes
  - Critical size, detailed neutron balance, etc.

- Another important application: teaching and training reactor operators, e.g. CROCUS at EPFL
PROTEUS Zero-Power Research Reactor at PSI
“Cold” Power Reactors  (Medium, High Flux Research Reactors)

- Neutron flux levels typically $\sim 10^{13}$ to $10^{14}$ n/cm$^2$-s  (similar to power plants)
  - Cooling by forced convection
  - Modest coolant temperature ($<100^\circ$C, i.e. near to ambient)
  - No energy conversion (“cold” calories)
  - Technology, relatively simple
  - Wide range of applications possible

- Common type: “swimming pool” reactors, e.g. SAPHIR at PSI
  - First reactor in Switzerland (1960)
  - Upgraded to 10 MW$_{th}$ (flux $\sim 10^{14}$ n/cm$^2$-s)
  - Fuel: MTR (materials test reactor) plate-type, 20% enr. uranium (LEU)
  - Operated until 1994
Research Reactor Applications: Example of SAPHIR

- Basic physics (condensed matter) research
  - Strong neutron source, as needed for neutron scattering experiments
- Radioisotope production
  - For medicine, industry
- Radiation damage studies
- Radiochemistry applications
- Si transmutation doping (industrial use)
- Neutron radiography

View into the pool of the blue Cerenkov radiation from the SAPHIR core
Other “Cold” Power Reactor Applications

- District heating
- Desalination of sea water
- Military applications

R: Reactor, P: Pump, H.E.: Heat Exchanger
“Hot” Power Reactors: Nuclear Power Plants (NPPs)

- Almost all commercial production of nuclear energy today, in the form of electricity
  - High-quality utilisation

- Demanding technology needed for energy conversion
  - Coolant temperature of at least ~ 300°C
  - Thermal efficiency ~ 30 to 35% for LWRs, 40 to 45% for HTRs
    - Typical thermodynamic cycle shown below
    - Other, more complicated schemes possible, e.g. co-generation of district heat

![Thermodynamic Cycle Diagram]

- R: Reactor
- P: Pump
- SG: Steam generator
- HP: High pressure turbine stage
- LP: Low pressure turbine stage
- C: Condenser
- A: Alternator
Nuclear Power Plants

NPP Types

- Current nuclear energy scene dominated by Light Water Reactors (LWRs: 85% of all NPPs)
  - 65% PWRs, 20% BWRs

- Others:
  - Gas-Cooled Reactors (GCRs: 9%)
  - Canadian Heavy Water Reactors (CANDUs: 5%)
Characteristic NPP Components

- Control rods
- Reflector
- Thermal shield
- Coolant outlet
- Blanket
- Core
- Containment
- Biological shield
- Vessel
- Coolant inlet
Reactor Core: Regular Lattice of “Unit Cells”

- Multiplying medium, in practice, is heterogeneous
  - Associated to each “unit cell” volume of fuel, certain volumes (in given geometry) of structural material (cladding) and moderator
  - Simple unit cell for LWR lattice
  - For CANDU, with cluster of fuel rods, effectively a “supercell”
Teaching Reactor CROCUS at EPFL
CROCUS Views

Horizontal cut

Vertical cut
Brief Description of CROCUS

CROCUS is a zero-power reactor, with a maximum thermal power of 100 W. Its main utilisation purpose is reactor physics related teaching. An important advantage of its low nominal power operation is that it is possible to approach the reactor core without any risk, once the radioactivity level is sufficiently low.

The core is in the form of a cylinder of about 60 cm diameter and 1 m height. It consists of 336 rods of 1.8% enriched UO$_2$, surrounded by 174 U-metal rods of 0.95% enrichment. Thus, effectively, the reactor’s “fuel elements” are simply single rods of uranium (clad in aluminium), maintained in their vertical position with the desired regular spacing (lattice pitch) by two horizontal grid plates, 1 m apart. The reactor core is situated at the centre of a cylindrical reactor tank, 1.3 m in diameter. Demineralised H$_2$O serves as the moderator, as also the reflector*.

*moderator zone surrounding the core to reduce the neutron leakage
Summary, Week 2

- Nuclear fuels: U, Th... only $^{235}\text{U}$ fissile; $^{238}\text{U}, ^{232}\text{Th}$ fertile (yield fissile $^{239}\text{Pu}, ^{233}\text{U}$)
- Distinct characteristics of fissile, fertile nuclides
- Most fissions at low energies in a “thermal” reactor
- $\eta_c$: neutron multiplication factor for fuel (fissile/fertile mixture)
- Four-factor formula for $k_\infty$ ($\eta_c, f, \varepsilon, p$)
- $k_{\text{eff}}$, non-leakage probability, critical size
- Nuclear fuel cycles
- Classification of reactors according to power
- Visit to CROCUS