Energy from nuclear fission: basics

M. Ripani
INFN Genova, Italy

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Why can fission produce energy?

Nuclear mass and nuclear binding energy

\[ M(Z,A) = ZM_p + (A - Z)M_n + B(Z,A) \]

\[ B(Z,A) < 0 \text{!!! i.e. a nucleus weighs less than the sum of proton and neutron masses} \]

\[ \varepsilon = \frac{|B(Z, A)|}{A} \]
How can fission produce energy?

\[ M(Z,A) \rightarrow M(Z_1, A_1) + M(Z_2, A_2); \quad Z = Z_1 + Z_2; \quad A = A_1 + A_2 \]

Energy balance (Q-value)

\[ Q_{fiss} = M(Z, A) - M(Z_1, A_1) - M(Z_2, A_2) = B(Z, A) - B(Z_1, A_1) - B(Z_2, A_2) = -\varepsilon A + \varepsilon_1 A_1 + \varepsilon_2 A_2 = -\varepsilon A + \bar{\varepsilon} A = (\bar{\varepsilon} - \varepsilon) A \]

\[ \bar{\varepsilon} = \frac{\varepsilon_1 A_1 + \varepsilon_2 A_2}{A_1 + A_2} \]

\[ Q_{fiss} > 0 \]

\[ Q_{fiss} < 0 \]

\[ \varepsilon = |B|/A \]
How to break a nucleus

Potential energy

Fragment kinetic energy

Quantum tunneling: it makes spontaneous fission possible, but unlikely

Coulomb barrier

- distance

- proton
- neutron
How to break a nucleus

I need to supply energy to overcome the barrier. If I can do this, I will have produced nuclear energy.

This process is called **Nuclear Fission**.
A special case, Uranium 235

By exploiting the energy released when a neutron is captured, fission is made much easier and can happen with high probability.

The nucleus breaks up but also a few (2-3) neutrons are emitted.

- 92 protons
- 143 neutrons
The chain reaction

Natural Uranium, from mines
→ 0.7 % U 235 + 99.3 % U 238

U-235 breaks up even when a neutron is captured at rest, while U-238 needs an energetic (fast) neutron, hence the reaction is less easy to occur → 235 is «fissile», 238 isn’t

Fragments are moving → kinetic energy → transferred to atoms → heat
Other neutron absorption processes yielding energy

\[ \sigma(\text{thermal neutrons}) \approx 5330 \text{ b} \text{ (barn, } 1 \text{ b}=10^{-24} \text{ cm}^2, \sigma \text{ is proportional to the reaction probability, see later)} \]

But no chain reaction

3He neutron detector

neutron detector based on a LiF film

available E 4.78 MeV
Another very relevant reaction mechanism is **neutron capture**

→ for heavy nuclei, addition of one more neutron can provide several MeV from binding energy
→ capture is **followed by gamma emission** (radiative capture) or **fission**

By quantum mechanical arguments, it is possible to show that at low energies (if the energy gained from the neutron capture is sufficient to produce the phenomenon of interest)

→ **cross section follows a 1/v law**, with v being the relative speed (essentially the n speed)

Example: Plutonium production from Uranium

\[ n + ^{238}\text{U} \rightarrow ^{239}\text{U} + \gamma \rightarrow ^{239}\text{Np} + \beta + \text{anti-}\nu \rightarrow ^{239}\text{Pu} + \beta + \text{anti-}\nu \]
Amount of energy and reaction products

When a uranium nucleus fissions into two daughter nuclei fragments, about 0.1% of uranium mass appears as fission energy of ~200 MeV (E=Mc²)

→ much bigger than any other exoenergetic nuclear reaction (in absolute terms)

- ~170 MeV appears as the kinetic energy of the daughter nuclei,
  - which fly apart at about 3% of the speed of light

- an average of 2.5 prompt neutrons are emitted, with a mean kinetic energy per neutron of ~2 MeV (total of 4.8 MeV)

- the average number of neutrons emitted is called V (order of 2-3)

- ~25 MeV are released in form of prompt gamma ray photons and fission product β decay

Chemical reactions vs nuclear fission

C+O₂ = CO₂ + Q (Q = 3.6 eV),
CH₄ + 2O₂ = CO₂ + 2H₂O + Q (Q = 9.22 eV),
⁵²⁵U+n = F₁+F₂+ ν n+Q (Q ~ 200 MeV)

⇒ Fission gives between 20 and 50 million times more energy
Physics: nuclear cross sections

Cross section: quantity that characterizes a nuclear reaction (elastic, inelastic scattering, etc.) connected to the range of the involved forces; effective area of a nuclear target

Here we will consider the total cross section, defined as follows:

\[ \sigma = \frac{dN_{\text{reac}}}{dSdt} \]

Given a flux \( \frac{dN_{\text{in}}}{dSdt} \)

counting number of incident particles per unit surface and unit time on a single nucleus (target)

and given an interaction rate \( \frac{dN_{\text{reac}}}{dt} \)

counting number of interacting particles (scattered or absorbed projectiles) per unit time, then

\[ \sigma = \frac{dN_{\text{reac}}}{dN_{\text{in}}} \]

\( \sigma \rightarrow \) physical dimensions of a surface
Nuclear cross sections

**Macroscopic target** comprising several nuclei with **density** $\rho$ (es. gr/cm³) and thickness $x$, struck by a particle beam of intensity $I$ (particles/sec) →

$$R = \frac{dN_{\text{reac}}}{dt} = I \frac{\rho x}{A} N_A \sigma$$

where $A$ is the target atomic weight (es. in gr.) and $N_A$ is the Avogadro number.

$\frac{\rho}{A} N_A$ is the **number density of nuclei** in the target (i.e. number of nuclei per unit volume).

This is all valid for a **small thickness** $x$.

For a target of **arbitrary thickness**, first divide it in thin slices of thickness $dx$ →

$$dR = \frac{dN_{\text{reac}}}{dt} = I(x) \frac{\rho}{A} N_A \sigma \, dx$$

$$dI = -I(x) \frac{\rho}{A} N_A \sigma \, dx$$

$$I(x) = I(0) \exp(-\frac{\rho}{A} N_A \sigma \, x)$$

$$\sum \equiv \frac{\rho}{A} N_A \sigma$$

Macroscopic cross section = probability of interaction per unit length

$$\frac{1}{\sum} = \text{Mean free path}$$

$$\sum \nu = \text{Frequency with which reactions occur, } \nu = \text{projectile speed}$$
Neutron density and flux

Neutron density \( n(r, E, t) \) [cm\(^{-3}\)] is the expected number of neutrons with energy between \( E \) and \( E + dE \), in the volume \( d^3r \) about \( r \), at a time \( t \).

Reaction density \( R(r, E, t) \) is the number of reactions in the volume \( d^3r \) about \( r \), at a time \( t \), initiated by neutrons with energy between \( E \) and \( E + dE = n(r, E, t) \sum v \).

We give a special name to the quantity \( n(r, E, t)v \). It is called the neutron “flux” \( \phi(r, E, t) \equiv n(r, E, t)v \) [cm\(^{-2}\) s\(^{-1}\)].

**Reaction density** \( \equiv \) **number of reactions per unit volume** \( \equiv R(r, E, t) = \Sigma \phi \)

Suppose you’ve got a reactor with 1 GW thermal power = 10\(^9\) Joule/sec
Assume each fission releases order of 200 MeV energy = 3.2x10\(^{-11}\) Joule
\( \rightarrow \) In the reactor the fission rate is about 3x10\(^{19}\) fissions/sec
\( \rightarrow \) Almost 10\(^{20}\) neutrons/sec emitted, about 2x10\(^{20}\) neutrinos/sec
\( \rightarrow \) \( \phi \sim 10^{14} \) neutrons cm\(^{-2}\) s\(^{-1}\)
Nuclear cross sections

Since the nuclear radius is roughly $10^{-12}$ cm, the geometrical cross sectional area of the nucleus is roughly $10^{-24}$ cm$^2 = 1$ barn

Hence we might expect that nuclear cross sections are of the order of $10^{-24}$ cm$^2 \equiv 1$ barn

However, quantum mechanical effects can make nuclear cross sections a lot bigger...

U-235

- U-235 can undergo fission at all neutron energies
- The process is particularly efficient for slow neutrons

Resonances:
\[ CM \text{ energy of neutron+nucleus system } + \text{ binding energy of the captured neutron} \]
\[ \text{match one of the energy levels in the compound nucleus} \]
U-238 can undergo fission at high neutron energy > 1 MeV
An interesting case: Thorium, an abundant material on Earth
Fissile, fissionable, fertile isotopes

- Heavy nuclei with a high fission cross section at low (thermal) neutron energies are called **fissile** (e.g. $^{233}\text{U}$, $^{235}\text{U}$, $^{239}\text{Pu}$, ...)

- Those with a non-zero fission cross section only at higher neutron energies are called **fissionable** (e.g. $^{238}\text{U}$, ...)

- Those that can produce a fissile isotope via neutron radiative capture and $\beta$ decay are called **fertile**, i.e. they can be used to **produce fuel** (e.g. $^{238}\text{U}$, ...)

- $\text{n} + ^{238}\text{U} \rightarrow ^{239}\text{U} + \gamma \rightarrow ^{239}\text{Np} + \beta + \text{anti-}\nu \rightarrow ^{239}\text{Pu} + \beta + \text{anti-}\nu$
  
  - **fissile**

- $\text{n} + ^{232}\text{Th} \rightarrow ^{233}\text{Th} + \gamma \rightarrow ^{233}\text{Pa} + \beta + \text{anti-}\nu \rightarrow ^{233}\text{U} + \beta + \text{anti-}\nu$

✓ Natural Uranium $\rightarrow$ 0.7 % $^{235}\text{U}$ + 99.3 % $^{238}\text{U}$ $\rightarrow$ **most reactors need 3-5 % $^{235}\text{U}$** $\rightarrow$ “enrichment” process

✓ Plutonium production is also called “breeding”

✓ Under certain conditions, a reactor can produce more Pu than it consumes $\rightarrow$ it is called “breeder”
Cross section (b)

**U-235, U-233, Pu-239**

U-233 and Pu-239 are more efficient than U-235 for high neutron energies ("fast" neutrons).
Burning-breeding-burning: the Uranium-Plutonium cycle and the long lifetime radioactive waste production (1 GW_e LWR)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Decay Constant</th>
<th>Production Rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>244, 245Cm</td>
<td>1.5 Kg/yr</td>
<td></td>
</tr>
<tr>
<td>241Am</td>
<td>11.6 Kg/yr</td>
<td></td>
</tr>
<tr>
<td>243Am</td>
<td>4.8 Kg/yr</td>
<td></td>
</tr>
<tr>
<td>239Pu</td>
<td>125 Kg/yr</td>
<td></td>
</tr>
<tr>
<td>237Np</td>
<td>16 Kg/yr</td>
<td></td>
</tr>
<tr>
<td>LLFP</td>
<td>76.2 Kg/yr</td>
<td></td>
</tr>
</tbody>
</table>

**LLFP** = Long Life Fission Products

**Transuranics** = Minor Actinides + Pu
Transuranics and waste

- Operating a fission reactor means that long-lived radioactive nuclides are produced.
- Many transuranics are long-lived and undergo $\alpha$ decay (followed by $\gamma$ emissions).
- Such radioactive nuclei can be dangerous for the environment and the human health (due to direct exposure in the case of $\gamma$’s and due to ingestion or inhalation in the case of $\alpha$ and $\beta$ decay).

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Half-life $T_{1/2}$ (years)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-239</td>
<td>24,000</td>
</tr>
<tr>
<td>Pu-242</td>
<td>$3.7 \times 10^5$</td>
</tr>
<tr>
<td>Am-241</td>
<td>433</td>
</tr>
</tbody>
</table>

Uranium 238 (92 protons, 146 neutrons)
Thorium 234 (90 protons, 144 neutrons)

Radioactivity, $A(t) = A_0 \exp(-t/\tau)$
IAEA Scheme for Classification of Radioactive Waste (2009)

1. **Exempt waste** (EW) – such a low radioactivity content, which no longer requires controlling

2. **Very short-lived waste** (VSLW) – can be stored for a limited period of up to a few years to allow its radioactivity content to reduce by radioactive decay. It includes waste containing radionuclides with very short half-lives often used for research and medical purposes

3. **Very low level waste** (VLLW) – usually has a higher radioactivity content than EW but may, nonetheless, not need a high level of containment and isolation. Typical waste in this class includes soil and rubble with low levels of radioactivity which originate from sites formerly contaminated by radioactivity

4. **Low level waste** (LLW) - this waste has a high radioactivity content but contains limited amounts of long-lived radionuclides. It requires robust isolation and containment for periods of up to a few hundred years and is suitable for disposal in engineered near-surface facilities. It covers a very broad range of waste and may include short-lived radionuclides at higher levels of activity concentration, and also long-lived radionuclides, but only at relatively low levels of activity concentration

5. **Intermediate level waste** (ILW) – because of its radioactivity content, particularly of long-lived radionuclides, it requires a greater degree of containment and isolation than that provided by near surface disposal. It requires disposal at greater depths, of the order of tens of metres to a few hundred metres

6. **High level waste** (HLW) – this is waste with levels of activity concentration high enough to generate significant quantities of heat by the radioactive decay process or waste with large amounts of long-lived radionuclides that need to be considered in the design of a disposal facility for such waste. Disposal in deep, stable geological formations usually several hundred metres or more below the surface is the generally recognized option for disposal
Breeding and burning: the Thorium-Uranium cycle

Much lower production of Transuranics
Remember that

✓ $1 \text{ GW}(th) = 1 \text{ GW thermal power}$

✓ $1 \text{ GW}(e) = 1 \text{ GW electrical power}$

✓ typically, for a fossil-fueled or nuclear power plant, a conversion factor between ~ 30 to 60 % has to be applied to go from thermal to electrical power
How much fuel?

Suppose you’ve got a reactor with 1 GW thermal power (1 GW_th → ~ 300 Mw_e) = 10^9 Joule/sec
Assume each fission releases order of 200 MeV energy = 3.2x10^{-11} Joule
→ In the reactor the fission rate is about 3x10^{19} fissions/sec
→ which means that e.g. 3x10^{19} (nuclei of $^{235}$U)/sec disappear (actually a bit more because of radiative capture)

<table>
<thead>
<tr>
<th>Fuel</th>
<th>Istantaneous consumption (per second)</th>
<th>Yearly consumption (90 % load factor)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uranium</td>
<td>0.012 g</td>
<td>340 Kg</td>
</tr>
<tr>
<td>Natural Gas</td>
<td>25 m³</td>
<td>700 million m³</td>
</tr>
<tr>
<td>Crude oil</td>
<td>0.02 tons</td>
<td>0.7 million tons</td>
</tr>
<tr>
<td>Lignite</td>
<td>100 Kg</td>
<td>2.8 million tons</td>
</tr>
<tr>
<td>Coal</td>
<td>40 Kg</td>
<td>1.1 million tons</td>
</tr>
</tbody>
</table>

For a thermal reactor (see later) loaded with mixed UO$_2$ fuel (density about 11 gr/cm$^3$) comprising 4% $^{235}$U and 96% $^{238}$U, this corresponds to 8500 Kg of fuel → 0.8 m$^3$

In practice, there has to be much more as the chain reaction needs the presence of fissile nuclei at all times → the reactor has to be critical at all times
However, $^{235}$U consumption is partly compensated by Plutonium ($^{239}$Pu) breeding

(*) load factor=percentage of time when the reactor is actually producing electricity
It is customary to adopt the following classification:

- **slow neutrons**: those with kinetic energy $T_n < 1$ eV
- in particular **thermal neutrons** have $T_n$ around 0.025 eV or 25 meV (the value of $kT$, where $k$ is the Boltzmann constant and $T$ is the temperature
- **epithermal neutrons**: $1$ eV $< T_n < 100$ keV (0.1 MeV)
- **fast neutrons**: $0.1$ MeV $< T_n < 20$ MeV

Obviously neutrons in general can have energies above 20 MeV but this is an extreme limit in reactor physics (e.g. neutrons from D+T fusion have 14 MeV fixed energy)
It is easy to show in non-relativistic kinematics that after a scattering off a nucleus with mass number $A$, the kinetic energy of the neutron changes according to the ratio

$$\frac{T'_n}{T_n} = \frac{m_n^2 + m_A^2 + 2m_nm_A\cos\theta_{CM}}{(m_n + m_A)^2}$$

Assuming an isotropic CM cross section that does not depend on $\cos\theta_{CM}$, the corresponding term averages out to zero, so that we can write on average

$$\frac{T'_n}{T_n} = \frac{m_n^2 + m_A^2}{(m_n + m_A)^2} \quad \Rightarrow \quad \text{Assuming } M_A \approx Am_n \Rightarrow \quad \frac{T'_n}{T_n} = \frac{1 + A^2}{(1 + A)^2}$$

For a heavy nucleus $A \gg 1 \Rightarrow T'_n \approx T_n$ or in other words, the neutron has to undergo many collisions in order to significantly lose energy.
Consider instead the case $A=1 \Rightarrow$ (target containing hydrogen, i.e. protons as nuclei) $T'_n = T_n/2$ i.e. on average a neutron will lose half of its energy at each collision and therefore few collisions are sufficient to rapidly decrease its energy

$\Rightarrow$ Moderators = light materials containing hydrogen = water, paraffin or graphite
The chain reaction and the critical reactor

- must not diverge (more and more fissions at each “generation”)
- must not die away (less and less fissions at each generation)

Precisely one neutron from each fission has to induce another fission event.

The remaining fission neutrons will then either be absorbed by radiative capture or will leak out from the system.

Suppose we can count the number of neutrons in one generation and in the next one. Then

\[ k \equiv \frac{\text{number of neutrons in one generation}}{\text{number of neutrons in the preceding generation}} \]

- The condition \( k=1 \) corresponds to a critical reactor
- \( k>1 \) is a supercritical reactor (chain reaction diverges)
- \( k<1 \) is a subcritical reactor (chain reaction dies away)
“Simple-minded” reactor kinetics

\[
\frac{dn(t)}{dt} = P(t) - L(t)
\]

\(n(t)\) = neutron population at time \(t\)
\(P(t)\) = neutron production at time \(t\) (mainly as fission products)
\(L(t)\) = neutron loss (fission+capture+leakage) at time \(t\)

All are functions of time as reactor evolves over time

Alternative definition

\[k \equiv \frac{P(t)}{L(t)}\]

Neutron lifetime \(\equiv \tau \equiv \frac{n(t)}{L(t)}\)

\[
\frac{dn(t)}{dt} = \frac{k-1}{\tau} n(t)
\]

Let’s assume \(k\) and \(\tau\) are time independent (not true...)

\[
n(t) = n_0(t) \exp\left(\frac{k-1}{\tau} t\right)
\]

- \(k=1 \rightarrow \text{steady state} \rightarrow \text{critical reactor}\)
- \(k>1 \rightarrow \text{increase} \rightarrow \text{supercritical}\)
- \(k<1 \rightarrow \text{decrease} \rightarrow \text{subcritical}\)

Time constant \(\equiv T \equiv \text{Reactor period} \equiv \frac{\tau}{k-1}\)
Delayed neutrons: crucial for reactor control

Typical neutron lifetime in a thermal power reactor $\sim 10^{-4}$ sec

If $k=1.001$, $T=0.1$ sec $\quad$ power will increase by 2.7 in 0.1 sec !!

Actually, we neglected the very small amount ($< 1 \%$) of delayed neutrons

Emitted by fragments after fission on time scale from ms to sec

The trick is to make the reactor critical thanks to that small fraction of neutrons

$\rightarrow$ Delayed neutrons dominate the reactor response time making it much longer

Reactors control manageable by absorbers: “control rods”
Physics of multiplication: path representation

Fission neutron  
\[ P_{NL} \] Absorbed in the system

\[ \eta = \text{average number of neutrons produced per neutron absorbed in the fuel} \]

\[ P_{NL} = \text{probability of non-leakage (for a finite system)} \]
\[ f = \text{conditional probability that, if neutron will be absorbed, it will be absorbed in fuel} \]
Physics of multiplication

Multiplication can be written as

\[ k = \frac{N_2}{N_1} = \eta \, f \, P_{NL} \]

\( N_1, \, N_2 \) = number of neutrons in two subsequent generations

\( \eta \) = average number of neutrons produced per neutron absorbed in the fuel

\( f \) = conditional probability that, if neutron will be absorbed, it will be absorbed in fuel

\( P_{NL} \) = probability of non-leakage

Infinite reactor \( \rightarrow P_{NL} = 1 \) \( \rightarrow k_\infty = \eta \, f \)

✓ This is a property of the material, not of the geometry
✓ For a finite, non-homogenous reactor \( \rightarrow \) effective \( k \), or \( k_{eff} \)

\( F \) = Fission cross section in the fuel

\( a \) = Absorption cross section in the fuel

\( \sigma_F \) = Fission cross section in the fuel

\( \sigma_a \) = Absorption cross section in the fuel

\( \nu \) = Average number of emitted neutrons
Simple considerations

\[ k = \frac{N_2}{N_1} = \eta f P_{NL} \]

If \( n \) is absorbed, it is absorbed in the fuel

\[ f < 1, \ P_{NL} < 1 \]

To have \( k \approx 1 \) \( \Rightarrow \) \( \eta \) significantly > 1

Which is indeed the case (on average):
variation of \( \eta \) with energy for
\( ^{233}\text{U}, \ ^{235}\text{U}, \ ^{239}\text{Pu}, \ ^{241}\text{Pu} \)
Physics of multiplication: visual representation

- Shielding assembly
- Reflector assembly
- Control assembly
- Fuel assembly
- Fuel
- Radiation shield
- Absorbing element
- Leakage
- Neutron emerging from fission
- This neutron is bounced back by the «reflecting» material
- Absorption in other materials
- Reactor «core»
- Fission 1
- Fission 2
- Fission 3
Neutron population and reactor classes

- It is easiest to maintain a fission chain reaction using slow neutrons.

- Hence most nuclear reactors until now (Gen. I to III+) use low mass number materials such as water or graphite to slow down or moderate the fast fission neutrons.

- Neutrons slow down to energies comparable to the thermal energies of the nuclei in the reactor core.

- **Thermal reactor**: average neutron energy comparable to thermal energies.

- It requires the **minimum amount of fissile material** for fueling.

  ✓ As an example, a **Light Water Reactor (LWR)** can start with 3% $^{235}$U + 97% $^{238}$U.

  ✓ **Burn-up** of $^{235}$U is compensated by **breeding** of $^{239}$Pu.

  ✓ After 1 year, the core may contain 1% $^{235}$U + 1% $^{239}$Pu.
Neutron population and reactor classes

However
the number of neutrons emitted per neutron absorbed in the fuel is largest for fast neutrons

→ but $\sigma_f$ is smaller

→ need much more fissile content and higher neutron fluxes to reach comparable power

→ to keep the neutron energy high, only high mass-number materials in the core

→ Fast reactor: average neutron energies above 100 keV

→ one can use the "extra" neutrons to convert or breed new fuel → “Fast Breeder”
The thermal reactor

- **Moderator (e.g. water)**
- **Fuel rod**
- **Control rod (e.g. Boron)**
- **Coolant (e.g. water)**

- **Fission**
- **Capture**
- **Escape**
- **Moderation**
The fast reactor

- Coolant: e.g. liquid metal
- Fuel rod
- 235
- 238
- 238
- 238

Capture

Fission

Capture

Scattering

Control rod (e.g. Boron)

→ See presentation on Generation IV reactors by R. Caciuffo

Coolant: e.g. liquid metal

Escape

Fuel rod

Liquid metal
Nuclear reactor zoo

Most current reactors
→ ordinary water serves as both coolant and moderating material in the reactor

There are two major types of Light Water Reactors (LWR):
1) pressurized water reactors (PWR)
2) boiling water reactors (BWR)

In a **PWR** the primary coolant is water maintained under very high pressure (~160 bar) → high coolant temperatures (> 300°) without steam formation within the reactor

In a **BWR**, the primary coolant water is maintained at lower pressure (~ 70 bar) → appreciable boiling (at ~ 285°) and steam within the reactor core itself → the reactor itself serves as the steam generator → no secondary loop and heat exchanger

In both PWR and BWR, the nuclear reactor itself and the primary coolant are contained in a **large steel pressure vessel** designed to accommodate the high pressures and temperatures
Nuclear reactor zoo

Heavy water (D$_2$O) reactor
→ deuteron has lower neutron capture cross section with respect to hydrogen
→ low-enrichment uranium fuels (including natural uranium)
→ Developed in Canada in the CANDU (CANadian Deuterium Uranium) series of power reactors and in the UK as Steam Generating Heavy Water Reactors (SGHWR).

Gas-based reactors
→ the early MAGNOX reactors developed in the UK: low-pressure CO$_2$ as coolant
→ High-Temperature Gas-cooled Reactor (HTGR, USA): high-pressure helium as coolant
→ Pebble-bed concept
→ Advanced Gas Reactors (AGR, Germany and UK)
Decay heat

Decay heat is the heat released as a result of radioactive decay: the energy of the alpha, beta or gamma radiation is converted into atomic motion.

In nuclear reactors decay of the short-lived radioisotopes created in fission continues at high power, for a time after shut down.

Heat production comes mostly from β decay of fission products.

A practical approximation is given by the formula:

\[
\frac{P}{P_0} = 6.6 \times 10^{-2} \left[ \frac{1}{(\tau - \tau_s)^{0.2}} - \frac{1}{\tau^{0.2}} \right]
\]

Where \( P \) is the decay power, \( P_0 \) is the reactor power before shutdown, \( \tau \) is the time since reactor startup and \( \tau_s \) is the time of reactor shutdown measured from the time of startup (in seconds).

At shutdown, the heat power is about 6.5 % (~200 MWth for a 1 GWe reactor) sufficient to melt the core....

About 1 hour after shutdown, the decay heat will be about 1.5% of the previous core power. After a day, the decay heat falls to 0.4%, and after a week it will be only 0.2%

Spent fuel rods are kept for long time in water pool, before storage or reprocessing.

Removal of decay heat very important ⇒ Fukushima...

“Heat sink” must not be compromised.
The nuclear fuel cycle

1. **Depleted uranium**
2. **Uranium enrichment**
3. **Fuel fabrication**
4. **Fresh fuel**
5. **Conversion**
6. **Reprocessing**
7. **Plutonium**
8. **Conversion**
9. **High-level waste**
10. **Spent nuclear fuel (SNF)**
11. **Waste disposal**

**“closed” fuel cycle**

- Reprocessing → fuel recycling
- Fuel recycling (Recycled uranium)

**“once-through” cycle**

- Cycle stops here → “open” fuel cycle
- Uranium hexafluoride
- Natural uranium
- Spent fuel
Control of abnormal operation should include some (negative) feedback mechanisms: e.g. if temperature (power) goes up, reaction cross section goes down.
Nuclear energy today in the world

Nuclear reactors in operation or in long-term shutdown as of July 2019

Total number of reactors = 449

Worldwide nuclear generating capacity and number of operating reactors (1965-2011)

Source: OECD/NEA – Nuclear Energy Today 2012

Nuclear reactors in construction as of July 2019

Total number of reactors = 61

Source: IAEA Power Reactor Information System (PRIS)
Share of nuclear power in total electricity (July 2019)

Source: IAEA Power Reactor Information System (PRIS)
Nuclear energy in the worldwide perspective

World Total Primary Energy Supply (TPES, 2015)

- Coal: 28.1%
- Natural gas: 21.6%
- Oil: 31.7%
- Nuclear: 4.9%
- Other: 1.5%
- Hydro: 2.5%
- Biofuels and waste: 9.7%

**13 647 Mtoe**

1. World includes international aviation and international marine bunkers.
2. In these graphs, peat and oil shale are aggregated with coal.
3. Includes geothermal, solar, wind, tide/wave/ocean, heat and other.

World electricity generation (2015)

- Coal: 39.3%
- Natural gas: 22.9%
- Oil: 4.1%
- Hydro: 16.0%
- Non-hydro renewables and waste: 7.1%

**24 255 TWh**

1. Excludes electricity generation from pumped storage.
2. Includes geothermal, solar, wind, tide/wave/ocean, biofuels, waste, heat and other.
3. In these graphs, peat and oil shale are aggregated with coal.

Source: IEA, Key World Energy Statistics, 2017

(*) 1 tonne oil equivalent (toe) = 41.868 GJ = 10 Gcal = 11.63 MWh
(**) 1 TW = 10^{12} Joule/s, 1 TWh = 3.6 \cdot 10^{15} J
Reactor types in use worldwide (end of 2016)

**PWR**; 64.7%

**BWR**; 17.3%

**PHWR**; 10.9%

**GCR**; 3.1%

**LWGR**; 3.3%

**FBR**; 0.4%

**PWR** = Pressurized Water Reactor

**BWR** = Boiling Water Reactor

**PHWR** = Pressurized Heavy Water Reactor

**GCR** = Gas-Cooled Reactor

**LWGR** = Light Water cooled, Graphite moderated Reactor

**FBR** = Fast Breeder Reactor

Source: [European Nuclear Society](https://www.european-nuclear-society.org/)

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Note: The image includes a pie chart illustrating the distribution of reactor types, with the following proportions:

- **PWR** (Pressurized Water Reactor): 64.7%
- **BWR** (Boiling Water Reactor): 17.3%
- **PHWR** (Pressurized Heavy Water Reactor): 10.9%
- **GCR** (Gas-Cooled Reactor): 3.1%
- **LWGR** (Light Water cooled, Graphite moderated Reactor): 3.3%
- **FBR** (Fast Breeder Reactor): 0.4%
Cost of electricity

LCOE (Levelized Cost Of Electricity) for various technologies (USD/MWh)

- Measures lifetime costs divided by energy production
- Calculates present value of the total cost of building and operating a power plant over an assumed lifetime
- Allows comparison of different technologies with unequal life spans, project size, different capital cost, risk, return, and capacities

Source:
IEA/NEA, Projected Costs of Generating Electricity, 2015
Assuming biomass feedstocks are dedicated energy plants and crop residues and 80-95% coal input.

2 Assuming feedstocks are dedicated energy plants and crop residues.

3 Direct emissions of biomass power plants are not shown explicitly, but included in the lifecycle emissions. Lifecycle emissions include albedo effect.

4 LCOE of nuclear include front and back-end fuel costs as well as decommissioning costs.

* Carbon price levied on direct emissions. Effects shown where significant.
Currently Commercially Available Technologies

Transport and storage costs of CCS are set to 10 USD\textsubscript{2010}/tCO\textsubscript{2}. 

Pre-commercial Technologies

5 Transport and storage costs of CCS are set to 10 USD\textsubscript{2010}/tCO\textsubscript{2}. 

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Perspectives and issues
Worldwide energy trends: projection on energy supply

Total primary energy supply by fuel type (in million tonnes oil equivalent)

1. In these graphs, peat and oil shale are aggregated with coal.
2. Includes international aviation and marine bunkers.
3. Includes biofuels and waste, geothermal, solar, wind, tide, etc.
4. Based on a plausible post-2016 climate-policy framework to stabilise the long-term concentration of global greenhouse gases at 450 ppm CO2-equivalent.

Source: IEA, Key World Energy Statistics, 2017
How long will U resources last?

As an example, **fuel fabrication** for a big nuclear power plant with **1000 MWe production**, requires about **160 tons natural U per year**

→ In the **current scheme** with about 450 reactors and **369,000 MWe capacity**, “conventional” (cheap) reserves would last for another **80 years** (maybe less if average reactor power will increase)

→ Should nuclear power increase as in some of the above scenarios, we should think about (more expensive) resources like phosphates (doable) or U from sea water (still under study)

→ **Switching to fast reactors/Thorium cycle would increase availability to a few 100/few 1000 years**

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<table>
<thead>
<tr>
<th>Country</th>
<th>Uranium (million tons)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Australia</td>
<td>1.14</td>
</tr>
<tr>
<td>Kazakhstan</td>
<td>0.82</td>
</tr>
<tr>
<td>Canada</td>
<td>0.44</td>
</tr>
<tr>
<td>USA</td>
<td>0.34</td>
</tr>
<tr>
<td>South Africa</td>
<td>0.34</td>
</tr>
<tr>
<td>Namibia</td>
<td>0.28</td>
</tr>
<tr>
<td>Brazil</td>
<td>0.28</td>
</tr>
<tr>
<td>Russian Federation</td>
<td>0.17</td>
</tr>
<tr>
<td>Uzbekistan</td>
<td>0.12</td>
</tr>
</tbody>
</table>

| Source: OECD/NEA, **Nuclear Energy Outlook**, 2008 |

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Lifetime of uranium resources (in years) for current reactor technology and future fast neutron systems (based on 2006 uranium reserves and nuclear electricity generation rate)

<table>
<thead>
<tr>
<th>Identified resources</th>
<th>Total conventional resources</th>
<th>Total conventional and unconventional resources</th>
</tr>
</thead>
<tbody>
<tr>
<td>Present reactor technology</td>
<td>100</td>
<td>300</td>
</tr>
<tr>
<td>Fast neutron reactor systems</td>
<td>&gt; 3 000</td>
<td>&gt; 9 000</td>
</tr>
</tbody>
</table>
Reactor ageing

Reactor construction starts and share of nuclear power in total electricity generation

Typical lifetime of a plant is 40 to 60 years → many are being or are going to be decommissioned

Nuclear Power in a Clean Energy System, IEA 2019
Construction costs

Projected overnight construction cost\(^1\) of nuclear power capacity and recent United States and Western European experience


Nuclear Power in a Clean Energy System, IEA 2019

\(^1\)Cost of a construction project if no interest was incurred during construction, as if the project was completed "overnight."
Scientists warn 20% of US nuclear capacity at risk; Toshiba liquidates UK new build company

Our pick of the latest nuclear power news you need to know.

Scientists say 20% of US nuclear capacity unprofitable or set to close

More than one third of operational U.S. nuclear plants, representing 20% of nuclear capacity, are unprofitable or scheduled to close, the Union for Concerned Scientists (UCS) said in a report published November 8.

Nuclear operators face continuing pressure from low wholesale electricity prices, driven by low gas prices and rising renewable energy capacity. The states of New Jersey, New York and Illinois have introduced support mechanisms for nuclear plants but much wider support is required to prevent a hike in carbon emissions, UCS said.

In May, Bloomberg New Energy Finance (BNEF) said more than a quarter of U.S. nuclear power plants do not earn enough
In 1 year, a typical high power reactor produces about 1,200 Kg of radioactive material in the fuel, of which 400 Kg are long-lived, with half-lives from a few hundred to a few hundred thousand years → they need to be segregated.

Spent fuel assemblies are put in interim safe storage for decades, but the same has to be done for the heavily activated structural materials when dismantling a decommissioned plant.

Some countries (most advanced are Finland, Sweden and France) have plans to store spent fuel and other High Level and possibly Intermediate Level Waste in underground repositories at a few hundred meters depth: the so-called geological repositories → After a few 100,000 years radioactivity dropped to safe levels.

Some countries (France, in particular) recycle the spent fuel by using specific processes to extract Uranium and Plutonium that can be used to produce fresh fuel (MOX, Mixed Oxide fuel).
Fast spectrum systems and waste incineration (transmutation)

Several Minor Actinides are characterized by a fission threshold around the MeV

- Such isotopes can be burnt in fast reactors or in fast Accelerator Driven Systems (ADS)
  (neutron spectrum from 10 keV to 10 MeV)

- See presentation on Generation IV reactors by R. Caciuffo and MYRRHA project by H. Abderrahim
Nuclear accidents

The fission process creates radioactive substances that can be dangerous for the environment and for human health. They must be contained as much as possible.

Safety measures are essential and are subject of continuous research and improvement.
Accidents into perspective...

Train carrying liquid gas explodes in Italy killing 12

At least 50 injured as freight train derails in Viareggio and hits homes of sleeping families
‘Avian incident’ knocks out 84% of massive California solar farm

By BLOOMBERG  JUNE 20, 2019  |  8:05 AM

An “avian incident” sparked a fire at one of California’s biggest solar farms, affecting 1,200 acres and knocking out 84% of the California Valley Solar Ranch’s generating capacity.

The June 5 incident didn’t damage solar panels at the 250-megawatt power plant, but distribution poles and cables need to be replaced, according to a regulatory filing Wednesday from owner Clearway Energy Inc. The company didn’t say exactly how the blaze was ignited.

About 40 megawatts of the San Luis Obispo County facility are in operation, and it’s expected to return to full service by July 1. Clearway expects the incident to cost $8 million to $9 million this year, after estimated insurance recovery.
Past and future

**Generation I**
- Early prototype reactors
  - Shippingport
  - Dresden
  - Magnox

**Generation II**
- Commercial power reactors
  - PWRs
  - BWRs
  - CANDU

**Generation III**
- Advanced LWRs
  - CANDU6
  - System 80+
  - AP600

**Generation III+**
- Evolutionary designs
  - ABWR
  - ACR 1000
  - AP 1000
  - APWR
  - EPR
  - ESBWR

**Generation IV**
- Revolutionary designs
- Enhanced safety
- Minimisation of waste and better use of natural resources
- More economical
- Improved proliferation resistance and physical protection