Chapter 3 NUCLEAR ENERGY

3.1. Introduction

The expression „nuclear energy" is used for the energy emitted by two nuclear reaction, *fission* and *fusion*, which change the structure of the atomic nuclei entering the reaction.

None other energy conversion process affects the atomic nucleus.

Fission means the reaction in which the nucleus of a heavy atom splits into smaller parts (lighter nuclei). The fusion is the opposite process, when two nuclei join together, or "fuse", to form a single heavier nucleus.

Both reactions present a common issue, the ***excess mass***, which constitutes the source of the released energy. This energy, related to the unity of mass, is enormous by comparison with the fossil fuel burning.

Until now, only the fission could be controlled and utilized to produce large amounts of electricity (some 6% of the world electricity.

The nuclear fission remains many decades the unique energy source, able to compete with the fossil fuels to generate electricity and heat both for industrial and domestic consumers.

The nuclear fusion is still too far from the commercial use, although large funds were spent for research and development works by some advanced countries. But, considering the quantities of raw materials existing on the earth, the energy potential of the fusion seems to be unlimited by comparison with the actual world consumption.

The main characteristics of the nuclear energy can be defined as follows:

- The largest potential comparatively with all other conventional sources together.
- An energy concentration, of 10^5 - 10^6 times higher than the best fossil fuel.
- Rated power high enough to substitute, in short time gaps, a significant installed power in thermo-power plants.
- The unique energy utilization is the generation of power and heat.
- Regarding the transport and storing, the costs for the nuclear fuel are much smaller than for the fossil fuels, because, for the same energy generated, the quantities are much smaller too.
- The possibilities for other use of the nuclear fuel than the energy generation are only military and have tremendous consequences.
- The main disadvantages of the nuclear energy are the potential environmental consequences of an accident with releasing of radioactive isotopes. Some major accidents have had important consequences on the nuclear industry development. In normal operating conditions, the nuclear

fission produces radioactive wastes whose safe storing represents a great difficulty.

3.2 The energy potential of fission reaction

Every chemical element may be characterized by two numbers:

- ***Z*** the order number in the element's table which represents the numbers of protons in the nucleus equal with the number of surroundings electrons;
- ***A*** the mass number (p.u.) which represents the atomic mass related to the atomic mass unit(1 a.m.u.= 1/12 atomic mass of the isotope C-12).

The difference ***A-Z*** represents the number of neutrons in the nucleus.

Although between protons act repulse Colombian forces, all the nucleus components are kept in a very small volume owing to the ***nuclear binding forces***. The existence of such forces was sustained by the mass excess, a common issue of all nuclei. The nucleus mass excess represents the difference between the sum of the rest masses of the protons and neutrons and the nucleus mass.

$$\Delta m = Z * m_p + (A - Z) * m_n - m_N . \qquad (3.1)$$

The binding energy may be calculated with

$$E_b = \Delta m * c^2 . \qquad (3.2)$$

Let us consider a mass excess equal with 1 a.m.u = 1.657 10^{-27} kg. The corresponding binding energy is

$$E_1 = 1.657 * 10^{-27} * \left(2.998 * 10^8\right)^2 = 1.489 * 10^{-10} J = 931\, MeV$$

The $^{238}_{92}U$ nucleus presents a mass excess of

$$\Delta m = 92 * 1.00814 + \left(238 - 92\right) * 1.00895 - 238.16 = 1.8955\, a.m.u.$$

so its binding energy equals

$$E_{238} = 1.8955 * 931 = 1766\, MeV .$$

The specific binding energy calculates by dividing the nucleus binding energy with the atomic mass number, ***A***. The isotope $^{238}_{92}U$ has a per nucleon binding energy of:

$$E_{238} / A_{238} = 1766 / 238 = 7.5\, MeV.$$

Other elements have different specific binding energy as the fig. 3.1 shows. This graph explains the energy release by both the fission and fusion reactions: if the reaction products have specific binding energy higher than the initial products, the reaction will be accompanied by an energy release. Thus, by splitting the $^{235}_{92}U$ nucleus, which has an binding energy of 7.5 MeV, into two easier nuclei, having a binding energy of some 8.4

MeV, the energy release is of 8.4-7.5 = 0.9 MeV/nucleon, or 0.9*235 = 211 MeV.

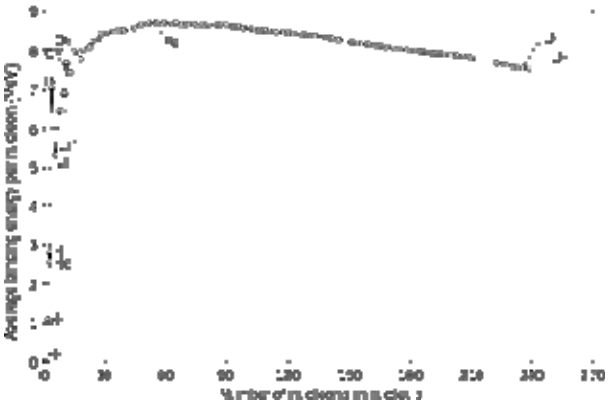

Fig.3.1 – The specific binding energy of the existing nuclei

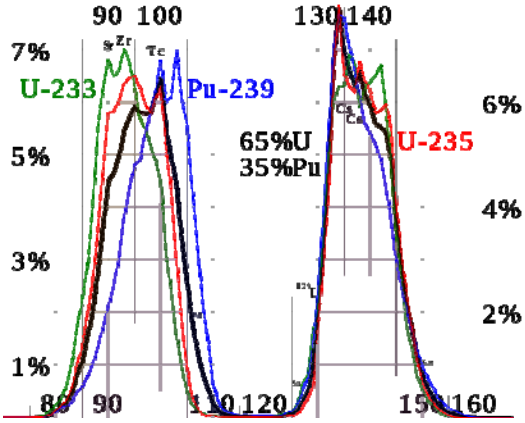

Fig.3.2 Fission products yields by mass for thermal fission

The fusion of two deuterium nuclei (binding energy 2.33 MeV) to form a helium nucleus (binding energy 8.48 MeV) can release an energy of 8.48-2*2.33= 4.02 MeV.

The reaction products are not the same for all fissions. The likelihood of certain isotope arising depends of its atomic mass, as fig.3.2 shows

A fission reaction of the ${}^{235}_{92}U$ nucleus has the general form:

$$ {}^{235}_{92}U + {}^{1}_{0}n \rightarrow {}^{A1}_{Z1}F1 + {}^{A2}_{Z2}F2 + \nu\,{}^{1}_{0}n + E, $$

where

$$ Z1 + Z2 = 92, \quad A1 + A2 + \nu = 235+1. $$

All the fission products present an excess of neutrons in the nuclei, so they are radioactive.

An frequently produced fission reaction is:

$$ {}^{235}_{92}U + {}^{1}_{0}n \rightarrow {}^{95}_{36}Kr + {}^{139}_{56}Ba + 2\,{}^{1}_{0}n + E. $$

The corresponding mass excess may be determined as follows:

Input masses		Output masses	
U-235	235.124	Kr-95	94.945
1 x n	1.00895	Ba-139	138.955
		2 x n	2.0179
total mass	236.13295		235.9179
mass excess	0.21505 u.a.m.		

The released energy reaches:

$$ E_{fiss.} = 0.21505 * 931 = 200.16\ MeV $$

This quantity is distributed to more energy carriers (table 3.1).

200 MeV seems to be a very small quantity of energy; converted in kWh it represents only $8.89*10^{-18}$ kWh. But, owing to

the very small mass of an atom, the quantity becomes very large, if related, e.g., to 1 kg U-235:

$$E_{kg} = \frac{N_A}{A} 200 = \frac{6{,}023 * 10^{26}}{235.124} 200 = 5.123 * 10^{26} MeV = 22768\, MWh$$

Table 3.1 The structure of the energy released by a U-235 fission

Energy carrier	Amount of energy	
	(MeV)	(%)
Fission products	168	84.0
Radiation β	8	4.0
Radiation γ	7	3.5
Neutrino	12	6.0
Secondary neutrons	5	2.5
Total	200	100.0

If consider the fission of all quantity of U-235 in a metric ton of natural uranium (concentration f = 0.714%), the released energy may be:

$$E_{nat} = \frac{f}{100} E_{kg} * 10^3 = \frac{0.714}{100} 22768 * 10^3 = 162663\, MWh = 6777 MWz.$$

To obtain an energy of 1MWh, a quantity of

$$m_{MWh} = \frac{1}{22768} kg = 4.39 * 10^{-5} kg = 43.9\, mg$$

must be used.

An energy amount of 1 MWh is contained in much larger quantities of fossil fuels:

- Equivalent coal (q = 7000 kcal/kg)

$$M_{ec} = \frac{860 * 10^3}{7000} = 122.85\, kg;$$

- Conventional oil (q= 10000 kcal/kg)

$$M_{eo} = \frac{860 * 10^3}{10000} = 86\, kg;$$

- Poor lignite (q=1800 kcal/kg)

$$M_{ec} = \frac{860 * 10^3}{1800} = 477.78\, kg;$$

The fossil fuels quantities needed to obtain 1 MWh are from $1.96*10^6$ to $1.09*10^7$ times larger than the uranium quantity.

Such estimation are purely illustrative because some factors like efficiency of conversion processes, material losses, partial utilization of nuclear fuel in reactor, artificial fuel production etc. were neglected.

3.3. The nuclear reactor

3.3.1 Basic phenomena

The neutrons support the fission reaction. To obtain the operation of a reactor at constant power level, the neutron density in the reactor core must be constant too. The nuclear

fuel behavior in the fission reaction depends both of the nucleus structure and the neutron speed or energy.

According to the „water drop" model for the fission mechanism, the nuclei having an even number of protons and an odd number of neutrons (like $^{235}_{92}U$), split mostly with slow speed neutrons, while the nuclei having both numbers even (like $^{238}_{92}U$) split mostly with high speed neutrons.

The energy of neutrons in the reactor core may be very various. A classification based on the neutron energy was adopted:

- *Fast neutrons, having energy higher than 0.1 MeV;*
- *Intermediate neutrons, having energy from 1 eV to 0.1 MeV;*
- *Slow neutrons, having energy less than 1 eV.*

There exists a direct relation between the energy, the speed of a particle and the temperature, so the slow neutrons mostly are called „thermal" neutrons. That is because temperature corresponding to the speed is close to the reactor core temperature.

The number of secondary neutrons from the fission reactions of U-235 may be from 1 to 5. The most frequent are reactions with two secondary neutrons (34%) and three neutrons (30%), so the average number of neutrons per fission is 2.43. If one of the secondary neutrons can initiate a new fission, other fissions will succeed, what is called a „chain reaction".

The fission reaction is a probabilistic event, characterized by the ***„efficacious section"***. For the interaction neutron-nucleus this parameter is called ***„microscopic section"***, while for the interaction between all the neutrons and nuclei in a volume unit of the reactor core this likelihood is called ***„macroscopic section"***.

The size of the microscopic section express in units of surface area, m^2. Because the values are very small, a conventional unit is mostly utilized:

$$1\,barn = 10^{-24} m^2.$$

The microscopic section is much larger for the fission of the U-235 with thermal neutrons than for the reaction of the U-238 with fast neutrons. Although the proportion of U-235 in nature is much smaller than of U-238 (0.714% and 99.2%), all the reactors operating today use U-235.

The secondary neutrons from fissions are fast, having an average energy over 2 MeV. To become useful for new fissions, whose energy must be reduced under 1 eV, where the microscopic section is the greatest. The energy of neutron may be transferred by elastic collisions to other particle in the reactor core. The collision must be elastic to conserve the neutrons density. To obtain this energy reduction, in the reactor core is introduced a substance called ***„moderator"***. This substance must contain lighter weight atoms, because the energy transfer in an elastic collision is higher when the „bullet" and the „target" have close masses.

In this reason, the best moderator is the hydrogen, but it cannot be used because it's very small density. The first

moderator utilized in nuclear reactors was the graphite (crystallized carbon, atomic mass number 12). Today the (light) water (H_2O) and the „heavy water" (D_2O, D is deuterium) are preferred.

The new arisen neutrons interact with nuclear fuel, moderator, cooling agent and structure materials. As consequence a loss of neutrons appears by absorption or escape from the reactor. The evolution of neutron density is characterized by the multiplication factor *k*, defined as:

$$k = \frac{number\ of\ neutron\ in\ the\ actual\ generation}{number\ of\ neutrons\ in\ the\ previous\ generation}.$$

Three different situations may exist:
- ***k* = 1**, critical regime, the delivered power is constant;
- ***k* < 1**, sub-critical regime, the delivered power lowers;
- ***k* > 1**, supra-critical regime, the delivered power grows.

The value of the multiplication factor results from some phenomena which happen in the reactor core with the participation of neutrons.

a) Consider ***N*** thermal neutrons which are absorbed by the fuel. After fission the neutron number becomes ***ηN***, where ***η*** is

$$\eta = \frac{\Sigma_f}{\Sigma_a}\upsilon .$$

υ is the average number of secondary neutrons from fission reactions. $\mathbf{\Sigma_f}$ and $\mathbf{\Sigma_a}$ are the macroscopic sections of thermal neutrons for fission and absorption reactions. This formula takes into account that only a fraction of the absorbed neutrons in the fuel initiate fissions. The macroscopic section is calculated as:

$$\Sigma = \sigma N ,$$

where ***σ*** is the microscopic section and ***N*** the number of fuel nuclei per volume unit. By considering the measuring units [m^2] for ***σ*** and [atoms/m^3] for ***N***, the macroscopic section size results in [m^{-1}]. As example, for natural uranium ***η*** = 1.02 and for U-235 is 2.06.

b) Some of the fast neutrons may initiate fission reactions. From such reactions result more neutrons than from thermal fissions. As consequence, the fast neutron fissions contribute to an increase of neutron flow. The fast fission factor is:

$$\varepsilon = \frac{number\ of\ neutrons\ from\ thermal\ and\ fast\ fissions}{number\ of\ neutrons\ from\ thermal\ fissions}$$

For example, ε = 1.02-1.08 if the fuel is natural or lightly enriched uranium.

c) some of the fast neutrons whose energy lowers, may be absorbed in the isotope U-238 by the so-called ***„resonance absorption"***. This phenomenon is due to the fact the microscopic absorption section of neutron in the fuel, for a range of

energies in the domain 10^2 – 10^4 eV, presents much higher values (resonance peaks). This phenomenon is evaluated by the probability to avoid the resonance capture – ***p***, defined as:

$$p = \frac{number\ of\ neutrons\ which\ cross\ the\ resonance\ domain}{number\ of\ neutrons\ which\ enter\ the\ resonance\ domain}$$

For the natural uranium, ***p*** = 0.9

d) The thermal neutrons can be absorbed in all the substances existing in the reactor core. The share of neutrons number which enter the fuel is evaluated by the thermal utilization factor, ***f***:

$$f = \frac{number\ of\ neutrons\ absorbed\ in\ the\ fuel}{number\ of\ neutrons\ absorbed\ in\ reactor}$$

Starting from the number of thermal neutrons, ***N***, absorbed in fuel, the next generation will consist of ***N*η*ε*p*f*** neutrons, so the multiplication factor becomes:

$$k = \eta \times \varepsilon \times p \times f,$$

what is known as „***the four factors equation***".

Some fast or thermal neutrons move away from the reactor core which has limited volume. Two other factors take these losses into account:

- p_f – the probability to avoid the escape of fast neutrons
- p_{th} – the probability to avoid the escape of thermal neutrons

By considering these losses, the previous equation may be written:

$$k = \eta \times \varepsilon \times p \times f \times p_f \times p_{th},$$

or the ***six factors equation***.

3.3.2 The reactor rated power

Analytically determination of the nuclear reactor power is too complicated to be treated here. A rough evaluation starts from the macroscopic fission section of the fuel:

$$\Sigma_f = f_{235} \times \rho \frac{N_A}{A} \sigma_f,$$

where:

- f_{235} - the share of fissile atoms in fuel,
- ρ - the fuel density,
- N_A – is the Avogadro's number,
- ***A*** - the atomic mass of fissile element.

For example, if the fuel is natural uranium, this section has the extent:

$$\Sigma_f = \frac{0.714}{100} 18.7 \frac{6.023.10^{23}}{235} 583.10^{-24} = 0.196\ cm^{-1}$$

Considering the density of neutrons ***n*** (neutrons/m^3) and those average speed, ***v*** (m/s), the neutron flow may be evaluated by:

$$\Phi = n \times v \text{ (neutrons/m}^3\text{/s)}.$$

The specific number of fissions (fiss./m^3/s) is:

$$N_f = \Phi \times \Sigma_f .$$

Finally, the reactor rated power is:

$$P = \frac{N_f \times V}{c_f} 10^{-6} \text{ (MW)},$$

where

- ***V*** is the reactor core volume, considered homogeneous regarding the fuel disposal,
- $\mathbf{c_f}$ is a constant representing the number of fissions to gather 1J ($3.1.10^{10}$ fissions/J).

3.3.3 Operation of the thermal nuclear reactor

During the operation of the reactor, the multiplication factor changes according to the set power or to the fuel state. To easier use of this factor, it is replaced by the derived "***reactivity, χ***":

$$\chi = \frac{k-1}{k},$$

or the deviation of the multiplication factor from unity.

The multiplication factor and the reactivity are influenced by the temperature change. To ensure a steadfast operating regime, the dependence coefficient of the reactivity upon temperature must be negative. In this situation, if the core temperature grows (for example, by a lack of cooling agent), the reactivity lowers and the chain reaction intensity lowers too. The time constant of this dependence is of some seconds or minutes.

A specific phenomenon of the reactor operation is the „***poisoning***" with fission products. Some fission products have much higher absorption section for neutrons than the fuel or other materials of the reactor core. Among them, the isotope $^{135}_{54}Xe$ (σ_a= 3.10^6 barn) is the most dangerous. Xe arises by two mechanisms: by uranium fission and by desintegration of the isotope $^{135}_{53}I$, a fission product itself. By a neutron absorption, $^{135}_{54}Xe$ convert into $^{136}_{54}Xe$ which has no more affinity for neutrons. On the other side, being radioactive, $^{135}_{54}Xe$ converts into $^{135}_{55}Cs$, fading away from the core.

The presence of „poison" isotopes affects the neutron flow extent. For the normal operation regime of the reactor, this

effect is compensated by the appropriate design of the core and the share of fissile atoms in the fuel.

The negative effect of the Xe becomes important when the reactor operates at partial power or after an uncontrolled operation ceasing (as consequence of a fault or damage). Because the radioactive half-life of the Xe-135 is lower than that of the I-135, after the chain reaction stops, the density of Xe-135 grows for a time until the atoms of I-135 disappear. Depending of the fuel state and the duration the reactor stops, Xe-135 makes impossible the restart of the reaction, until its density lowers enough. This is a predictable and normal behavior of the thermal reactor and not at all dangerous.

3.4 Burning the nuclear fuel

During the operation of a reactor, the amount of fissile atoms reduces and the enrichment level of the fuel diminishes. By considering a constant neutron flow, the reducing of the number of fissile atoms may be written:

$$dN = -\sigma_a \Phi N dt .$$

σ_a takes into account two phenomena, capture and fission, both having as result a disappearance of a fissile atom.

If at instant $t = 0$ the number of fissile atoms was N_0, then, at a certain moment, t, this number will be

$$N = N_0 e^{-\sigma_a \Phi t}.$$

The diminution of the fissile atoms determines a slowing down of the chain reaction and, consequently, a continuous reduction of the delivered power. Because a reactor is an energy source, an operating regime with lowering power is not acceptable. Every energy source must be controllable regarding starting, stopping and the power level in normal operation. To make possible these demands, the reactor is equipped with a control system and a shut down system.

The fresh fuel contains more fissile material than is necessary to maintain the chain reaction, thus presenting a reactivity excess. To control the reaction intensity, among the nuclear fuel are arranged a number of rods containing neutron absorbing elements, like cadmium. At the beginning of the operation period, these rods are completely introduced in the reactor core.

During the normal operation, as the reaction intensity begins to slow down, the process computer orders to draw out, little by little, the absorbing rods until the power comes back to the prescribed level. This action ends when the absorbing rods are entirely extracted from the core, so the reaction can be controlled no more. At this moment, the chain reaction must be stopped and the used fuel replaced by a fresh one.

Although the used fuel must be replaced, it contains still fissile atoms. The energy developed by a mass unit of fuel during its utilization is called „***burning rate***":

$$B = \frac{P_r T_b}{G} \quad (MW_d/tU),$$

where:

- P_r is the rated power of the reactor,
- T_b is the burning period of the fuel and
- G is the initial amount of fuel.

$1MW_d$ = 24 MWh, is an energy unit, frequently used in nuclear power.

The burning rate varies from a reactor type to another depending on the designing characteristics like fuel enrichment, moderator material, cooling agent. For the existing reactors, the burning rate is from 4000 to 40000 MW_d/tU.

3.5 The fast reactor

Although the microscopic section for uranium fission with fast neutrons is smaller than that for slow neutrons, it is possible to maintain the chain reaction without slowing down the neutrons resulted from fissions. The fuel for fast reactors must be higher enriched than for thermal reactor (for example 10-20% U-235 instead 2-5%). Also, to avoid the neutron slow down, the light or heavy water can not be used as cooling agent.

The fast reactor theory is well known, but there exist such reactor only for research and developing to the commercial types. However, this reactor has a great importance for the future of nuclear power.

3.6 The artificial nuclear fuel

Besides the fissile isotope U-235, the natural uranium contains the U-238 isotope which has a very small section of fission with fast or thermal neutrons. However, the U-238 nuclei, which represent a large majority of the natural uranium, can give capture reaction, after that becomes highly radioactive and converts, in two steps of desintegration β^- until Pu-239. This isotope has a behavior to neutrons, similar to the U-235. The radioactive half-life of Pu-239 is of only 24000 years, while for U-238 it is of some 5.000.000 years. That is why Pu-239 is no more present on the earth.

The capture reaction of U-238 is

$$^{238}_{92}U + ^{1}_{0}n \rightarrow ^{239}_{92}U \rightarrow ^{239}_{93}Np \rightarrow ^{239}_{94}Pu$$

The radioactive half-life of U-239 is of 23.5 min., while for Np-239 is of 2.35 days.

Further, by successive neutron absorptions in the nuclei, appear other isotopes: Pu-240, Pu-241 and Pu-242. The four isotopes of plutonium were detected in the spent fuel but with different shares: 60-70% Pu-239, 20-35% Pu-240, 5-10% Pu 241 and

0-5% Pu-242. The behavior of these isotopes to thermal neutrons follows the previous presented rule: Pu-239 and Pu-241 are fissile, Pu-240 and Pu-241 are not.

A no fissile isotope which can convert into a fissile one, like U-238, is called ***„fertile"***.

Another fertile isotope is Th-232, which exists on the earth. This isotope can transform into U-233, a fissile isotope, after capturing a neutron and two steps of desintegration β^-

$$^{232}_{90}Th + ^{1}_{0}n \rightarrow ^{233}_{90}Th \rightarrow ^{233}_{91}Pa \rightarrow ^{233}_{92}U\,.$$

A desintegration β^- means a nuclear reaction which transforms a neutron in nucleus into a proton and an electron. The new proton determines a growth of the number Z with unity, while the atomic mass remains unchanged.

Table 3.2: The fission and absorption sections of some isotopes

	U-233	U-235	U-238	Pu-239	Pu-240	Pu-241	Pu-242
microscope section for fission (barn)	527	583	0	738	4	971	4.10^{-5}
microscopic section for absorption (barn)	581	693	2.8	1025	350	1336	7.4
neutrons/ fission	2.52	2.43	0	2.91	3	3	0

The fissile isotopes of Pu-239 and U-233 represent the *artificial nuclear fuel*. The generation rate of the artificial fuel depends of some reactor features like nature of moderator, fuel enrichment, energy of neutrons. This process may be characterized by the conversion coefficient, *c*:

$$c = \frac{number\ of\ new\ fissile\ nuclei}{number\ of\ fissile\ nuclei\ burned}$$

If $c < 1$, the reactor is called *convertor.* All the thermal reactors operating today belong to this type.

If $c \geq 1$, the reactor is called *breeder*. Only the reactors which operate with fast neutrons (fast reactor) can be of this type. The artificial fuel Pu-239 is obtained from the fertile isotope U-238 which represents over 99% of the natural uranium. This way, the reserve of nuclear fuel may grow some 90 times in comparison with the existing amount of U-235.

The new generated fuel can be used, in different manner, to fill new reactors. These reactors will generate other quantities of artificial fuel, thus, after a time gap, the installed power will double. This time gap is called *doubling time.*

If $\boldsymbol{G_i}$ represents the initial load of fissile isotopes in the reactor,

$$g_i = \frac{G_i}{P_n}$$

is the specific initial load, where $\boldsymbol{P_n}$ is the rated power. The spent fuel still contains a lower share of fissile isotopes, $\boldsymbol{G_f}$.

The result concerning the artificial fuel may be expressed by

$$s = \frac{G_f - G_i}{P_n T_b} \left(\frac{t}{MWy} \right).$$

s represents the specific gain of fuel of the reactor.

To evaluate the doubling time, firstly must specify the fuel strategy. Two possibilities will be considered.

a) *The linear doubling time*.

Yearly, the new produced fuel is stored until the initial fuel load for a identical reactor is gathered:

$$t_l = \frac{g_i}{s} = \frac{G_i}{|G_s - G_i|} T_b .$$

b) *The exponential doubling time*.

Considering a sufficient great number of identical reactors, the yearly fuel gain is immediately utilized for the initial load of another reactor. In this way, the new fuel contributes ceaselessly to generate new quantities.

Defining the relative yearly fuel gain by:

$$h = \frac{1}{t_l},$$

the installed power in reactors will vary as the next table shows.

Year	1	2	3	...	n
Power (p.u.)	1	1+h	$(1+h)^2$	...	$(1+h)^{n-1}$

The installed power will double after t_e years:

$$t_e = \frac{\ln 2}{\ln(1 + h)}.$$

Such evaluations are very rough, because some factors were neglected, for example:

- Operating periods of time when the reactor power is lower than the rated one;
- Between the moments when the spent fuel is extracted from the reactor and when the new fuel can be utilized for an initial charge of other reactor may pass several months.

Accurate evaluations gave doubling times from 7 to 25 years.

3.7 The nuclear fuel

The term „nuclear fuel” is assigned to a group of natural and artificial isotopes, which contribute to power generation in nuclear reactors. This group may be divided into two sections according to those isotopes participation to the energy processes.

The first group contains ***fissile isotopes*** which direct participate to the fission reaction: U-235 (natural isotope), U-233 and Pu-239 (artificial isotopes).

The second group contains ***fertile isotopes*** which transform in the artificial fissile isotopes, mentioned above: U-238 and Th-232. In this section could be considered the artificial isotope Pu-240, because it may transform into Pu-241, an artificial fissile isotope.

3.7.1 Natural uranium

The uranium ore contains three isotopes:

- $^{234}_{92}U$ with a share of 0.0057%;
- $^{235}_{92}U$ with a share of 0.714%;
- $^{238}_{92}U$ with a share of 99.2274%.

Uranium is not considered a very scarce element in nature because the whole amount is greater than that of other elements of interest like Au, Ag, Hg, Cd etc. The average content of uranium in the earth crust is about 0.2 g/t while the ocean water contents some 0.04-3.0 mg/t.

Using these average figures, the total amount of uranium on earth may be evaluated at $4.1.10^{13}$ tones.

To be economic efficient extracted, the ore must contain at least 0.03% (0.3 kg/t) uranium.

The uranium resources are classified by the level of confidence in the estimates and by production cost categories.

From the first point of view, there are three categorizations:

- Identified Resources,
- Undiscovered Resources, and
- Unconventional Resources.

Identified resources are deposits with known location, quantity, and quality, based on specific measurements for which economic extraction is feasible with existing technologies and under current market conditions. The evaluated amount on earth is about 6,306,000 tones.

Undiscovered resources are expected to exist on the basis of analogies from geological knowledge of previously discovered deposits and regional geological mapping. The total amount is considered to be 6,800,000 tones.

Unconventional resources are generally very low-concentration ores or minor byproducts from other mineral production, which would require new or innovative technology or substantially different levels of demand and market prices for their extraction. The total amount may be of 10,000,000-22,000,000 tones.

According to the production costs of uranium concentrate, there exist four categorizations: less than 40 USD/kg, less than 80 USD/kg, less than 130 USD/kg, and less than 260 USD/kg.

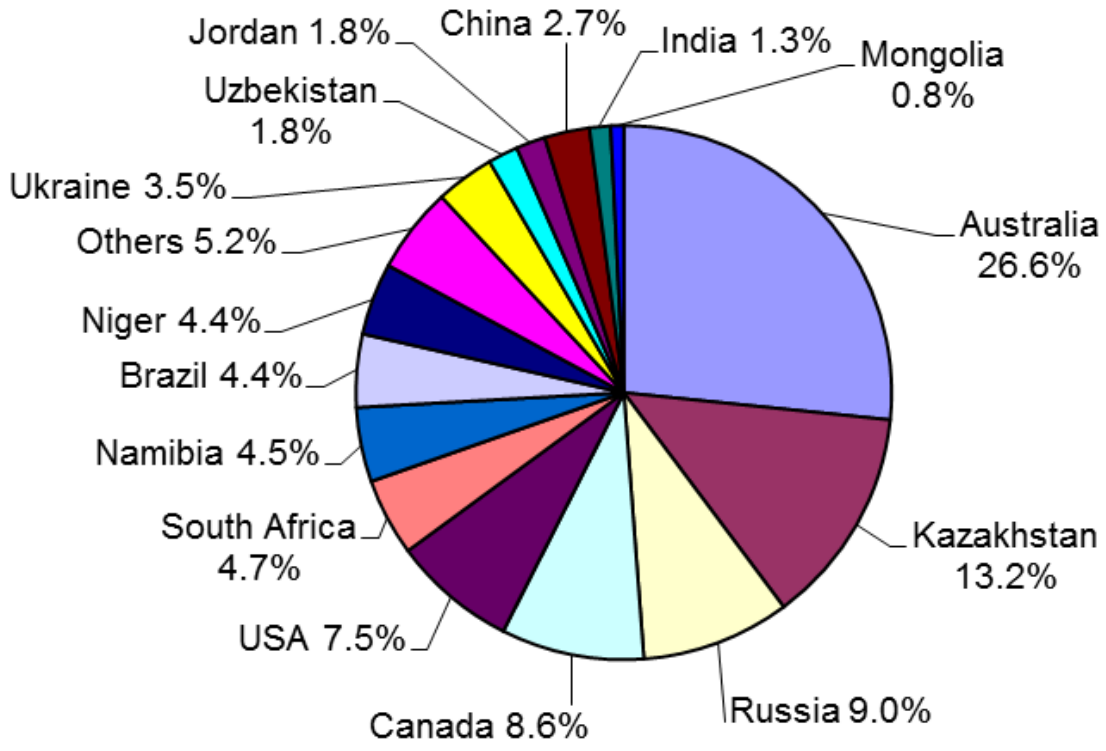

Fig.3.1 Distribution of Identified Uranium Resources at less than USD 260/kgU production costs.

3.7.2 The cycle of the nuclear fuel

From the natural ores, until burning in reactors, uranium traverses more processes, which aggregate in a diagram known as the ***cycle of the nuclear fuel*** (fig.3.2). This cycle starts from uranium ores mining and ends with nuclear wastes disposal and storage. An important part of the spent fuel consists in uranium and plutonium which are recycled to be again utilized. The cycle stages are processed in distinct plants, generally situated in remote zones. These processes last from some weeks to several months, depending of the utilized technologies and the reactor type.

Concentration means the separation of substances containing uranium from the rest of the ore and preparation of uranium for the next stage.

The main operations of concentrations are:

- Sorting the ores according the abundance in uranium;
- Breaking and fine milling
- Solving in acid solution
- Separation of uranium with selective solvents

• Precipitation with base solution and dehydration of the precipitate. The result is known as ***„yellow cake"*** and contains 70-85% uranium.

Further, the concentrate is refined in order to obtain a higher pureness. Finally, the concentrate is chemically converted into UO_2.

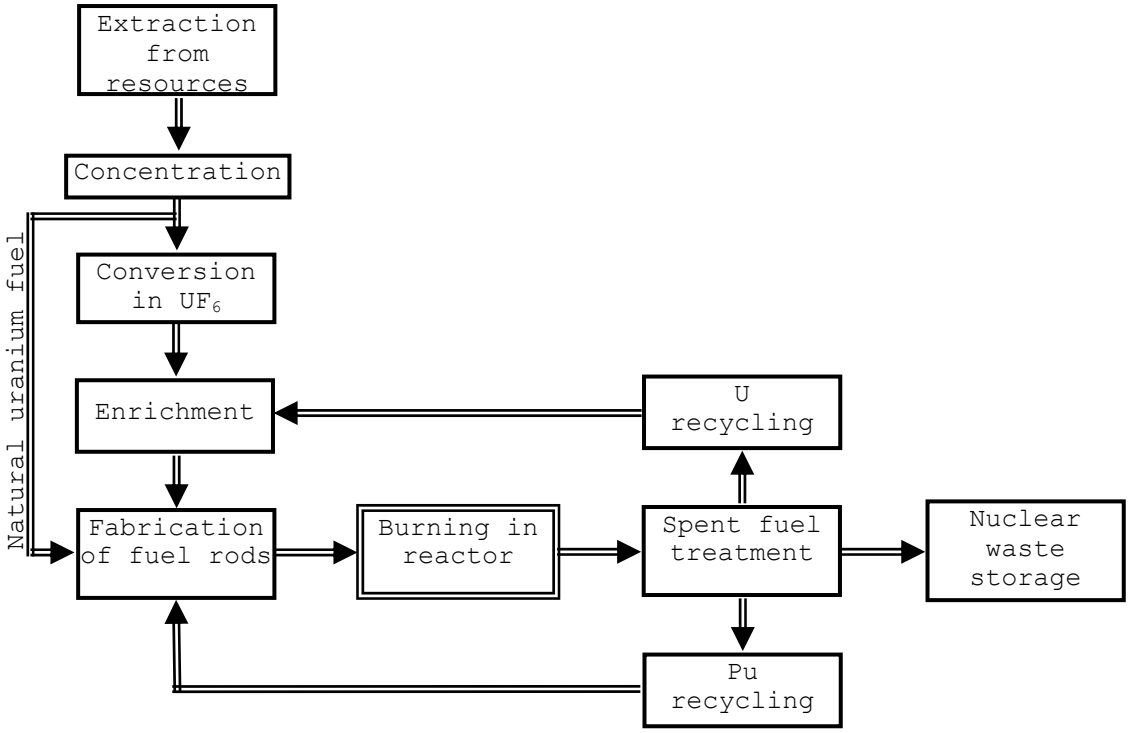

Fig. 3.2 The nuclear fuel cycle

Conversion in UF_6 is a necessary operation if the uranium must be enriched with the fissile isotope U-235, according to the reactor type where it will be used. All the enrichment methods use the uranium in a gaseous shape. Among all the substances containing uranium, only the UF_6 can be changed from solid to gas at low temperature level. This substance is solid at the room temperature (20^0C) and sublimes at $56,5^0$ C.

The source of F is the HF acid, a high corrosive substance, by direct reaction as well as with gaseous F obtained by the HF acid electrolysis.

Uranium enrichment

The small share of the fissile isotope U-235 in the natural uranium allows, however, to run the chain reaction, but he burning degree of the fuel is low enough. The enrichment in fissile isotope is wide utilized in order to improve the fuel utilization as well as to cheapen the reactors.

The enrichment realizes through the isotope separation. Because the isotopes of any element have the same number ***Z***, their chemical behaviors are the same and, consequently, the separation cannot be made by chemical methods.

But, having different atomic mass number, a lot of physical separation methods can be used for separation. Gaseous

diffusion, centrifugal separation, aero-dynamical separation are founded on this property.

A newer separation method, the laser separation, exploits a general property of materials, namely the selective absorption of radiation.

Let us consider a mixture of two isotopes, having the shares ***N*** and 1-***N***, which enter a separation device (fig.3.3).

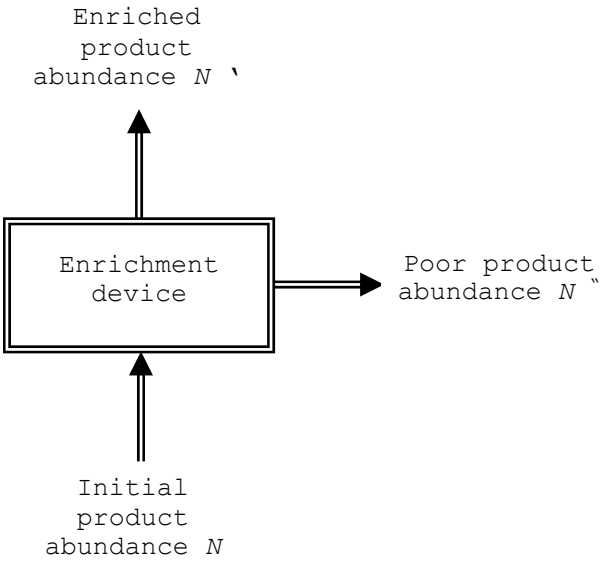

Fig.3.3 An isotope separation step

The abundance of the isotope with the share ***N*** is:

$$R = \frac{N}{1 - N}$$

The separation factor of the device is defined as:

$$\alpha = \frac{R'}{R}.$$

If the enriched product amount is ***f*** (p.u. from the initial product), the poor product share will be ***1-f***. The material conservation equation for this separation device is:

$$N = fN' + (1 - f)N''.$$

Because the mass number difference between the uranium isotopes is only 1, the separation factor is small enough, however depending on the separation method. To obtain the desired enrichment, the enrichment operation must be repeated more times. The technique is to connect the required number of separation devices in series, to form a cascade enrichment installation.

The material flow has two components:

- a direct one which move forward until the final enriched product and
- a reverse one which move backwards, in order to recover the fissile isotope.

In this respect, the enrichment installation has two sections (fig. 3.4), each one having more separation devices. The final residuum contains however fissile isotope at the lowest concentration possible for the separation method.

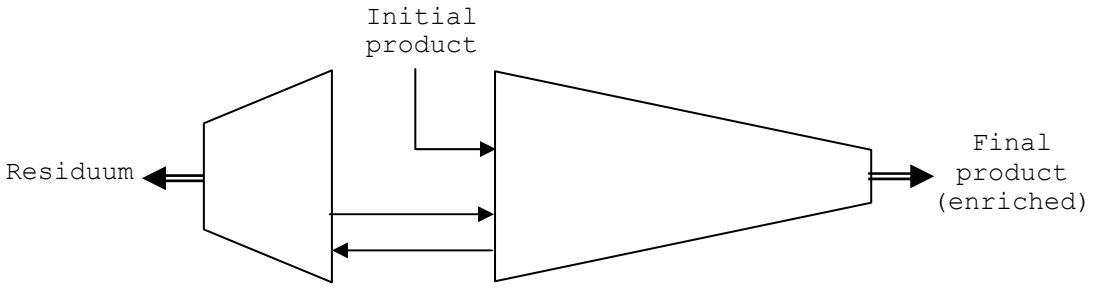

Fig. 3.4 Enrichment installation diagram

Enrichment methods

a) Gaseous diffusion

Consider a gaseous mixture with two components, having different molecular masses, in a closed vessel having a porous wall. Beyond the porous wall the mixture will content a higher share of the lighter gas. This fact explains by the average thermal speed of the molecules, higher for the lighter gas. Owing to the different speed, lighter molecules will often cross the wall by pores than the heavier ones.

The separation factor, α, defined on the basis of the kinetic theory of gases is:

$$\alpha = \sqrt{\frac{molecular\ mas\ of\ {}^{238}_{92}UF_6}{molecular\ mass\ of\ {}^{235}_{92}UF_6}} = 1.0043$$

The practical obtained value of the separation factor hardly reaches 50% of the theoretical one. In this situation, a great number of separation steps may be necessary to obtain the desired enrichment level. For example, to prepare the enriched uranium containing 90 % fissile isotope (for nuclear bomb) the enrichment cascade must have over 3000 steps.

The diffusion method was the first utilized to enrich uranium for military use during the Second World War. Some great installation are still operational today, having enough capacity to cover all the world demand for enriched uranium both for civil and military use.

b) Centrifugal separation

If the gas mixture considered above is introduced into a cylindrical vessel which fast rotates, the heavier molecules move to the vessel wall while the lighter ones remains mostly near the axe. Depending on the rotation speed, the centrifugal force will separate the gas molecules more or less.

The theoretical separation factor is

$$\alpha = e^{\frac{(M_1 - M_2)\omega^2 r^2}{2RT}},$$

where M_1 and M_2 are the molecular masses of $^{238}_{92}UF_6$ and $^{235}_{92}UF_6$ respectively, ω is the angular speed of the rotation and r is the vessel radius. R is the perfect gas constant and T is the absolute temperature of the process. Depending on the rotational speed, the separation factor α may reach values of 1.2 – 1.5. As consequence, the number of separation devices connected in series will be much lower for the same enrichment degree than for the diffusion method. For this reason the investment costs will be lower too.

c) The laser method is based on the radiation absorption of the gas molecules, having as result a perturbation of its energy equilibrium. The laser radiation makes possible the fine control of the photons energy, according to the highest absorption power of the selected molecules. No other source of photons can do that.

The UF_6 mixture is exposed to a laser radiation which is mostly absorbed by the $^{235}_{92}UF_6$ molecules. When the molecule equilibrium is enough damaged, it dissociates in $^{235}_{92}UF_5$ and atomic *F*. The main method advantage is that $^{235}_{92}UF_5$ is a solid substance at the process temperature. It can be easy separated by filtration.

This method reaches the highest separation factor. As consequence, the residuum is very poor in U-235, enabling to extract the fissile isotope from the residuum of the other methods.

3.8 Fabrication of fuel rods

The natural or enriched uranium must be prepared in an adequate form to be introduced in the reactor. The preparation concerns three aspects: the chemical formula, the form and dimensions and the protection cover.

From the beginning of the atomic era until now, three chemical forms of uranium (natural or enriched) were utilized. Chronologically, these forms were: metal (U), uranium dioxide (UO_2) and uranium carbide (UC).

Metallic uranium was utilized in the first research reactor (Fermi 1942) and afterwards in the first type of civil reactor (GCR). The operation of these reactors demonstrates some major inconvenient of the metallic shape:

- The melting temperature, 1133^0C, is too low to make possible a high enough efficiency of the reactor. The fuel temperature is raising from the cover to the middle of the fuel rod.
- The crystallization network of the metallic uranium change the system (allotropic deviation) with the raising temperature, lowering the mechanical properties of the fuel.

- The gaseous fission products cannot escape from the metal which get out of the initial shape by inflation.

For the next types of reactors was adopted the uranium dioxide. Its physical properties are more convenient for utilization in reactors:

- The melting temperature is about 2800^0C, what is high enough for the existing reactors;
- It may be compressed from powder, maintaining empty micro holes where the gaseous product of fission may be stored.

However, the uranium dioxide has a poor thermal conductivity which imposes some limits to the conduction path length of the heat until the cooling agent.

The uranium carbide presents even a better thermal behavior than the dioxide. It is recommended for the high temperature reactors (HTGR) where no metallic protection cover does resist.

Concerning the **physical shape** of the fuel, the main constraint is the good evacuation of the heat from the fuel to the cooling agent. As mentioned before, the thermal conductivity of the fuel is poor. By this reason the fuel is shaped in cylindrical or prismatic pelets with transversal dimension of some few centimeters. Because such a thin shape cannot have a great length, the fuel is prepared in pieces of some few centimeters.

For the uranium carbide, the spherical form is preferred from reasons of mechanical strength. This sphere has a diameter of some 5 cm and is covered with a layer of silicium carbide which has a very high mechanical strength.

The ***protection cover*** is necessary to avoid the dissipation of the fuel or fission products because the erosion action of the cooling agent. The fuel little pieces are introduced into cylindrical or prismatic metallic tubes which are tight closed by soldering. The material for the protection tubes must present some properties:

- Inertia to the chemical action of the fuel and cooling agent as well as adequate mechanical strength at the core temperature.
- Transparency and lack of affinity for neutrons in order to not weaken the neutron flow.

Some variants were utilized for the protection material. For the beginning, an aluminium-magnesium alloy (magnox), which can withstand temperature till to 450^0C, was used. For higher temperature, up to 600^0C, the stainless steel proved good behavior. Another solution, widely used today is the zirconium allied with Sn, Fe and Cr, called „zircalloy”. Its main property is a high transparency to neutrons, while the temperature limit is similar to the stainless steel.

The basic component of the fuel is the so called „needle” or „pencil”. That is a tube of at the most 1 m length and 0.5 mm wall thickness, filled with fuel pelets. More some needles are assembled in a bundle having a circular or rectangular

transversal section, according to the shape of fuel channels in the reactor vessel. A channel will be filled with more bundles.

For example, the fuel needle for the CANDU reactor has a diameter of 13 mm and a length of 0.5 m. A cylindrical bundle consists of 37 needles disposed on three layers. A fuel channel is filled with 8 bundles.

3.9 The spent fuel processing

The burning period of the fuel ends when the density of fissile atoms lowers under a limit able to maintain the chain reaction. The composition of the spent fuel consists mainly of U-238 which was not converted into Pu, residual U-235, isotopes of Pu and the fission products.

The fission products are very different, having a common feature: all are radioactive with half-life duration from some few hours to hundreds or thousands of years. The protection material becomes radioactive too and cannot be recycled.

The fuel processing means the separation of the U and Pu isotopes from the spent fuel. The U and Pu isotopes will be recycled, while the others components will be stored for ever.

The spent fuel processing develops in the some stages:

- *„Cooling"* the fuel after extraction from reactor. This operation means to keep the spent fuel a long period of time in a pool, immersed in water, until its radioactivity diminishes under a level which makes possible the next stages. The cooling period may reach several months, even more than a year.
- Cutting the bundles in small pieces of some few centimeter lengths.
- Immersing the cut fuel in a corrosive agent (hot HNO_3) where the fuel dissolves and the protection material falls at the vessel bottom. At the same time, the gaseous fission products go out from the solution and must be captured. The structure material can be separated and goes to be stored.
- Chemical treatment of the solution with a selective substance which reacts with the uranium and plutonium, extracting them from the acid solution. The remaining liquid contains the fission products. It will be dehydrated and prepared for long time storage.
- Chemical separation of U and Pu isotopes and making its ready for recycling.

3.10 Nuclear reactors

The nuclear reactors which operate today for electricity generation, differ by the fuel, moderator and cooling agent. Many solutions were tested or even operated, but only few of them proved technical feasibility and economical effectiveness.

a) Fuel:

- natural uranium,
- enriched uranium,

- mixed fuel U-Pu or U-Th.

b) Moderator:
  - graphite, C
  - light water, H_2O
  - heavy water, D_2O.

c) Cooling agent:
  - gas, CO_2, He
  - light water
  - heavy water
  - liquid salts
  - melted metal

The fuel type and the neutrons energy are the ultimate factors for selection of the moderator and the cooling agent.

For example if the fuel is natural uranium, the light water cannot be used as moderator or cooling agent because its affinity for neutrons is too high. If the fissions are performed by fast neutrons, the light and heavy water cannot be utilized as cooling agent to not slow down the neutrons.

Graphite moderated reactors

The graphite was the first used moderator. It was selected for the first high power reactors installed during the Second World War in USA to obtain Pu-239 for military purpose. The graphite allows a higher conversion factor than others substances.

From the 1960's begun the era of nuclear civil power in England and French with the GCR type (**G**as **C**ooled **G**raphite Moderated Reactor). Its fuel was the natural metallic uranium protected with Magnox alloy, while the cooling agent was CO_2. The burning rate was of only 3500-4500 MWd/t. The overall efficiency of the nuclear power plant was of 18-20% much lower than that of the thermal plants. This type of reactors is now decommissioned.

For the next type, AGR (**A**dvanced **G**raphite Moderated **R**eactor), the fuel was light enriched uranium (2%) and the protection material was stainless steel. The burning rate mounted to some 18000 MWd/t, and the fresh steam parameters in the turbine inlet line were up to 530^0C and 160 bar. As consequence, the plant efficiency became close to that of thermal plants.

Unfortunately, although the technical performances of this type were good, from costs reason it was not developed beyond some reactors in England.

The graphite reactor cooled with boiling light water, was developed by the former USSR (LWGR or RBMK) to 1000 MWe rated power. This type was involved in a serious accident (Tchernobil 1986) and as consequence no new reactors of this type were commissioned. The Tchernobil accident stopped the development of the graphite reactors.

However, the high temperature reactor (HTGR) may have a promising future owing to its two main features:

- The cooling agent temperature may go beyond 800^0C, making this reactor of interest for process heat in chemistry, construction materials or metallurgy.
- The highest conversion factor, *c*, among all types of thermal reactors, meaning the highest Pu production.

Heavy water moderated reactors (HWR)

Heavy water is the best moderator because it absorb the most few neutrons by comparison with the others moderators. As consequence, the heavy water allows to use the natural uranium as fuel without the disadvantages of the graphite. There are two main types:

- PHWR – Pressurized Heavy Water Cooled and Moderated Reactor. The pressure in the cooling circuit must be high enough to not allow the vaporization of the heavy water. This type of reactor is installed at Cernavoda plant (2 CANDU units of 700 MWe). The fuel is natural uranium and the burning rate reaches 8000-10000 MWd/t.
- BHWR – Boiling Water Cooled and Moderated Reactor. The cooling heavy water vaporizes before leaving the reactor core. This type is little developed.

Light water moderated reactors (LWR)

The use of light water as moderator or cooling agent imposes the enriched uranium as fuel.

The first utilization of this type of reactor was a military one for the propulsion of submarines and afterwards for surface great ships. The light water has the most powerful moderation action and combined with high enriched uranium allows the smallest dimensions of a reactor. Also, the burning rate of these reactors may reach 30,000-35,000 MWd/t. Benefiting by the experience of the military reactor, the introduction of the light water reactors was promoted by the great military power USA and the former UdSSR. Today, this category of reactors occupies the dominant position among the civil reactors.

There are two main types of light water reactors:

- PWR – **P**ressurized Light **W**ater Cooled and Moderated **R**eactor
- BWR – **B**oiling Light **W**ater Cooled and Moderated **R**eactor.

The PWR type is preferred owing to its smaller size for the same rated power. The operating regime is easier controlled because the cooling agent remains liquid after crossing the core of the reactor.

The BWR type has the advantage of a smaller power consumption to circulate the cooling agent. Because the high latent heat of water vaporization, a quantity of water which boils in reactor extracts more heat that the same quantity of water from a pressurized reactor.

Fast breeder reactor (FBR)

The chain reaction is maintained by the fast neutrons, so the moderator must not to be present. Having a conversion factor $c > 1$, this type of reactors makes possible the conversion of the total amount of the fertile isotope U-238, that is over 99% of the natural uranium, as well as the isotope Th-232.

The nuclear fuel is strongly enriched, over 15-20%. The efficient use of the secondary neutrons can be made by surrounding the reactor core with a layer of natural uranium where take place the conversion in Pu-239.

Concerning the cooling agent, the most tested solution is the melted sodium. The main advantage is the great temperature gap between the melting level (90^0C) and the boiling one (1500^0C) what allows a little higher pressure than the atmospheric one in the cooling circuit. However, the melting temperature of 90^0C, impose to maintain a high enough temperature of the entire cooling circuit in order to avoid the sodium to turn into solid state.

Despite the great importance of the fast breeder reactor for the future of the nuclear power, its emergence in commercial use is still expected. The financial reasons and technical difficulties as well as the consequences of two serious accidents have delayed its development.

3.11 The nuclear power in present and near future

Nowadays, 31 countries have 437 nuclear reactors in operation (table 3.3). This development was strongly encouraged by the events on the energy market in 1973. The possible exhaustion of the fossil fuels has determined a steep growth of the oil price. One of the available energy source to replace the fuel was the nuclear technology because the reactor's rated power was much higher than those based on renewable.

Table 3.3 Nuclear Energy: capacity, generation and operating experience at 1 January 2010

Country	Reactors in operation		Net generation in 2009 generation TWh	Nuclear share of electricity generation in 2009
	Units number	Capacity MWe		
1.South Africa	2	1 800	11.6	4.8
2.Canada	18	12 577	85.1	14.8
3.Mexico	2	1 300	10.1	4.8
4.U S A	104	100 683	796.9	20.2
Total North America	**124**	**114 560**	**892.1**	
5. Argentina	2	935	7.6	6.9
6. Brazil	2	1 766	12.2	2.9
Total South America	**4**	**2 701**	**19.8**	
7. Armenia	1	376	2.3	45.0
8. China	11	8 438	65.7	1.9
9. India	18	3 984	14.7	2.2
10. Japan	54	46 823	263.1	28.9
11. Korea (Republic)	20	17 647	141.1	34.8
12. Pakistan	2	425	2.6	2.7

13. Taiwan, China	6	4 949	39.9	20.7
Total Asia	**112**	**82 642**	**529.4**	**2**
14. Belgium	7	5 863	45.0	51.7
15. Bulgaria	2	1 906	14.2	35.9
16. Czech Republic	6	3 678	25.7	33.8
17. Finland	4	2 696	22.6	32.9
18. France	59	63 260	391.8	75.2
19. Germany	17	20 470	127.7	26.1
20. Hungary	4	1 859	14.3	43.0
21. Lithuania			10.0	76.2
22. Netherlands	1	482	4.0	3.7
23. Romania	**2**	**1 300**	**10.8**	**20.6**
24. Russian Federation	31	21 743	152.8	17.8
25. Slovakia	4	1 711	13.1	53.5
26. Slovenia	1	666	5.5	37.8
27. Spain	8	7 450	50.6	17.5
28. Sweden	10	8 958	50.0	37.4
29. Switzerland	5	3 238	26.3	39.5
30. Ukraine	15	13 107	77.9	48.6
31. United Kingdom	19	10 097	62.9	17.9
Total Europe	**195**	**168 484**	**1 105.2**	
TOTAL WORLD	**437**	**370 187**	**2 558.1**	**14**

After the year 1986 when has produced a very serious damage of a nuclear reactor (Tchernobil, Ukraine) the demand for new reactor has decreased. A relaxation of oil price took place in the same period.

From year 2000, the oil price has begun a new climb, while the "global warming" became more evident. As consequence, the nuclear option was revaluated and new reactors were ordered, despite the opposition of the ecologists.

Unfortunately, in 2011 a more serious damage of a nuclear plant took place in Japan (Fukuoka) because an earth quake followed by a "tsunami".

Actually, 55 new reactors are under construction, the most of them in Asia. Romania has a project to install 2 new reactors at Cernavoda, which will double the installed power of the plant.

A major advantage of the nuclear energy is the price of kWh, situated on the second place after the hydropower. By comparison with the price of 1 kWh generated in thermo power plant, the nuclear kWh is 2 times lower in Germany, 1.8 times in France etc.

The contribution of the nuclear power to the world electricity has reached, in 2009, 14%, close to the hydro energy share. In some countries like France, Belgium, Slovakia, Ukraine, the nuclear electricity share is higher or close to 50%.

One of the technical performances of an electricity source is the reliability. It can be expressed by the percent duration of normal operating time by the comparison with the planned one. The nuclear plants have become, step by step, more reliable, thus being now at least similar with the other power plants.

A strong argument in the favor of nuclear energy is the absence of greenhouse gases emissions during the operation.

3.12 Environmental aspects

The main environmental problem of the nuclear energy represents the radioactive wastes. Some wastes with low or medium level of radioactivity results from all the stages of the nuclear fuel cycle. High level of radioactivity present the products of fission and the protection case of the spent fuel.

All these wastes remain radioactive for long periods of time, until hundreds and thousands of years. No method to reduce the natural or artificial radioactivity was discovered yet. The unique solution is the storage of these wastes in remote areas, far from the people and fauna access. For high level wastes, the underground storage is considered the safest solution.

The danger of radioactivity is strongly increased if a nuclear accident happens. Unfortunately, the fact was twice times demonstrated.

In normal operation regime, the nuclear plants are quite safe and do not affect the population in the neighboring zone. However, the nuclear plant must to be located at a security distance from the human settlements. At the same time, the radioactive inhalation of the people as far as some kilometers distance away from the plant have to be examined periodically.

Excess mass- defect de masă
Binding force -forţă de legătură
reactor core – zona activă a reactorului
efficacious section – secţiune eficace(efectivă)
radioactive half-life – durată de înjumătăţire