1 Nuclear fission energy

[K. Kugeler, E. Kugeler, N. Pöppe, Z. Alkan, W. Grätz]

1.1 Principles of fission reactors

1.1.1 The fission process

The conversion of mass into energy and vice versa follows Einstein's well-known relation,

$$\Delta E = \Delta m \cdot c^2 , \tag{1.1}$$

where Δm is the change of mass, ΔE the change of energy, and c the velocity of light (2.998×10^8 m/s). Therefore the conversion of 1 g of matter is correlated to the release of an amount of energy of 9×10^{13} J. Related to the usual unit of masses in nuclear physics (1 atomic mass unit = 1 amu = 1.66×10^{-24} g) the conversion of 1 amu into energy corresponds to 931 MeV (1 eV = 1.602×10^{-19} J). Table 1.1 contains data of particles that are important in reactor physics.

Details of the fission process and relevant aspects of nuclear physics as well as much more details of reactor physics are presented in a famous, broad literature. For further information, the reader is referred to [58Wei, 61Gla, 63Gla, 66Wil, 83Zie, 60Dre, 70Bel, 82Eme, 64Tai, 58Eth, 64Mee, 81Ben, 85Wil, 71Fra, 73Fos, 83Lam, 92Bri, 83Kes, 92Kni, 74Old, 53Cas, 76Smi].

Table 1.1. Data of nucleons and electrons.

Particle	Symbol	Mass [g]	Mass [amu]	Charge [As]
neutron	n	1.6748×10^{-22}	1.008665	—
proton	p	1.6725×10^{-24}	1.007277	$+1.602 \times 10^{-19}$
electron	e	9.1080×10^{-28}	0.00055	-1.602×10^{-19}

The nucleons are bound in the nucleus with binding energies which depend on the mass number, $A = N + Z$, of the nucleus. The binding energy, E_B, is related to the proton mass, M_p, the neutron mass, M_n, and the mass of the whole nucleus, M_A, via

$$E_B = \Delta m \cdot c^2 = (Z \cdot M_p + N \cdot M_n - M_A) \cdot c^2 , \tag{1.2}$$

where Δm denotes the mass defect. Figure 1.1 shows the dependence of the binding energy per nucleon on the mass number. This curve can be explained by the liquid-drop model of the nucleus, in which a typical relation for the binding energy per nucleon (E_B/A, in MeV/nucleon) is:

$$\frac{E_B}{A} = 14 - \frac{13}{A^{1/3}} - \frac{0.585 \cdot Z^2}{A^{4/3}} - \frac{19.3 \cdot (A - 2Z)^2}{A^2} + \delta,$$

with
$\delta = +33/A^{7/4}$ for Z even, N even,
$\delta = -33/A^{7/4}$ for Z odd, N odd, (1.3)
$\delta = 0$ for Z even, N odd,
$\delta = 0$ for Z odd, N even.

The first term corresponds to the energy caused by the nuclear forces, the second is a surface correction, the third is the Coulomb repulsion inside the nucleus, the fourth term corresponds to the symmetry of nucleons in the nucleus, whereas the fifth term represents an addition of energy for the last unpaired particle.

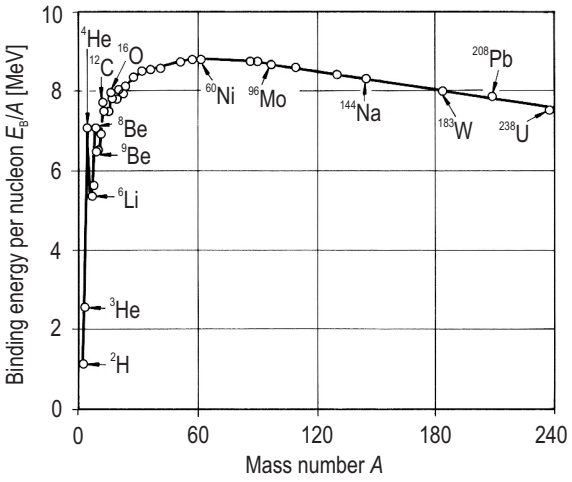


Fig. 1.1. Average binding energy per nucleon as a function of the mass number.

At low mass numbers there are some nuclei with small local maxima of binding energy. These nuclei have relatively high binding tendency and are very stable. A flat maximum of the curve occurs at $A \approx 60$. Therefore two methods for producing energy by exothermal nuclear reactions are principally feasible:

– fusion of light nuclei,
– fission of heavy nuclei.

In addition, the radioactive decay of unstable isotopes delivers energy, too. Nuclear fission occurs after absorption of a neutron by fissile nuclei of U-235, U-233, Pu-239 or Pu-241. The fission reaction, e.g. for U-235, can be described by the expression

$$^{235}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{236}_{92}\mathrm{U}^{*} \rightarrow {}^{A_1}_{Z_1}\mathrm{X}_1 + {}^{A_2}_{Z_2}\mathrm{X}_2 + \nu \cdot {}^{1}_{0}\mathrm{n} + Q\,. \tag{1.4}$$

Two fission products (X_1, X_2) and ν neutrons are released. The reaction occurs with an intermediate excited state of U-236. For the nucleons, conservation equations are valid. The number of nucleons and the charge number must be conserved:

$$236 = A_1 + A_2 + \nu\,, \tag{1.5}$$

$$92 = Z_1 + Z_2\,. \tag{1.6}$$

The release of energy during fission can be estimated from the curve of binding energy. For the U-235 nucleus, the average binding energy is around 7.6 MeV per nucleon, for fission products ($A \approx 80$ to 150) the average binding energy results to nearly 8.45 MeV per nucleon. The difference of 0.85 MeV per nucleon is released in form of energy during fission. For the U-235 nucleus this means around 200 MeV per fission event,

$$^{235}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{137}_{56}\mathrm{Ba} + {}^{97}_{36}\mathrm{Kr} + 2 \cdot {}^{1}_{0}\mathrm{n} + \mathrm{energy}\,. \tag{1.7}$$

There is a mass difference of $\Delta m = 0.2043$ amu corresponding to a release of energy of 190 MeV (see Table 1.2 for a detailed energy balance). As an average value in calculations in nuclear technology a value of 200 MeV per fission is assumed today. This energy is distributed on the different reaction products as listed in Table 1.3.

Table 1.2. Detailed mass balance in the fission process of U-235.

	Protons	Neutrons	Mass [amu]
Before fission			
U-235	92	143	235.1167
n	0	1	1.0090
total	92	144	236.1257
After fission			
Ba-137	56	81	136.9514
Kr-97	36	61	96.9520
2 n	0	2	2.0180
total	92	144	235.9214

Table 1.3. Distribution of energy during the fission process of U-235.

Form of energy	Value [MeV]	Proportion [%]	Remark
kinetic energy of fission products	174	83	used inside reactor core
kinetic energy of prompt neutrons	5	2	used inside reactor core
prompt γ-radiation	8	4	partly used in reactor core
β^--decay of fission products	7	3	partly used in reactor core
delayed γ-decay of fission products	6	3	partly used in reactor core
neutrinos	10	5	not usable

The main part of energy immediately after fission occurs as kinetic energy of the fission products and is converted into thermal energy in the reactor core.

The kinetic energy of the prompt neutrons and a part of the energy of the promptly produced γ-radiation, as well as parts of the decay energy of β^-- and γ-radiating fission products remain inside the reactor core and are used for the generation of heat and finally of electricity in power reactors.

Parts of the energy of β^-- and γ-decay of the fission products are released later even after shutdown of the nuclear chain reaction in the reactor. For many years a considerable amount of decay heat is set free, causing main questions of safety during operation of reactors and intermediate and final storage of high-level radioactive waste. The neutrino energy cannot be used in reactors. In addition to the values given in Table 1.2 there are some capture reactions caused by γ-radiation which result in an additional amount of energy of 5 MeV staying inside the core. Therefore, around 200 MeV per fission is a good practical value for application in fission technology. The energy production in fission is very large compared to that in chemical reactions such as burning processes of fossil fuel. The reaction of one carbon atom with oxygen yields about 4 eV.

A comparison of fuel required for equivalent energy production between burning carbon and fissioning uranium results in a factor C,

$$C = \frac{200\,\mathrm{MeV/atom}}{235\,\mathrm{g/mol}} \Big/ \left(\frac{4\,\mathrm{eV/atom}}{12\,\mathrm{g/mol}}\right) = 2.55 \times 10^6 \,. \tag{1.8}$$

This feature is one of the big advantages to use nuclear energy, a very small amount of material is necessary to supply even very large power plants.

In the commercial power reactors introduced today low-enriched uranium of around 4 % U-235 is used. A simple estimate gives the following number: 1 g U-235 contains 2.56×10^{21} nuclei. The total fission of 1 g releases an energy of 7.88×10^{10} J. Therefore a useful relation is:

$$1\,\text{g U-235} \;\hat{=}\; 1\,\text{MW d}\;(= 2.4 \times 10^4\,\text{kWh})\,. \tag{1.9}$$

Natural uranium (containing 0.71 % U-235) delivers 170 kWh/g, whereas uranium enriched to 4 % allows an energy production of around 10^3 kWh/g. In a real reactor there occurs breeding of fissile plutonium-239 during operation, and this material is partly fissioned in situ. This effects somewhat changes the numbers given above.

During the fission process neutrons are produced, too, which are necessary for the chain reaction. The number of neutrons per fission depends on the type of fuel and on the energy of the neutrons which induce the fission process (see Fig. 1.2a). The neutrons have an energy distribution which is similar to a Maxwellian distribution with a maximum at 0.7 MeV and an average value at around 2 MeV (see Fig. 1.2b).

The fission products are generated with a distribution of mass numbers, often named the fission yield, which shows maximum values at $A \approx 100$ and $A \approx 140$ (see Fig. 1.2c). The shape of the distribution differs depending on the type of fuel and the energy of the neutrons which induce the fission process.

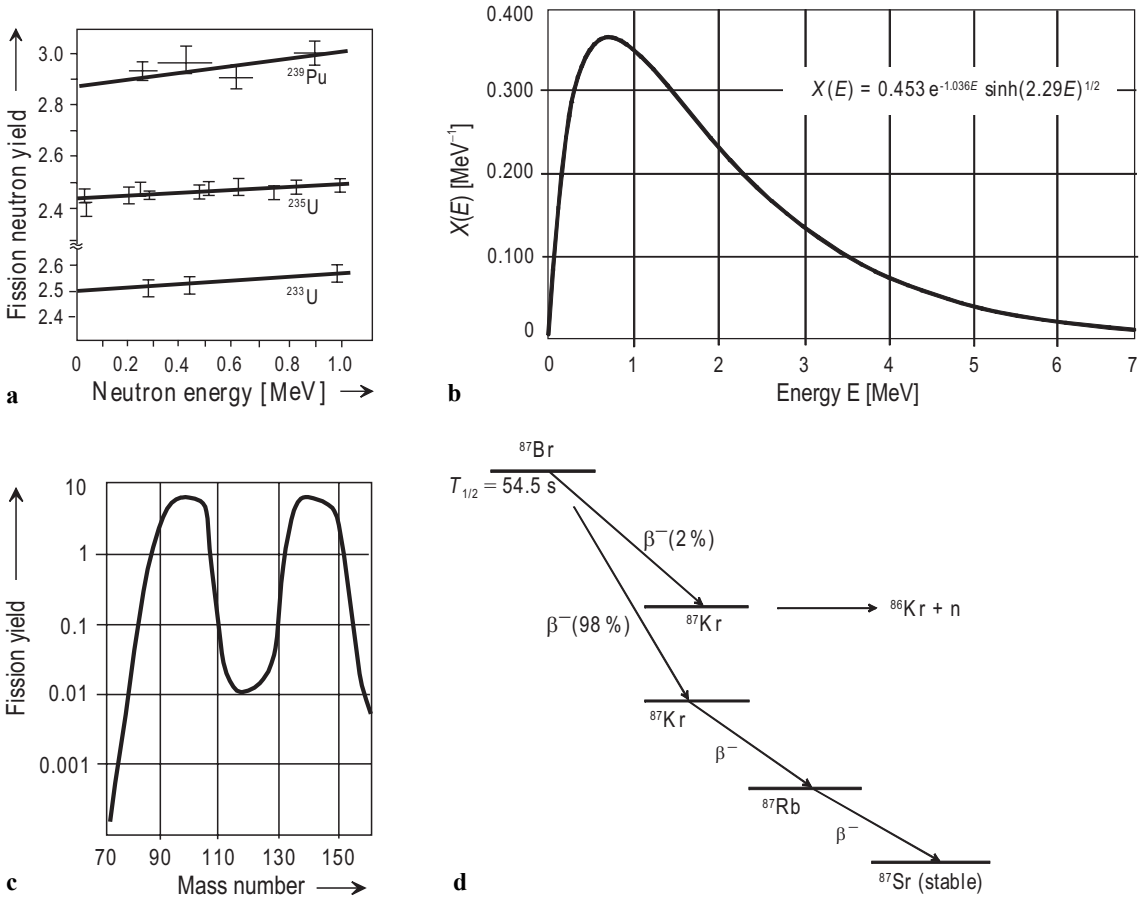


Fig. 1.2. Some aspects of the fission reaction, **(a)** number of neutrons in fission dependent on energy and fissionable material, **(b)** energy spectrum of prompt fission neutrons (U-235), **(c)** fission yield of U-235, **(d)** emission of delayed neutrons (example of Br-87 as emitter).

Nearly all neutrons (99.35 % in U-235 fission) are generated immediately during fission (after 10^{-12} s). Beside these prompt neutrons there are some delayed neutrons produced during the total fission process. The delayed neutrons are emitted from some nuclides as bromium or iodine, as explained in Fig. 1.2d. The fission product Br-87 carries an excess of neutrons and energy. There is a probability that the decay occurs by a β^--decay with the emission of a neutron from the excited state of Kr-87. This process occurs like an emission of a neutron with a half-life of 54.5 s from the fission process for this specific example. There are other fission products with this behavior, too, as explained in Sect. 1.1.7.5. The delayed neutrons are fundamentally important for the control of the fission process. Without these neutrons the characteristic times to control reactors would be far too short for technical actions.

The delayed emission of β^-- and γ-radiation from the radioactive fission products causes the decay heat. Figure 1.3 contains the curve of the decay heat related to nominal thermal reactor power dependent on time after shutdown, t, and on the time of operation before shutdown, t_o. This curve indicates that immediately after shutdown of the reactor around 6 % of the nominal plant power are produced as decay heat, after 1 hour around 1 % and even after years parts of per mill of power. The safe removal of decay heat from the fuel elements is one of the key questions of safety in nuclear technology, not only in the reactor, but in the intermediate and final storage of radioactive waste, too.

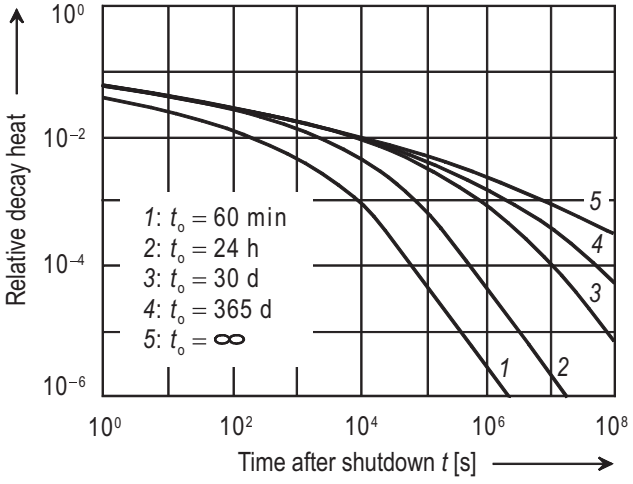


Fig. 1.3. Decay heat production related to the thermal reactor power dependent on the time after shutdown, t, and on the operation time before shutdown, t_o.

1.1.2 The controlled chain reaction and the critical arrangement

When generating energy from nuclear fission, it is essential that a controlled chain reaction takes place within the reactor system. In total, the number of neutrons of successive generations has to stay constant. According to Fig. 1.4 this means that in a finite reactor system a relation of the following kind must be applicable for the criticality constant k:

$$k = \frac{P}{A+L} = 1, \quad P = A + L, \tag{1.10}$$

where P denotes the neutron production, A the total absorption and L the leakage of neutrons from the system. All neutrons which are produced in the system are finally absorbed or leak out. If there are no neutron losses due to leakages in an infinite system, the following equation can be applied:

$$k_\infty = \frac{P}{A} = 1, \quad P = A, \tag{1.11}$$

which indicates that all neutrons produced disappear just by absorption.

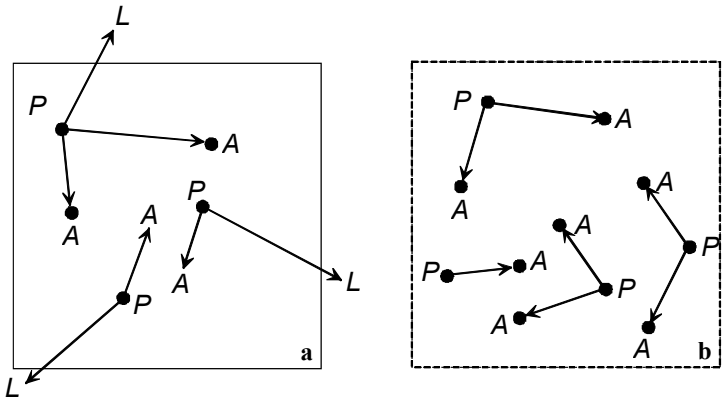


Fig. 1.4. Neutron fate in reactor system: **(a)** finite system, **(b)** infinite system.

With every fission, 2 to 3 neutrons are emitted, which again can cause further fission. Thus, in a system containing fissionable material, it is possible to run either an uncontrolled or a controlled chain reaction.

If l stands for the average time between two successive neutron generations, the following equation applies to the change of the amount of neutrons in an infinite system:

$$\frac{\mathrm{d}n}{\mathrm{d}t} = \frac{n(k-1)}{l} = n\frac{\Delta k}{l}. \tag{1.12}$$

The variable k is, as defined above, meant to mark the multiplication factor for the number of neutrons; its size is determined by the neutron fate inside the reactor, as will be explained in more detail in the following. The solution to the simple differential equation is:

$$n(t) = n(0)\cdot\exp\left(\frac{\Delta k\cdot t}{l}\right). \tag{1.13}$$

This means that, if $\Delta k > 0$ is chosen, the number of neutrons will increase exponentially from generation to generation. If Δk remains greater than zero for an extended period of time, the number of neutrons and therefore the power output will rise unlimitedly. This is the case with uncontrolled chain reactions, i.e. the nuclear bomb. If, however, with the appropriate reactor design $\Delta k = 0$ is kept, it is a controlled chain reaction. The power output remains constant and the reactor is called critical. Control and shutdown procedures are carried out by small deviations from the set value. The number of neutrons, n, is proportional to the power output generated. Figure 1.5 shows the discussed cases in comparison. A subcritical system with $\Delta k < 0$ is included, too.

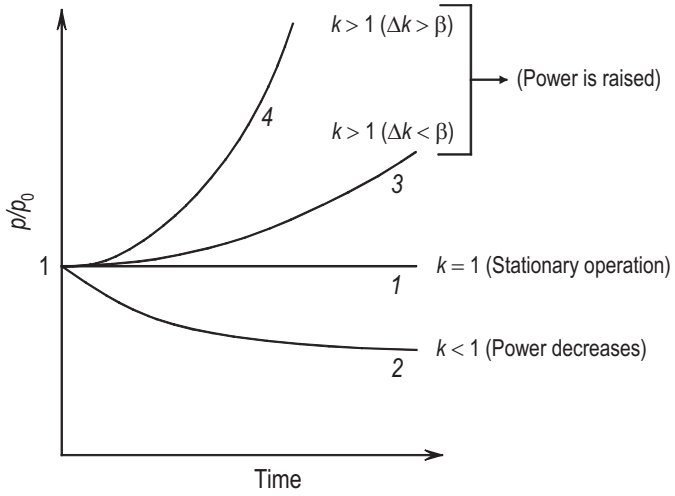


Fig. 1.5. Qualitative dependence of power output on time (parameter: criticality value k, without temperature feedback); β denotes the fraction of delayed neutrons, $\beta = 0.65$ % for ^{235}U as fissionable material.

Those neutrons that have emerged from fission experience different fates when reacting with the nuclei inside the reactor. Apart from fission events due to interaction with U-235 nuclei, elastic and inelastic scattering as well as absorptions are possible. This applies to fission materials, breeding materials, moderators, cooling medium, structure materials and absorbers in shutdown elements. The probability of such reactions is characterized by so-called cross sections, σ. The variable σ makes it possible to describe a certain reaction; a more detailed explanation will follow in Sect. 1.1.4.

A description of the neutrons' behavior inside the reactor can be given with the four-factor formula shown in Fig. 1.6. Assuming that the cycle starts with 100 thermal neutrons, a fraction $f \cdot 100$ is absorbed in the fuel. The rest, $(1-f) \cdot 100$, however, is absorbed in other materials of the reactor. A fraction $(\sigma_a^f / \sigma_a^t) \cdot f \cdot 100$ leads to fission, from which another $(\sigma_a^f / \sigma_a^t) \cdot f \cdot \nu \cdot 100$ new fission neutrons with high energies, called fast neutrons, are produced. Here the ratio σ_a^f / σ_a^t characterizes the amount of absorption by fission related to the total absorption in U-235. Through the additional fast fission of U-238 this number of fast neutrons is increased to $\varepsilon \cdot (\sigma_a^f / \sigma_a^t) \cdot \nu \cdot f \cdot 100$. The quantity $\eta = (\sigma_a^f / \sigma_a^t) \cdot \nu$ is regarded as the neutron recovery per reaction. Therefore $\varepsilon \cdot \eta \cdot f \cdot 100$ neutrons pass through the area of slowing down. With a probability of $(1 - w_1)$ this leads to an escape out of the reactor system in the fast part of the neutron spectrum. In addition, another fraction $(1 - p)$ is lost by absorption in the resonances of the U-238 which is arranged in every reactor using low-enriched uranium. Accordingly, after passing through the resonance area, $w_1 \cdot p \cdot \eta \cdot f \cdot \varepsilon \cdot 100$ neutrons are still available. The now thermal neutrons will be lost, with a probability of $(1 - w_2)$, through leakage out of the reactor with the diffusion now setting in. After running through this cycle, $w_1 \cdot w_2 \cdot p \cdot \eta \cdot f \cdot \varepsilon \cdot 100$ neutrons are available for further fission. In a critical reactor this will be just about 100 neutrons. This is the same amount of neutrons as started the cycle.

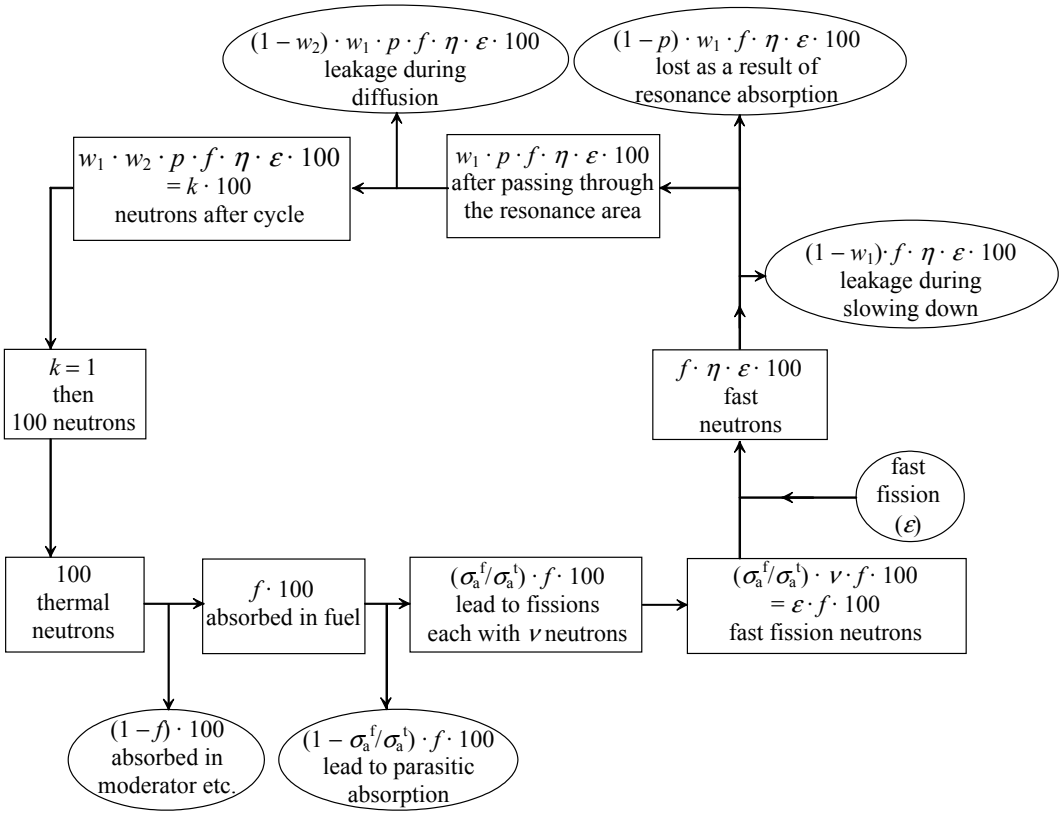


Fig. 1.6. Description of the neutron cycle in a reactor: finitely expanded medium according to the four-factor formula.

Therefore the so-called four-factor formula, principally describing the fate of neutrons in a thermal reactor, can be applied:

$$w_1 \cdot w_2 \cdot p \cdot \eta \cdot f \cdot \varepsilon = w_1 \cdot w_2 \cdot k_\infty = k = 1 . \tag{1.14}$$

The value k_∞ is called the multiplication constant of an infinite reactor and it always remains greater than 1 in a finite reactor system because $w_1 \cdot w_2 < 1$. With the factor $w_1 \cdot w_2$ the size and geometry of the finite reactor is determined. An outline of the terms used for k_∞ in the four-factor formula is given in Table 1.4.

An even critical reactor shows the value $k = 1$. In this case, the number of neutrons remains constant in each generation and hence the reactor power output stays constant in time as well.

Typical numerical values for the factors listed in Table 1.4 are given in Sect. 1.1.5 for the PWR. It is the task of reactor statics to determine these values for the stationary design, depending on fuel element design, core design, as well as on operation parameters.

Table 1.4. Terms in the four-factor formula (Z denotes the number of fissions).

Factor	Notation	Definition	Description
f	thermal utilization	$\frac{\sigma_a(\text{fuel})}{\sigma_a(\text{total})}$	$\frac{\text{absorption in fuel}}{\text{total absorption}}$
η	neutron recovery	$\frac{\nu \cdot \sigma_f(\text{fuel})}{\sigma_a(\text{fuel})}$	$\frac{\text{absorption for fission}}{\text{absorption in fuel}}$
ε	fast fission factor	$\frac{Z_{\text{therm}} + Z_{\text{fast}}}{Z_{\text{therm}}}$	$\frac{\text{total fissions}}{\text{thermal fissions}}$
p	resonance escape probability	$\frac{\sigma_a(\text{therm.})}{\sigma_a(\text{total})}$	$\frac{\text{thermal absorption}}{\text{total absorption}}$

The different parameters change in time in a very complex manner, in particular by the burn-up of fuel during operation as well as by the build-up of fission products, of new fissionable isotopes and of actinides. Therefore, computer codes employing a three-dimensional representation of the core and containing the time-dependent variation of isotopes in the reactor are used today to calculate the criticality state, described by k for the whole operation time. As mentioned above, prompt and delayed neutrons destine the neutronic behavior of a reactor.

At this point, the great practical and, particularly, the safety-engineering importance of power output changes as mentioned in Fig. 1.5, should be pointed out. Especially power output excursions, as shown in curve 4, have to be avoided as they can lead to the destruction of the reactor. This would happen, if the reactivity inside the reactor changed promptly, with Δk being greater than the number β of delayed neutrons involved. In this case, only the prompt neutrons determine the generation time, which would then be of the order of 10^{-5} s. It would be impossible to control the resulting increased power output by technical controlling systems, and hence, if taking place unlimitedly, this increase could destroy the reactor. In general, this can be avoided by inherent self-regulation mechanisms, i.e. negative temperature coefficient (as explained in Sect. 1.1.7.4) and well-suited design of reactors. The operating area of these reactivity values is shown in the curves 2 and 3. A negative value of Δk causes a decrease in power output, a positive one leads to a slow, exponential increase of power. With the help of technical systems, both these changes can be reliably carried out and controlled.

1.1.3 Principle of a nuclear reactor

In a nuclear power plant, mass and energy conversion proceeds as shown in Fig. 1.7a. Fissionable materials (U-235, U-233, Pu-239 and Pu-241) are used to generate energy by nuclear fission in the reactor. Following Einstein's relation, $\Delta E = \Delta m\, c^2$, mass is converted into energy. A cooling medium, which is led through the reactor core, carries the heat towards the steam generator. The steam usually supplies a steam turbine process, the turbine powers the generator that produces electric energy, which is then delivered to the electrical grid. For the future, the combination of nuclear reactors with gas turbine processes or combined cycles will also be realized, to gain higher efficiencies.

In the reactor, the fuel material is heated as a consequence of the fission process. From the fission process, two fission products and 2 to 3 neutrons as well as energy emerge. The fission products are radioactive, which leads to the possible damage of the environment in case of accidents, as will be discussed in more detail in the following. The released neutrons ensure the maintenance of the chain reaction, which guarantees the continuous operation of the reactor.

The core basically contains fuel elements, consisting of fuel and breeding materials as well as structure material, the moderator and the cooling agent. In thermal reactor systems moderators are used to slow down the neutrons from high velocities to lower ones. This is necessary because cross sections for fission in the thermal energy range are particularly high. Light and heavy water as well as graphite are suitable moderator materials that are also of current technical importance. In fast reactor systems, moderation is practically renounced completely to realize a fast neutron spectrum.

The heat that is released during nuclear fission is transferred out of the core area by the cooling agent. Suitable cooling agents are light and heavy water, CO_2 and helium for thermal rectors, as well as liquid metals such as sodium or lead for fast systems.

Nuclear reactors serve as heat sources in power plants to produce electricity, for powering ships, as material testing reactors, as research reactors for the production of neutrons as well as to produce new fission materials and special radioactive isotopes. Nuclear reactors can deliver nuclear process heat too. As an example cogeneration plants to produce electricity and district heat or process steam have been realized. High temperature processes using nuclear heat like hydrogen production by steam reforming of methane, coal gasification or thermochemical water splitting are further interesting options. In future, reactor systems are likely to be used as well for the conversion of long-lived radioactive isotopes. This could be important for reducing the storage time in final storage systems.

Today thermal reactors are of special importance in reactor technology. Fission heat is used inside the steam generator to power the subsequent turbine process (see Fig. 1.7c).

Figure 1.7b shows a basic scheme of a reactor core. It is indicated that the cooling medium is passing in the reactor and is heated up. In water cooled reactors, the cooling medium serves as a moderator at the same time. The figure contains a qualitative picture of temperature of coolant, the canning of fuel elements and fuel across the length of a fuel rod.

The reactor is controlled and shut down by shutdown rods that are inserted into the core. The primary circuit is arranged inside a reactor protection building or reactor containment. According to today's state of the art, in a PWR the primary system is connected to a steam turbine process. The power house, containing all parts of the turbine cycle, is attached to the reactor protection building. Nowadays, nuclear power plants are usually operated with large availability and under good economic conditions. The main question of reactor safety is the retention of fission products inside the plant, and the demand connected with this always to be able to cool and shut down the plant securely.

The spent fuel elements which contain, beside the fission products, long-lived fission products and actinides have to be stored reliably to avoid any contact of radioactive material with the environment. Initially this is done by a long-term intermediate storage of spent fuel elements. Then, after their final conditioning, the fuel elements can be stored in a final disposal place. Alternatively, in a reprocessing plant fission products, fuel and breeding material are separated. The fission products are then stored in glass containers (borosilicate glass blocks). After a long-time intermediate storage to reduce the decay heat, these glass containers are inserted into a final storage, namely salt mines, granite structures, or other suited geological repositories. The remaining fuel and the breeding material leaving the reprocessing plant are used to fabricate new fuel elements for the operation of nuclear reactors.

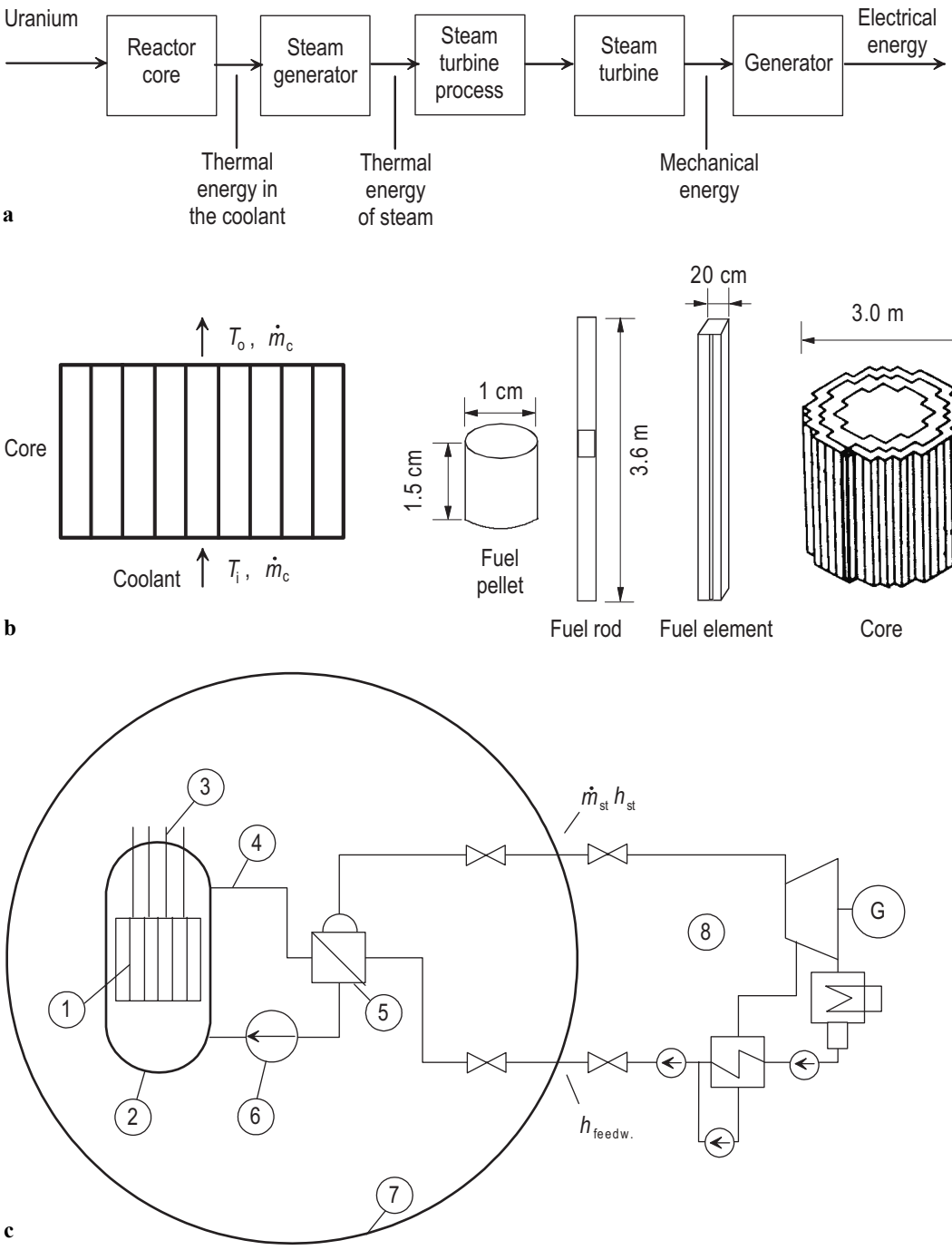


Fig. 1.7. Concept of nuclear reactors: **(a)** chain of conversion, **(b)** qualitative picture of heat removal from the core and of the components of a fuel element, **(c)** utilization of thermal fission energy for the production of electrical energy (1: reactor core, 2: reactor pressure vessel, 3: control and shutdown system, 4: coolant pipes, 5: steam generator, 6: coolant pump, 7: reactor containment, 8: steam turbine plant).

As far as energy transport in the whole system is concerned, the following equations hold: The thermal work during a span of one year is given by

$$A_{\mathrm{th}} = \int_0^{1\mathrm{a}} P_{\mathrm{th}}(t) \cdot \mathrm{d}t = P_{\mathrm{th}}^0 \cdot T = Z \cdot \Delta E \,, \tag{1.15}$$

where T is the full-power operation time in one year, Z is the number of fissions in the reactor in one year, corresponding to T hours of full power. The energy per fission event is

$$\Delta E = \Delta m \cdot c^2 \,, \tag{1.16}$$

where $\Delta m \cdot Z$ is the amount of uranium (and plutonium) fissioned in the reactor. The thermal power of the reaction is described by

$$P_{\mathrm{th}} = \frac{Z \cdot E_{\mathrm{f}}}{T} = \dot{m}_{\mathrm{c}} \cdot c_{\mathrm{c}} \cdot (T_{\mathrm{o}} - T_{\mathrm{i}}) \,, \tag{1.17}$$

with $E_{\mathrm{f}} = 200$ MeV/fission and $\dot{m}_{\mathrm{c}}$ as the coolant mass flow in the core, c_{c} as specific heat of the coolant, and T_{o}, T_{i} as outlet and inlet temperature, respectively, of coolant to the core. On the other hand, the relation

$$P_{\mathrm{th}} = \dot{m}_{\mathrm{st}} \cdot (h_{\mathrm{st}} - h_{\mathrm{feedw.}}) = P_{\mathrm{el}} / \eta_{\mathrm{tot}} \tag{1.18}$$

is valid with $\dot{m}_{\mathrm{st}}$ as flow of steam on the secondary side, and h_{st}, $h_{\mathrm{feedw.}}$ as the enthalpies of steam and feedwater, respectively. The total efficiency of electricity production is denoted by η_{tot}, which includes some efficiencies of different steps of power conversion:

$$\eta_{\mathrm{tot}} = \eta_{\mathrm{SG}} \cdot \eta_{\mathrm{th}} \cdot \eta_{\mathrm{turb}} \cdot \eta_{\mathrm{gen}} \cdot \eta_{\mathrm{Plant}} = \prod_i \eta_i \,, \tag{1.19}$$

with the efficiencies of steam generator, η_{SG}, of steam cycle, η_{th}, of turbine, η_{turb},, of generator, η_{gen}, and of the plant including the house load, η_{Plant}. Characteristic values for a large modern PWR are: $\eta_{\mathrm{th}} = 0.4$, $\eta_{\mathrm{turb}} = 0.98$, $\eta_{\mathrm{gen}} = 0.99$, $\eta_{\mathrm{Plant}} = 0.95$. Therefore a value of around 0.33 results for the total net efficiency of these types of plants today.

In comparison to conventional, fossil-fueled power plants, there result special aspects in the case of nuclear power plants, which make it necessary to discuss some physical questions as well as some neutron-physical aspects of the core design more closely. Safety questions are of special importance because of the high activity inventory of reactors as well as the serious radiological consequences in case of releases of radioactivity into the environment.

1.1.4 Some necessary fundamentals

1.1.4.1 Cross sections

The neutrons inside the core interact with the isotopes, and, depending on the type of isotope, reactions can be nuclear fission, elastic scattering, inelastic scattering, neutron capture (with successive γ-emissions or p- and α-emission, respectively) or neutron-reproducing reactions.

Table 1.5 shows the characteristics of the different reactions as well as some important examples. The desired main reaction inside the core of a nuclear reactor, nuclear fission, takes place with U-235, U-233, Pu-239 and Pu-241 isotopes. U-235 is the only isotope to be found in the natural environment, the other types of nuclides are produced artificially.

With neutron capture reactions, one neutron is absorbed by the nucleus. The new isotope produced by this is in an excited state. Some reactions of this kind proceed via γ-emissions to the ground state ((n,γ)-reaction), while in other reaction processes a proton ((n,p)-reaction) or a helium nucleus ((n,α)-reaction) is emitted. For thermal reactors, scattering processes that are responsible for slowing down the fast fission neutrons to the thermal energy area, are of special importance.

Table 1.5. Neutron reactions and important examples.

Reaction	Characterization	Example
nuclear fission	(n,f)	$^{235}U + n \rightarrow {}^{137}Ba + {}^{97}Kr + 2n$
neutron capture	(n,γ), (n,p), (n,α)	$^{235}U + n \rightarrow {}^{236}U + \gamma$
		$^{10}B + n \rightarrow {}^{7}Li + {}^{4}He$
elastic scattering	(n,n)	$^{12}C + n \rightarrow {}^{12}C + n,\ \Delta E = 0$
inelastic scattering	(n,n')	$^{238}U + n \rightarrow {}^{238}U + n',\ \Delta E > 0$
neutron producing reactions	(n.2n), (n.3n)	$^{238}U + n \rightarrow {}^{237}U + 2n$

For elastic scattering the conservation equations of energy and momentum in terms of classical mechanics are valid without any emission of radiation. In the case of inelastic scattering, a fraction of the kinetic energy involved in the scattering process is converted into γ-radiation energy. In general, the inelastic scattering is a threshold reaction, i.e. the kinetic energy of the neutron captured has to be above a certain level of energy (mostly above 100 keV) to enable such a process to take place. Above a threshold energy of around 1 MeV, some isotopes (e.g. U-238) can absorb one neutron and release two or more neutrons ((n.2n)-reaction, (n.3n)-reaction).

To attain a quantitative description of the reaction process, the terms cross section, neutron flux and reaction rate have been introduced to reactor technology. In particular, the term cross section is suitable to quantitatively record reaction rates of neutrons with nuclei.

From a simple consideration of the transmission of a neutron flow through a target with the thickness d and the number of nuclei per volume unit, N, the following exponential law can be deduced for the decrease of intensity I_0 to I:

$$I = I_0 \cdot e^{-\sigma N d} . \tag{1.20}$$

The quantity σ, defined by this simplified derivation,

$$\sigma = \frac{\ln(I_0/I)}{N \cdot d}, \tag{1.21}$$

is called the microscopic cross section for a neutron reaction. The dimension is 1 barn = 10^{-24} cm^2. Additionally, macroscopic cross sections with the dimension cm^{-1} can be used applying the equation

$$\Sigma = N \cdot \sigma, \tag{1.22}$$

where N is the number of nuclei per volume. Cross sections depend on the neutron energy and partly on the scattering angle. Figure 1.8 represents the energy dependence of different cross sections for U-235.

(n,γ)-reactions take place for instance in U-238 and Th-232. For U-238, the corresponding cross section versus the neutron energy is shown in Fig. 1.9a. For these isotopes, resonance absorptions within the epithermal area are particularly remarkable, as they play an important role for balancing the neutron action at rising power and temperature and hence are of essential importance for the reactor's safety behavior (see Sect. 1.1.7.4).

The resonance absorption rises in case of rising fuel temperatures, causing a negative feedback on the criticality of reactors. Finally some distinct characteristics of moderator materials should be mentioned. For the necessary slowing-down of fast fission neutrons to thermal energy in a thermal reactor, moderators are needed. Common materials today are H_2O, D_2O, and carbon. Figure 1.9c shows the total cross sections for light hydrogen and carbon, while Fig. 1.9d represents the cross sections of B-10, Cd, and In, which are used as absorber materials in control elements. Pu-239 is formed during reactor operation from U-238, with the fission cross section showing a characteristic resonance behavior in the thermal energy region (Fig. 1.9b). All cross sections of important materials in reactor technology have been measured today dependent on energy and, in the high-energy region, partly dependent on scattering angles. They are included in specific libraries for cross sections to be used in large computer programs for reactor design.

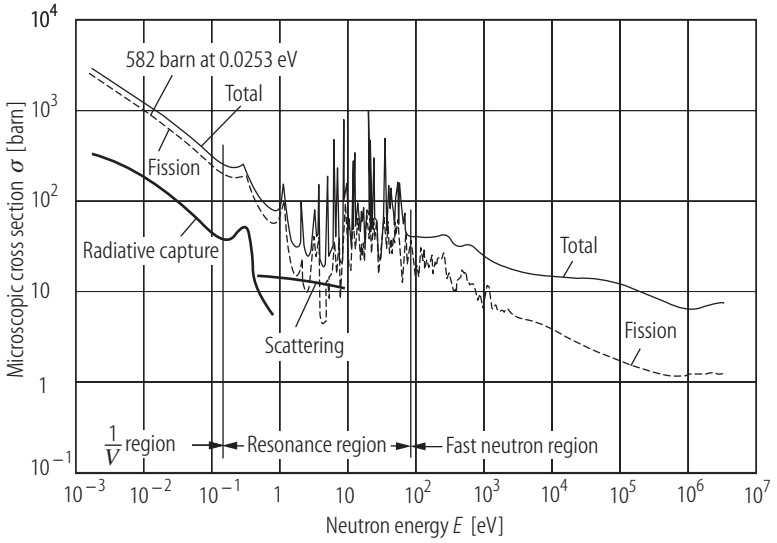


Fig. 1.8. Some cross sections of U-235 versus neutron energy, *E*.

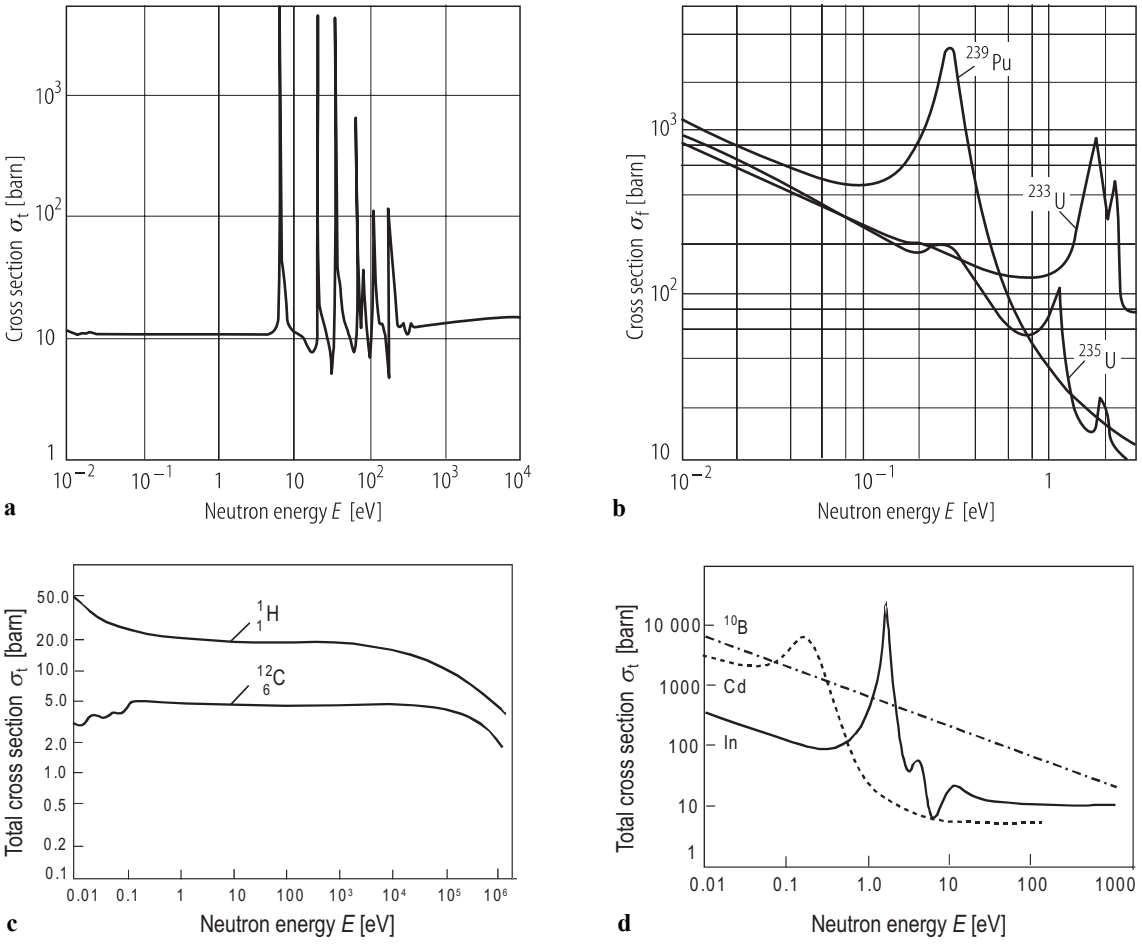


Fig. 1.9. Cross sections of some important reactor materials versus neutron energy, *E*: **(a)** resonance absorption cross section of U-238; **(b)** fission cross sections of different fuels (U-235, U-233, Pu-239); **(c)** cross sections of moderators (H-1, C-12); and **(d)** cross sections of absorbers (B-10, Cd, In).

1.1.4.2 Neutron flux and reaction rates

In order to set up neutron balances and to be able to quantitatively describe a reactor core in any operation or failure situation, the terms neutron flux and reaction rate are essential. If all neutrons have the same velocity υ, and n stands for the number of neutrons per volume unit, the neutron flux, ϕ, can be defined as

$$\phi = n \cdot \upsilon \,, \tag{1.23}$$

with the dimension $cm^{-2}s^{-1}$. If the neutrons show a spatial and energy distribution $n(\vec{r}, E)$, the flux per energy unit follows the relation

$$\phi(\vec{r}, E) = n(\vec{r}, E) \cdot \upsilon(E) \,, \tag{1.24}$$

with the dimension $cm^{-2}s^{-1}eV^{-1}$ and $\upsilon = (2\,E/m)^{1/2}$. By integration via a neutron spectrum one attains the following equation for the neutron flux, which is then only depending on spatial coordinates:

$$\phi(\vec{r}) = \int_{(E)} n(\vec{r}, E) \cdot \upsilon(E) \cdot \mathrm{d}E \,. \tag{1.25}$$

The probability of a certain reaction taking place is determined by the product of the neutron flux, $\phi\,[cm^{-2}s^{-1}]$, the number of nuclei per volume, $N\,[cm^{-3}]$, as well as the proportionality constant, i.e. the cross section, $\sigma\,[cm^2]$. The reaction rate, R $[cm^{-3}\ s^{-1}]$, is then defined by the expression

$$R = \phi \cdot \sigma \cdot N \,. \tag{1.26}$$

As cross sections are mostly energy-dependent, the reaction rate can be calculated by integration:

$$R = \int_{(E)} \phi(E) \cdot \sigma(E) \cdot N \cdot \mathrm{d}E \,, \tag{1.27}$$

with $\phi(E) \cdot \mathrm{d}E$ as neutron flux distribution. The reaction rate will also, sometimes strongly, depend on the location, according to the spatial distribution of the neutron flux and the concentrations N. Thus, for the rate, R $[cm^{-3}s^{-1}]$, one has:

$$R(\vec{r}) = \int_{(E)} \phi(\vec{r}, E) \cdot \sigma(E) \cdot N(\vec{r}) \cdot \mathrm{d}E \,. \tag{1.28}$$

The sufficiently exact determination of neutron fluxes, $\phi(E, \vec{r})$, and reaction rates, $R(\vec{r})$, is the task of diffusion and transport calculations. The neutron flux distribution, $\phi(E, \vec{r})$, is strongly energy-dependent, as shown in Fig. 1.10. In the spectrum, one distinguishes between thermal, epithermal and fast regions.

1.1.4.3 Neutron spectrum

The energy spectrum of neutrons in the reactor is the basis to calculate weighted cross sections, which are necessary to find important reactor parameters like criticality constant, reactivity coefficients, efficiency of control rods, power density, and neutron-induced loads on materials. The spectrum contains neutrons with energies between 10 MeV (fission neutrons) and around 0.025 eV (thermal neutrons). Often one talks about the fast, epithermal and thermal part of the neutron spectrum. Approximately in these regions simple equations can be used derived from slowing-down theory and from the theory of thermalization. Figure 1.10 shows the total spectrum qualitatively.

The fast spectrum ($10^5 < E < 10^7$ eV) can be described by the equation

$$\phi(E) \cdot \mathrm{d}E \approx C_1 \cdot \sqrt{E} \cdot \mathrm{e}^{-C_2 E} \cdot \mathrm{d}E \,. \tag{1.29}$$

This relation corresponds to the energy distribution of the fission neutrons. During the slowing-down process in the epithermal part of the spectrum ($1 < E < 10^5$ eV), an energy dependence of the flux like

$$\phi(E)\cdot \mathrm{d}E \approx \frac{C_3}{E}\cdot \mathrm{d}E \tag{1.30}$$

can be used to describe the conditions of weakly absorbing media, whereas in the thermal part of the spectrum ($10^{-3} < E < 1$ eV) the expression

$$\phi(E)\cdot \mathrm{d}E \approx C_4 \cdot E \cdot \mathrm{e}^{-E/kT}\cdot \mathrm{d}E \tag{1.31}$$

is a good approximation describing the thermal equilibrium. This is nearly a Maxwellian distribution of energies and velocities in a thermal reactor system. In case of fast reactors the thermal part of the spectrum does not exist.

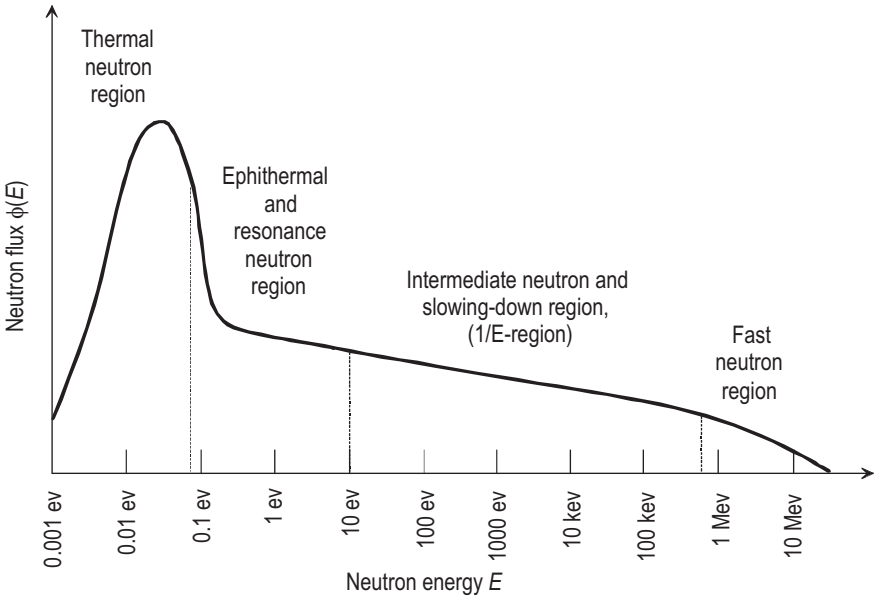


Fig. 1.10. Qualitative relation between neutron flux and neutron energy in a thermal reactor.

1.1.4.4 Diffusion of neutrons

The transport of neutrons in the reactor is described by the integrodifferential equation of transport, the Boltzmann equation. Approximately the diffusion equation can be applied, if groups of neutrons with different energies in the reactor are considered. The thermal neutrons as a one-group representation in a well-moderated thermal reactor can serve as an example. The derivation of the diffusion equation can be explained with regard to Fig. 1.11.

The number of neutrons inside the volume element ΔV is changed by three effects: there can be a source of neutrons ($Q(t)\cdot \mathrm{d}V$), absorption of neutrons ($\Sigma_\mathrm{a}\cdot \Phi \cdot \mathrm{d}V$) and a leakage of neutrons through the surface ΔF of the volume ΔV ($\int \vec{j}\cdot \mathrm{d}\vec{f}$).

Altogether the neutron balance for the volume ΔV delivers the equation

$$\frac{\partial}{\partial t}\int\limits_{(\Delta V)} \left(n(t)\cdot \mathrm{d}V\right) = \int\limits_{(\Delta V)} Q\cdot \mathrm{d}V - \int\limits_{(\Delta V)} \Sigma_\mathrm{a}\cdot \phi \cdot \mathrm{d}V - \int\limits_{(\Delta F)} \vec{j}\cdot \mathrm{d}\vec{f}\,. \tag{1.32}$$

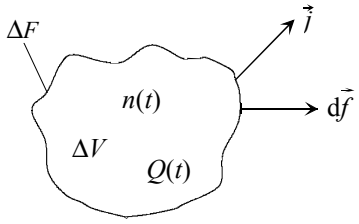


Fig. 1.11. Volume element in the reactor core for the derivation of the diffusion equation.

In case of monoenergetic neutrons, $\phi = n \cdot v$ is valid. According to the law of Gauss, the surface integral can be transformed into a volume integral:

$$\int_{(\Delta F)} \vec{j} \cdot \mathrm{d}\vec{f} = \int_{(\Delta V)} \mathrm{div}\, \vec{j} \cdot \mathrm{d}V \,. \tag{1.33}$$

The connection between the current density of neutrons, $\vec{j}$, and the flux, ϕ, is given by the first law of Fick:

$$\vec{j} = -D \cdot \mathrm{grad}\, \phi \,, \tag{1.34}$$

where D is the diffusion coefficient in the medium considered. Detailed theoretical considerations show that the diffusion coefficient is connected to the cross section of neutron transport, Σ_{tr}, and the mass number, A, of the isotopes in the diffusion medium by

$$D = \frac{1}{3 \cdot \Sigma_{\mathrm{tr}}} \cdot \frac{1}{1 - 2/3A} \,. \tag{1.35}$$

Integration of the balance equation, taking into account Gauss' and Fick's laws and considering very small values of ΔV, results in the time-dependent diffusion equation,

$$\frac{1}{v} \frac{\partial \phi}{\partial t} = Q - \Sigma_{\mathrm{a}} \cdot \phi + D \cdot \Delta \phi \,, \tag{1.36}$$

with $\mathrm{div}(\mathrm{grad}\phi) = \Delta\phi$ as the Laplacian operator. In many applications of reactor statics the time-independent diffusion equation is used in the form:

$$D \cdot \Delta\phi - \Sigma_{\mathrm{a}} \cdot \phi + Q = 0 \,, \tag{1.37}$$

with the source strength of neutrons,

$$Q = \nu \cdot \Sigma_{\mathrm{f}} \cdot \phi \,, \tag{1.38}$$

in a thermal reactor as an example. Some characteristics of neutron diffusion are easily explained considering a point source of neutrons.

Applying the Laplace operator for spherically symmetric conditions,

$$\Delta\phi = \frac{\partial^2 \phi}{\partial r^2} + \frac{2}{r} \frac{\partial \phi}{\partial r} \,, \tag{1.39}$$

for a point neutron source the following flux distribution is derived:

$$\phi(r) = \frac{c}{r} \cdot \mathrm{e}^{-r/L} \,, \tag{1.40}$$

where $L^2 = D/\Sigma_{\mathrm{a}}$ is called the square of the diffusion length, L, which is connected with the migration length of neutrons in a diffusion medium by the expression

$$L^2 = \frac{1}{6}\,\overline{r^2}\,. \tag{1.41}$$

Some characteristic diffusion parameters for different media which are important in reactor technology are given in Table 1.6. These data are valid for neutrons with thermal energies.

The shortest diffusion passes are therefore found in H_2O-moderated systems, the passes in C-moderated reactors are much longer, the longest are found in D_2O systems. The diffusion lengths of thermal neutrons in fuel materials are in the order of 1 cm and smaller.

The average diffusion time or lifetime, $\overline{\tau}$, of a thermal neutron can be defined as the time, during which the neutron with the average thermal velocity v_{th} diffuses before it is absorbed inside the reactor.

$$\overline{\tau} = \frac{\lambda_a}{v_{th}} = \frac{1}{\Sigma_a \cdot v_{th}}\,. \tag{1.42}$$

The variable λ_a is the average free path for absorption. For the important moderators the relevant values follow from Table 1.7.

Table 1.6. Characteristic diffusion parameters of some important materials in reactor technology.

Material	Σ_a [cm^{-1}]	D [cm]	L [cm]	$\sqrt{\overline{r^2}}$ [cm]
H_2O	2.21×10^{-2}	0.146	2.570	6.295
D_2O (0.16 % H_2O)	3.75×10^{-5}	0.883	153.4	375.7
C	2.77×10^{-4}	0.795	53.57	131.2
U-238	0.62	0.852	1.172	2.871
UO_2 (3 % enrich.)	0.554	0.352	0.797	1.952

Table 1.7. Thermal diffusion times of important moderators (T_h = 0.025 eV, v_{th} = 2200 m/s).

Material	Σ_a [cm^{-1}]	$\overline{\tau}$ [s]
H_2O	2.21×10^{-2}	2.06×10^{-4}
D_2O	3.75×10^{-5}	1.21×10^{-1}
C	2.77×10^{-4}	1.64×10^{-2}

The characteristic diffusion times always have to be regarded in connection with the times to slow down the neutrons from fission energies to thermal energies. In well-moderated systems normally the relation τ_S (slowing down) << τ_D (diffusion) is valid. The half-life of the neutrons themselves ($\tau_{1/2}$ = 11.7 min) does not play a role in reactor operation because of the large value compared to τ_S and τ_D.

Solutions of the diffusion equation for specific geometries are presented in Sect. 1.1.5 in connection with critical reactor systems. When applying the diffusion equation, boundary conditions have to be fulfilled: as an example, neutron flux and neutron stream have to be continuous if there is transport between different media; the flux distribution obeys symmetric conditions and the flux always has to be positive. Furthermore the flux vanishes outside the boundary if there is a vacuum outside the diffusion medium (see Fig. 1.12).

Transport theory delivers the result that the flux becomes zero at the extrapolated boundary of the reactor. For this distance, d, the relation

$$d = 0.71 \cdot \lambda_{tr} = 0.71 / \Sigma_{tr} \tag{1.43}$$

can be applied, with Σ_{tr} as the macroscopic cross section for transport in the diffusing medium. In order to calculate the spectrum of neutrons in a reactor the diffusion equation is applied for many groups of neutrons and terms to describe the transport of neutrons between different groups (Fig. 1.13).

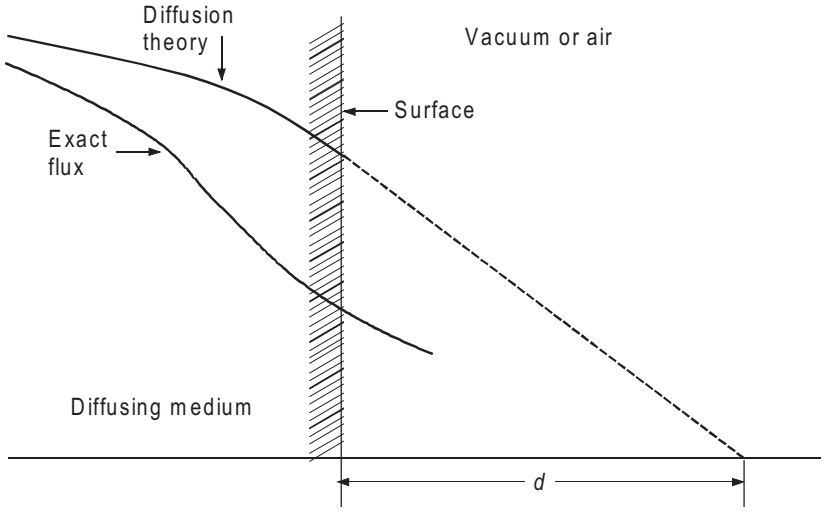


Fig. 1.12. Behavior of the neutron flux on a free (vacuum) surface: extrapolation distance, *d*.

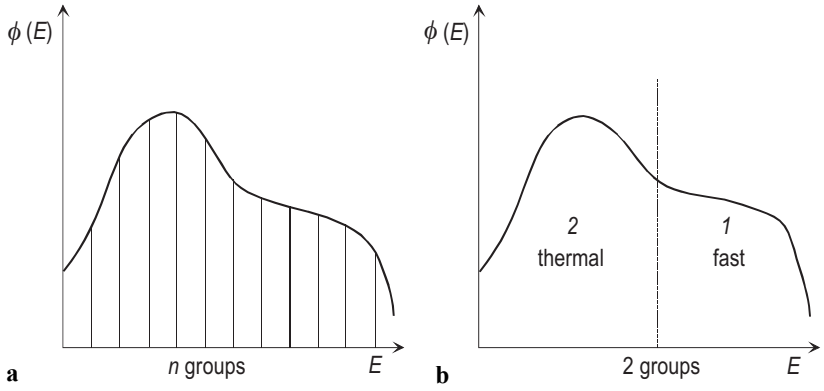


Fig. 1.13. Neutron energy spectrum in different groups: **(a)** *n* groups, **(b)** 2 groups.

For the case of a two-group calculation of the reactor spectrum the following system of coupled differential equations can be applied:

$$\begin{aligned} -D_1 \cdot \Delta\phi_1 + \Sigma_{a1} \cdot \phi_1 + \Sigma_{1\to 2} \cdot \phi_1 &= \nu \cdot \Sigma_{f1} \cdot \phi_1 + \nu \cdot \Sigma_{f2} \cdot \phi_2, \\ -D_2 \cdot \Delta\phi_2 + \Sigma_{a1} \cdot \phi_2 &= \Sigma_{1\to 2} \cdot \phi_1. \end{aligned} \tag{1.44}$$

Here D_1, Σ_{a1}, ν, Σ_f are averaged about the fast spectrum, correspondingly D_2, Σ_{a2} are averaged values for the thermal part; $\Sigma_{1\to 2}$ represents a transfer cross section for the neutron transport from the fast to the thermal group in this simple model. This parameter can be calculated using slowing-down theory (see Sect. 1.1.4.5). Today computer programs are used to calculate the neutron spectrum very precisely in more than 100 groups.

1.1.4.5 Slowing down of neutrons

In thermal reactors the fast fission neutrons slow down by collisions with light nuclei, until thermal equilibrium is realized. Usually an energy limit of around 1 eV is used as a boundary between thermal and epithermal spectrum. For neutron energies above this value the moderator nuclei are assumed to be free and without binding effects. In this energy region elastic collisions can be assumed.

Below these limits complex chemical binding effects and crystallic structures of moderators must be taken into account. The slowing-down process of the neutrons between around 2 MeV and 1 eV is named moderation and is done by moderators like H_2O, D_2O or carbon. In all thermal reactors in the energy region mentioned above there are strong resonance absorbers like U-238 and Th-232. The influence of these absorbers on the neutron economy and on the reactivity feedback effects in case of temperature changes is very important; the factor p in the four-factor formula, the resonance escape probability, describes these effects.

The moderator enables the neutrons to pass the resonance area without too high parasitic absorption. For the collision process of neutrons with light nuclei, energy and momentum conservation are valid, the laws of elastic collisions can be applied. Furthermore in most practical cases the elastic scattering of neutrons can be assumed spherically symmetric in the center-of-mass system. Applying the conservation laws in the laboratory system one obtains the relation:

$$E = E_0 \cdot \left(\frac{A^2 + 2A \cdot \cos\vartheta + 1}{(A+1)^2} \right), \tag{1.45}$$

for the kinetic energy of the neutron before collision (E_0) and after collision (E). The angle between the direction of the incident and the scattered neutron is denoted by ϑ. With the abbreviation

$$\alpha = \left(\frac{A-1}{A+1} \right)^2, \tag{1.46}$$

one obtains:

$$E = E_0 \cdot \frac{1}{2}\left((1+\alpha) + (1-\alpha)\cdot\cos\vartheta\right). \tag{1.47}$$

For $\vartheta = 0$ the energy of the scattered neutron attains a maximum ($E_{max} = E_0$), for $\vartheta = \pi$ the minimum energy is given by $E_{min} = \alpha \cdot E_0$. The energy of the elastically scattered neutron lies between E_0 and $\alpha \cdot E_0$ corresponding to the angle ϑ. Isotropic scattering in the center-of-mass system includes that the probability of all scattering angles is the same:

$$w(E,\vartheta)\cdot \mathrm{d}\vartheta = \frac{\mathrm{d}\Omega}{4\pi} = \frac{2\pi\cdot\sin\vartheta\cdot\mathrm{d}\vartheta}{4\pi} = \frac{1}{2}\sin\vartheta\cdot\mathrm{d}\vartheta. \tag{1.48}$$

Starting from this probability distribution, the probability of scattering from energy E_0 to E is given by

$$w(E_0 \to E)\cdot \mathrm{d}E = w(\vartheta)\cdot\mathrm{d}\vartheta = w(\vartheta)\cdot\frac{\mathrm{d}\vartheta}{\mathrm{d}E}\cdot\mathrm{d}E. \tag{1.49}$$

Applying the relation $E = E_0 \cdot f(\alpha, \vartheta)$ leads to:

$$w(E_0 \to E) = \frac{1}{(1-\alpha)\cdot E} \tag{1.50}$$

for $\alpha \cdot E_0 < E < E_0$. In the other regions of the energy scale, $w = 0$ is valid. Figure 1.14 shows this distribution. The distribution is normalized to 1,

$$\int_{\alpha E_0}^{E_0} w(E_0 \to E)\cdot\mathrm{d}E = \frac{1}{E_0(1-\alpha)}\cdot\int_{\alpha E_0}^{E_0}\mathrm{d}E' = 1. \tag{1.51}$$

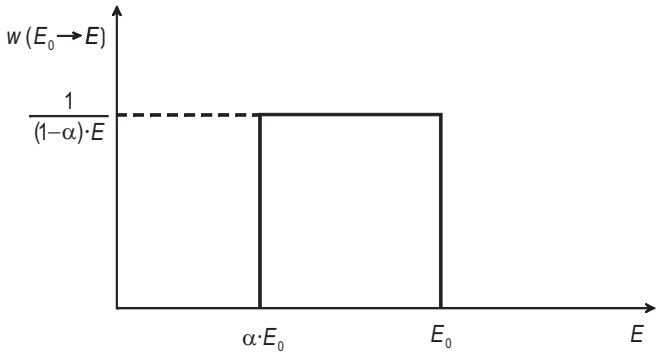


Fig. 1.14. Probability distribution $w(E_0 \to E)$.

The average energy loss of the neutron during a collision is given by

$$\Delta \bar{E} = \int_{\alpha E_0}^{E_0} (E_0 - E) \cdot w(E_0 \to E) \cdot \mathrm{d}E = \frac{1-\alpha}{2} \cdot E_0 , \tag{1.52}$$

whereas the average relative loss of energy results to be:

$$\Delta \bar{E} / E_0 = (1-\alpha)/2 . \tag{1.53}$$

This value is large for light nuclei and very small for heavy nuclei. The average energy of the elastically scattered neutron is calculated from

$$\bar{E} = \int_{\alpha E}^{E_0} E \cdot w(E_0 \to E) \cdot \mathrm{d}E = E_0 \cdot \frac{1+\alpha}{2} . \tag{1.54}$$

As the average relative loss of energy during collision is constant, a new variable instead of energy can be introduced. This is the lethargy, u, defined by

$$u = \ln(E_0 / E) . \tag{1.55}$$

With decreasing energy the lethargy is rising up with constant steps between each collision. The average gain of lethargy during a collision follows from

$$\overline{\Delta u} = \int_{\alpha E_0}^{E_0} \ln\left(\frac{E_0}{E}\right) \cdot w(E_0 \to E) \cdot \mathrm{d}E \tag{1.56}$$

and delivers the expression

$$\overline{\Delta u} = 1 + \frac{\alpha}{1-\alpha} \cdot \ln \alpha = \xi . \tag{1.57}$$

Using the definition of α one obtains

$$\overline{\Delta u} = \xi = 1 - \frac{(A-1)^2}{2A} \cdot \ln\left(\frac{A+1}{A-1}\right). \tag{1.58}$$

For small values of A the logarithmic energy decrement, ξ, can be approximated by

$$\xi = \frac{2}{A + 2/3} . \tag{1.59}$$

On average, the lethargy is increased by the value ξ by each collision. If there are n collisions, then $E_0, E_1, E_2, \ldots E_n$ represent the average energies after each collision. One obtains

$$\ln\left(\frac{E_0}{E_n}\right) = \ln\left(\frac{E_0}{E_1}\cdot\frac{E_1}{E_2}\cdot\frac{E_2}{E_3}\cdot\ldots\cdot\frac{E_{n-1}}{E_n}\right) = \ln\left(\frac{E_0}{E_1}\right)^n = n\cdot\ln\left(\frac{E_0}{E_1}\right). \tag{1.60}$$

The number of collisions which is necessary to slow down a neutron from fast energy, E_0, to thermal energy, E_{th}, is then calculated by

$$n = \frac{1}{\xi}\cdot\ln\left(\frac{E_0}{E_{th}}\right) \approx \frac{A+2/3}{2}\cdot\ln\left(\frac{E_0}{E_{th}}\right). \tag{1.61}$$

In Table 1.8, some characteristic values of the quantities discussed above are given for several important reactor materials.

Table 1.8. Characteristic values in slowing-down processes, given for different reactor materials.

Material	A	α	ξ	n
H_2	1	0	1	18
D_2	2	0.111	0.726	25
C	12	0.716	0.158	115
O_2	32	0.779	0.120	152
U	238	0.983	0.0084	2167

A good moderator is characterized by large values of ξ and Σ_s. The product, $\xi\cdot\Sigma_s$, is the slowing-down power. In a system with hydrogen as moderator and weak absorption the interaction between neutrons and moderator nuclei is described by the collision density, $F(E)$. Here,

$$F(E)\cdot dE = \Sigma_t(E)\cdot\phi(E)\cdot dE \tag{1.62}$$

is the number of interactions in the energy interval $E \ldots E + dE$. The slowing down in a hydrogen-containing system is thus characterized by the equation

$$F(E)\cdot dE = \frac{S_0}{E_0}\cdot dE + dE\cdot\int_E^{E_0}\frac{F(E')}{E'}\cdot dE'. \tag{1.63}$$

Here, S_0 is the source strength for emitting neutrons with energy E_0, the function $1/E'$ inside the integral includes the collision probability function for the moderator hydrogen.

The integral equation

$$F(E) = S_0/E_0 + \int_E^{E_0}\frac{F(E')}{E'}\cdot dE' \tag{1.64}$$

is solved by the function

$$F(E) = C/E. \tag{1.65}$$

With the starting condition $F(E_0) = S_0/E_0$, the result is

$$F(E) = S_0/E. \tag{1.66}$$

Thus, for the neutron flux in the slowing-down region one obtains the result

$$\phi(E) = \frac{S_0}{\Sigma_s(E)\cdot E}. \tag{1.67}$$

This 1/E-dependence of the flux in the epithermal part of the energy spectrum is just valid for an infinite, very weakly absorbing medium. In a mixture of moderators (for instance H_2O in light water reactors) the flux behavior corresponds to

$$\phi(E) \approx \frac{S}{\xi \cdot \Sigma_s(E)} \cdot \frac{1}{E}. \tag{1.68}$$

In all reactors, resonance absorbers (U-238, Th-232) are included in the fuel. In this case a much more complex integral-equation has to be solved to calculate the spectrum. In a very simplified theory a slowing-down density, $q(E)$, can be defined by the expression

$$q(E) = \phi(E) \cdot \xi \cdot \Sigma_s \cdot E. \tag{1.69}$$

If neutrons are absorbed, for instance in resonance absorbers, the slowing-down density can be balanced as follows:

$$\begin{aligned} &q(E - \mathrm{d}E) = q(E) + \Sigma_a(E) \cdot \phi(E) \cdot \mathrm{d}E, \\ &q(E) - \frac{\mathrm{d}q}{\mathrm{d}E} \cdot \mathrm{d}E \approx q(E) + \Sigma_a(E) \cdot \phi(E) \cdot \mathrm{d}E, \\ &-\mathrm{d}q = \Sigma_a(E) \cdot \phi(E) \cdot \mathrm{d}E. \end{aligned} \tag{1.70}$$

This approximation leads to a differential equation for the slowing-down power:

$$-\mathrm{d}q = \Sigma_a(E) \cdot \frac{q(E)}{\xi \cdot \Sigma_s \cdot E} \cdot \mathrm{d}E, \tag{1.71}$$

with the solution

$$q(E) = q_0 \cdot \exp\left\{ -\int_E^{E_0} \frac{\Sigma_a(E')}{\xi \cdot \Sigma_s \cdot E'} \mathrm{d}E' \right\}, \tag{1.72}$$

where q_0 corresponds to the number of fast fission neutrons at $E_0 \approx 2$ MeV. For thermal neutrons, $q(E) = q(E_{th})$ is applied, and the ratio,

$$p = \frac{q(E_{th})}{q_0} = \exp\left\{ -\int_{E_{th}}^{E_0} \frac{\Sigma_a(E')}{\xi \cdot \Sigma_s \cdot E'} \mathrm{d}E' \right\}, \tag{1.73}$$

is calculated as the above-mentioned resonance escape probability which is an important factor in the four-factor formula describing the criticality of a reactor. The prompt fission neutrons are released around 10^{-14} s after fission. The slowing down of the fast fission neutrons by collisions occurs during a characteristic time τ_s. The energy loss in time can be calculated using a continuous slowing-down model:

$$-\frac{\mathrm{d}E}{\mathrm{d}t} = \upsilon \cdot \xi \cdot \Sigma_s \cdot E. \tag{1.74}$$

The velocity of the neutrons is given by

$$\upsilon = (2E/m)^{1/2}. \tag{1.75}$$

Integration of the differential equation delivers the result:

$$t_s = -\int_{E_0}^{E_{th}} \frac{\mathrm{d}E}{\sqrt{\frac{2E}{m}} \cdot \xi \cdot \Sigma_s \cdot E}, \tag{1.76}$$

with the limits $E_0 = 2$ MeV and $E_{th} = 0.025$ eV. The solution is

$$t_s = \frac{\sqrt{2m}}{\xi \cdot \Sigma_s} \cdot \left(\frac{1}{\sqrt{E_{th}}} - \frac{1}{\sqrt{E_0}} \right) \approx \frac{2}{v_{th} \cdot \xi \cdot \Sigma_s}, \tag{1.77}$$

because $E_0 >> E_{th}$. The slowing-down time, t_s, is mainly given by the slowing-down parameter, $\xi \cdot \Sigma_s$. Typical values for thermal systems are contained in Table 1.9.

Table 1.9. Slowing-down parameters of different moderators in thermal reactors.

Moderator	$\xi \cdot \Sigma_s$ [cm^{-1}]	t_s [s]
H_2O	1.36	6.68×10^{-6}
D_2O	0.18	5.05×10^{-5}
C	0.06	1.52×10^{-4}

1.1.4.6 Resonance escape probability

One very important factor in the value of the criticality constant, k, of a thermal reactor is the resonance escape probability, p. Resonance absorption is caused by the resonances of U-238 or Th-232 in the energy region between 5 and 1000 eV. Applying slowing-down theory, the variable p is calculated to be

$$p = \exp\left\{ - \int_{E_{th}}^{E_0} \frac{\Sigma_a(E)}{\bar{\xi} \cdot (\Sigma_s(E) + \Sigma_a(E))} \cdot \frac{dE}{E} \right\}, \tag{1.78}$$

for a homogeneous mixture of absorber and moderator. In the formula, E_{th} denotes the thermal energy, E_0 the energy of fast neutrons (≈ 2 MeV), and $\bar{\xi}$ the logarithmic energy decrement of the mixture of isotopes. The term $\Sigma_a(E)$ includes all absorptions, especially in the resonances. This expression for p is often used in the form

$$p = \exp\left\{ - \frac{1}{\bar{\xi}} \cdot \frac{N_a}{\Sigma_s} \cdot \int_{E_{th}}^{E_0} \sigma_{a,eff} \cdot \frac{dE}{E} \right\}, \tag{1.79}$$

with N_a denoting the number of absorbing nuclei, and Σ_s the total scattering cross section. The effective absorption cross section, $\sigma_{a,eff}$, is defined by the equation

$$I = \int_{E_{th}}^{E_0} \sigma_{a,eff} \cdot \frac{dE}{E} = \int_{E_{th}}^{E_0} \frac{1}{1 + N_a \cdot \sigma_a / N_s \cdot \sigma_s} \cdot \sigma_a(E) \cdot \frac{dE}{E}. \tag{1.80}$$

This integral is called the effective resonance integral and is mainly depending on the type of resonance absorber and on the ratio N_a/N_s. It can be calculated or measured. Figure 1.15 shows the values of this integral in case of U-238 and Th-232 for different ratios of Σ_s/N_a.

In the case of infinite dilution, $N_s >> N_a$, the definition reduces to

$$I_\infty = \int_{E_{th}}^{E_0} \sigma_a(E) \cdot \frac{dE}{E}. \tag{1.81}$$

The values measured for this specific case of a very weakly absorbing mixture are 220 barn for U-238 and 60 barn for Th-232.

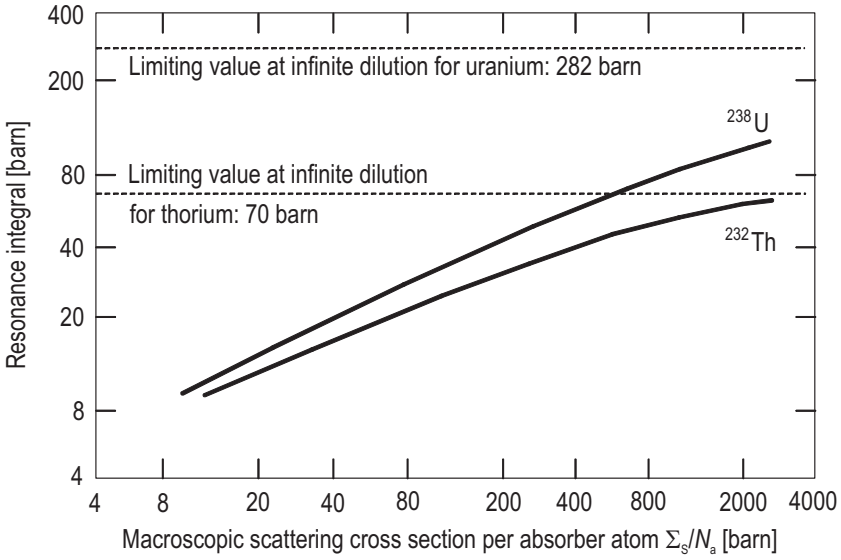


Fig. 1.15. Effective resonance integrals of U-238 and Th-232, respectively.

If the resonance absorber is arranged in a lump structure, as it is the case in all reactors developed today (fuel rods, plates, blocks, pebbles with coated particles), the resonance absorption inside the lump is much lower than at the surface of the lumped body. This effect is caused by the very strong absorption by the resonances. Therefore the overall effect of resonance absorption is smaller in a heterogeneous arrangement compared to a homogeneous mixture.

Theoretically analyzing the calculation of different neutron fluxes in the absorber and in the moderator, as well as taking into account geometrical considerations for the flight and absorption of neutrons in such a heterogeneous arrangement results in specific expressions for the resonance integral. Typical results are:

$$I = \int_{E_{\mathrm{th}}}^{E_0} \sigma_{\mathrm{a,eff}}(E) \cdot \frac{\mathrm{d}E}{E} = A + B \cdot \frac{S}{M}, \tag{1.82}$$

where A and B are constants for the absorbing material, and S/M is the ratio of surface to mass of fuel.

For UO_2, as an example, the values are $A = 11.51$ barn, $B = 22.1$ barn g/cm^2. For a fuel rod used in LWRs, the ratio S/M means $2/(\rho \cdot R)$. Specifically for the PWR fuel, I would be around 20 barn for rods of 1 cm diameter.

To calculate p-factors for heterogeneous systems which are precise enough for practical estimations, extensive computer codes are used today.

With the index 1 relating to the fuel, and 2 to the moderator, a simple approximation to the p-factor is:

$$p = \exp\left\{ -\frac{N_1 \cdot V_1}{N_2 \cdot V_2} \cdot \frac{1}{\bar{\xi}\sigma_{s_2}} \cdot I \right\}. \tag{1.83}$$

Values of p between 0.8 and 0.9 can be realized by the right choice of moderator and fuel volumes (N_2/N_1) and by heterogeneous arrangement of fuel (and fertile material U-238 in fuel with low enrichment) and moderator inside the reactor core.

1.1.5 Reactor equations and critical reactors

1.1.5.1 Reactor equations

In the following, the condition for a reactor with finite dimensions being critical shall be discussed. The substantial composition and the inner arrangement of fissionable materials, breeding materials, structure materials and fission products shall be given, and hence, the quantity k_∞ is determined. In a finite reactor, neutrons flow out the system over the edges, so that a value $k_\infty > 1$ has to be chosen to realize $k = 1$ for the system with finite dimensions.

To simplify matters, a thermal reactor with a neutron group within the thermal energy scope shall be considered. For this aim, suitable average values for the material quantities have to be defined. Source neutrons that emerge from fission as fast neutrons are posed as source in an appropriate form.

On the basis of the diffusion equation,

$$\frac{\partial n}{\partial t} = D \cdot \Delta\phi - \Sigma_\mathrm{a} \cdot \phi + Q\,, \tag{1.84}$$

through integration via the thermal energy scope ($0 < E < E_\mathrm{th}$), one obtains:

$$\int_0^{E_\mathrm{th}} \frac{\partial n}{\partial t} \cdot \mathrm{d}E = \int_0^{E_\mathrm{th}} D_\mathrm{th} \cdot \Delta\phi \cdot \mathrm{d}E - \int_0^{E_\mathrm{th}} \Sigma_\mathrm{a,th} \cdot \phi \cdot \mathrm{d}E + \int_0^{E_\mathrm{th}} Q \cdot \mathrm{d}E\,, \tag{1.85}$$

where in the first place ϕ depends from E, $\vec{r}$ and t, i.e. $\phi(E,\vec{r},t)$ is valid, correspondingly $n(E,\vec{r},t)$ applies. Usually the variables can be summed up with a product approach for the flow,

$$\phi(E,\vec{r},t) = F(\vec{r}) \cdot G(E) \cdot T(t)\,. \tag{1.86}$$

A thermal neutron flow can be defined by:

$$\phi_\mathrm{th} = \int_0^{E_\mathrm{th}} \phi(E) \cdot \mathrm{d}E\,. \tag{1.87}$$

Accordingly, the following values within the thermal energy scope shall be determined:

$$\frac{\phi_\mathrm{th}}{\upsilon_\mathrm{th}} = \int_0^{E_\mathrm{th}} n(E) \cdot \mathrm{d}E\,, \tag{1.88}$$

$$\overline{\Sigma_\mathrm{a,th}} \cdot \phi_\mathrm{th} = \int_0^{E_\mathrm{th}} \Sigma_\mathrm{a,th}(E) \cdot \phi(E) \cdot \mathrm{d}E\,, \tag{1.89}$$

$$\overline{D_\mathrm{th}} = \frac{1}{\phi_\mathrm{th}} \cdot \int_0^{E_\mathrm{th}} D_\mathrm{th}(E) \cdot \phi(E) \cdot \mathrm{d}E\,. \tag{1.90}$$

Thus, the diffusion equation can be put into the form:

$$\frac{1}{\upsilon_\mathrm{th}} \cdot \frac{\partial \phi_\mathrm{th}}{\partial t} = \overline{D_\mathrm{th}} \cdot \Delta\phi - \overline{\Sigma_\mathrm{a,th}} \cdot \phi_\mathrm{th} + Q_\mathrm{th}\,. \tag{1.91}$$

Finally, for the thermal neutron source, the approach

$$Q_\mathrm{th} = k_\infty \cdot \overline{\Sigma_\mathrm{a,th}} \cdot \phi_\mathrm{th} = \varepsilon \cdot p \cdot \nu \cdot \overline{\Sigma_\mathrm{f}} \cdot \phi_\mathrm{th} \tag{1.92}$$

can be made. With the help of average value estimation in the thermal energy scope it has been achieved that the thermal flow, $\phi_\mathrm{th}(\vec{r},t)$, depends on position and time only. This leads to the diffusion equation:

$$\frac{1}{\overline{\overline{v_{\mathrm{th}}}}}\cdot\frac{\partial\phi_{\mathrm{th}}}{\partial t}=\overline{D_{\mathrm{th}}}\cdot\Delta\phi_{\mathrm{th}}-\overline{\Sigma_{\mathrm{a,th}}}\cdot\phi_{\mathrm{th}}+k_{\infty}\cdot\overline{\Sigma_{\mathrm{a,th}}}\cdot\phi_{\mathrm{th}}\,. \tag{1.93}$$

Using the thermal diffusion length,

$$L_{\mathrm{th}}^2=\frac{\overline{D_{\mathrm{th}}}}{\overline{\overline{\Sigma_{\mathrm{a,th}}}}}\,, \tag{1.94}$$

the differential equation of the thermal flux follows as:

$$\frac{1}{\overline{\overline{D_{\mathrm{th}}}}\cdot\overline{\overline{v_{\mathrm{th}}}}}\cdot\frac{\partial\phi_{\mathrm{th}}}{\partial t}=\Delta\phi_{\mathrm{th}}+\frac{k_{\infty}-1}{L_{\mathrm{th}}^2}\cdot\phi_{\mathrm{th}}\,. \tag{1.95}$$

Introducing the material buckling, B_{m}^2, the diffusion equation in its final form is obtained:

$$\frac{1}{\overline{\overline{D_{\mathrm{th}}}}\cdot\overline{\overline{v_{\mathrm{th}}}}}\cdot\frac{\partial\phi_{\mathrm{th}}}{\partial t}=\Delta\phi_{\mathrm{th}}+B_{\mathrm{m}}^2\cdot\phi_{\mathrm{th}}\,. \tag{1.96}$$

The thermal neutron flow, ϕ_{th}, can now be represented by a product approach:

$$\phi_{\mathrm{th}}(\vec{r},t)=F(\vec{r})\cdot T(t)\,. \tag{1.97}$$

Hence, from the diffusion equation one obtains:

$$\frac{1}{\overline{\overline{D_{\mathrm{th}}}}\cdot\overline{\overline{v_{\mathrm{th}}}}}\cdot\frac{1}{T}\frac{\mathrm{d}T}{\mathrm{d}t}=\frac{\Delta F}{F}+B_{\mathrm{m}}^2\,. \tag{1.98}$$

Since the material buckling, B_{m}^2, depends only on parameters that are initially not depending on position, the terms

$$\frac{1}{T}\cdot\frac{\mathrm{d}T}{\mathrm{d}t}\,,\quad\frac{\Delta F}{F} \tag{1.99}$$

are independently equal to a constant which here shall be B_{g}^2. For the dependence on position, the differential equation

$$\Delta F+B_{\mathrm{g}}^2\cdot F=0 \tag{1.100}$$

can be derived. This equation describes the spatial change of the flow in the reactor. In a critical reactor, i.e. a reactor in which $\partial\phi_{\mathrm{th}}/\partial t=0$ or $\mathrm{d}T/\mathrm{d}t=0$ applies,

$$B_{\mathrm{m}}^2=B_{\mathrm{g}}^2 \tag{1.101}$$

has to be valid. This can be pointed out directly by the comparison of the partial differential equation for the neutron flow, ϕ_{th}, and the equation for the position-dependent fraction F. In a non-critical reactor,

$$B_{\mathrm{m}}^2\neq B_{\mathrm{g}}^2 \tag{1.102}$$

applies. If the reactor is over-critical, then $\partial\phi/\partial t>0$ or $\mathrm{d}T/\mathrm{d}t>0$, and therefore

$$B_{\mathrm{m}}^2>B_{\mathrm{g}}^2\,. \tag{1.103}$$

If, however, the reactor is under-critical, then $\partial\phi/\partial t<0$ or $\mathrm{d}T/\mathrm{d}t<0$, and hence

$$B_{\mathrm{m}}^2<B_{\mathrm{g}}^2\,. \tag{1.104}$$

For a critical reactor, which is always considered in reactor statics, from the relations mentioned above follows:

$$B^2 \cdot L_{\mathrm{th}}^2 = k_\infty - 1, \tag{1.105}$$

with the corresponding differential equation,

$$\Delta\phi + B^2 \cdot \phi = 0, \tag{1.106}$$

where $\phi_{\mathrm{th}} = \phi$ and $B_{\mathrm{m}}{}^2 = B_{\mathrm{g}}{}^2 = B^2$ have been set here. Thus, the criticality condition can be written in the form

$$1 = k_\infty \cdot \frac{1}{1 + B^2 \cdot L_{\mathrm{th}}^2}, \quad 1 = k_\infty \cdot W_{\mathrm{th}} \tag{1.107}$$

can be chosen, with W_{th} as probability for the thermal neutrons to remain inside the reactor system. An extension, when also considering a group of fast neutrons, leads to:

$$1 = k_\infty \cdot W_{\mathrm{th}} \cdot W_{\mathrm{s}} \tag{1.108}$$

for the critical reactor. Here, W_{s} represents the probability for fast neutrons to remain in the system. By dealing closely with the slowing-down process of neutrons in the context of the Fermi-age theory, for W_{s} the following can be found: with τ as Fermi-age,

$$W_{\mathrm{s}} = \mathrm{e}^{-B^2 \cdot \tau}, \quad \tau = \int_{E_{\mathrm{th}}}^{E_0} \frac{D(E)}{\xi \cdot \Sigma_{\mathrm{s}}} \cdot \frac{\mathrm{d}E}{E}. \tag{1.109}$$

For a reactor system in which two groups of neutrons (thermal and fast) are included, the criticality condition can be derived:

$$1 = k_\infty \cdot W_{\mathrm{th}} \cdot W_{\mathrm{s}} = k_\infty \cdot \frac{1}{1 + B^2 \cdot L_{\mathrm{th}}^2} \cdot e^{-B^2 \cdot \tau}. \tag{1.110}$$

In order to be able to record still small deviations from the even critical state, with the help of the criticality quantity, k_{eff}, one can also write:

$$k_{\mathrm{eff}} = k_\infty \cdot W_{\mathrm{th}} \cdot W_{\mathrm{s}} = 1. \tag{1.111}$$

All in all, the physical characteristics of the multiplying medium are included in k_∞, the measurements of the system about the quantity B^2 in W_{th} and W_{s}. The quantity B^2 for an even critical reactor is determined by the solution of the reactor equation for a special reactor geometry. The difference between an infinitely expanded medium and a reactor with finite dimensions shall be repeated. For the infinite medium the multiplication factor is

$$k_\infty = \frac{\text{rate of neutron production}}{\text{rate of neutron absorption}} = \frac{R_{\mathrm{P}}}{R_{\mathrm{A}}}, \tag{1.112}$$

$$k_\infty = \frac{\int \varepsilon \cdot p \cdot \nu \cdot \overline{\Sigma_{\mathrm{f}}} \cdot \phi_{\mathrm{th}} \cdot \mathrm{d}V}{\int \overline{\Sigma_{\mathrm{a,th}}} \cdot \phi_{\mathrm{th}} \cdot \mathrm{d}V}. \tag{1.113}$$

Here, the leakage, R_{L}, equals zero. For a finite reactor,

$$k_{\mathrm{eff}} = \frac{R_{\mathrm{P}}}{R_{\mathrm{A}} + R_{\mathrm{L}}} = 1 \tag{1.114}$$

applies for stationary operation; R_L is larger than zero. The smaller the reactor, the larger the leakage of neutrons. Overall, the critical condition with k_∞ given prescribes a particular size of the reactor. In anticipation of results in succeeding sections, it shall be mentioned here that, for instance, for a ball-shaped critical reactor, $B \sim 1/R$ would apply.

By solving the reactor equations for different geometries – spherical, cubic and cylindrical system – the relations shown in Table 1.10 can be derived. Figure 1.16 shows the characteristic flux functions for the different geometries. The radial and axial dependences of these functions are quite similar to each other.

Table 1.10. Differential equations, solutions and bucklings for important reactor geometries.

Geometry	Differential equation	Solution for flux	Geometrical buckling
spherical	$\frac{\partial^2\phi}{\partial r^2}+\frac{2}{r}\frac{\partial\phi}{\partial r}+B^2\cdot\phi=0$	$\phi(r)=\frac{C}{r}\cdot\sin\left(\frac{\pi r}{R}\right)$	$B^2=\left(\frac{\pi}{R}\right)^2$
rectangular	$\frac{\mathrm{d}^2\phi}{\mathrm{d}x^2}+\frac{\mathrm{d}^2\phi}{\mathrm{d}y^2}+\frac{\mathrm{d}^2\phi}{\mathrm{d}z^2}+B^2\cdot\phi=0$	$\phi(x,y,z)=C\cdot\cos\left(\frac{\pi x}{a}\right)\cdot\cos\left(\frac{\pi y}{b}\right)\cdot\cos\left(\frac{\pi z}{c}\right)$	$B^2=\left(\frac{\pi}{a}\right)^2+\left(\frac{\pi}{b}\right)^2+\left(\frac{\pi}{c}\right)^2$
cylindrical	$\frac{\partial^2\phi}{\partial r^2}+\frac{1}{r}\frac{\partial\phi}{\partial r}+\frac{\partial^2\phi}{\partial z^2}+B^2\cdot\phi=0$	$\phi(r,z)=C\cdot\cos\left(\frac{\pi Z}{H}\right)\cdot J_0\left(\frac{2.405\cdot r}{R}\right)$	$B^2=\left(\frac{\pi}{H}\right)^2+\left(\frac{2.405}{R}\right)^2$

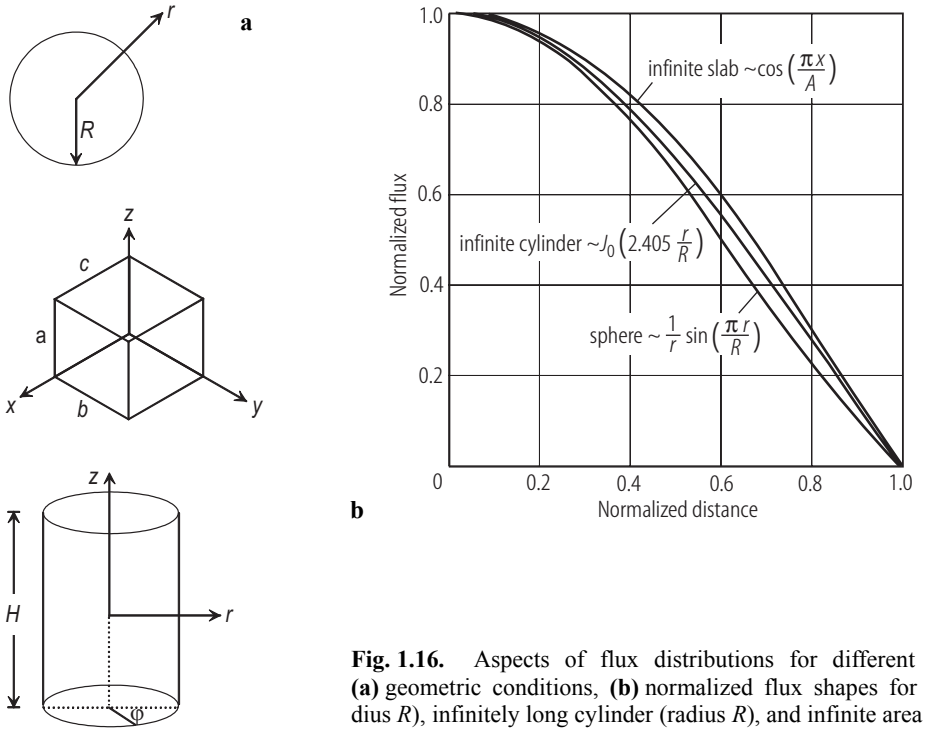


Fig. 1.16. Aspects of flux distributions for different reactor geometries: **(a)** geometric conditions, **(b)** normalized flux shapes for critical spheres (radius R), infinitely long cylinder (radius R), and infinite area slab (thickness A).

1.1.5.2 Aspects of criticality

A very important question in reactor physics is which ratio of the core dimensions must be chosen to get a minimum core volume and therefore, at given material composition, a minimum content of fissile material inside the core. For the cylindrical system the critical condition is

$$B^2 = \left(\frac{\pi}{H}\right)^2 + \left(\frac{2.405}{R}\right)^2 . \tag{1.115}$$

The volume, $V = \pi \cdot R^2 \cdot H$, should be minimized. From the relation

$$V(H, B^2) = \frac{\pi \cdot H \cdot (2.405)^2}{B^2 - (\pi / H)^2} \tag{1.116}$$

and from $\partial V / \partial H = 0$, one obtains:

$$H = \sqrt{3} \cdot \pi / B \quad \text{and} \quad R = \sqrt{3/2} \cdot 2.405 / B , \tag{1.117}$$

with $R/H = 2.405/\left(\sqrt{2} \cdot \pi\right) = 0.55$. The minimum volume of the reactor would be $V = 148/B^3$. Reactors with optimum neutron economy have to fulfill the requirement $H/D \approx 0.9$ (where D denotes the core diameter).

For new reactor concepts with self-acting decay heat removal (see Sect. 1.1.4), a large H/D ratio is advantageous. Other reactor shapes give similar results as the cylindrical core for optimum neutron economy (see Table 1.11).

Table 1.11. Optimum core dimensions and volumes for different core shapes.

Core shape	Optimum dimensions	Minimum volume
cube	$a = b = c = \frac{\sqrt{3}\,\pi}{B}$	$161/B^3$
cylinder	$H = \frac{\sqrt{3\pi}}{B}$, $R = \frac{2.405}{B}\sqrt{\frac{3}{2}}$	$148/B^3$
sphere	$R = \frac{\pi}{B}$	$130/B^3$

The composition of cores from moderator and fuel has strong influence on the criticality, as explained in Fig. 1.17. Owing to the energy dependence of cross sections, the curves show a strong dependence on the volume content of fuel for different moderators.

Without any moderator a critical mass of around 50 kg, corresponding to a radius of 8.6 cm, would be necessary for the unreflected system. Using more moderator material inside the core, the necessary amount of uranium drops very much, because in the area of thermal spectra the cross sections for fission rise up significantly. The smallest amount of uranium to make an H_2O-moderated system critical would be around 1.5 kg.

In all cases of moderators a minimum critical mass is possible. If the amount of moderator is increased further, the critical mass rises up, too, because the moderator absorbs more neutrons. An exception is D_2O which causes nearly no parasitic absorption.

Real power reactors today are operated with low enrichment and contain large amounts of U-238. Therefore the level of enrichment and the resonance absorption of U-238 influence the critical mass drastically, as shown in Fig. 1.18. Even at small values of enrichment (4 % in LWRs) a relatively small mass of U-235 is sufficient to make the reactor critical (around 4 kg).

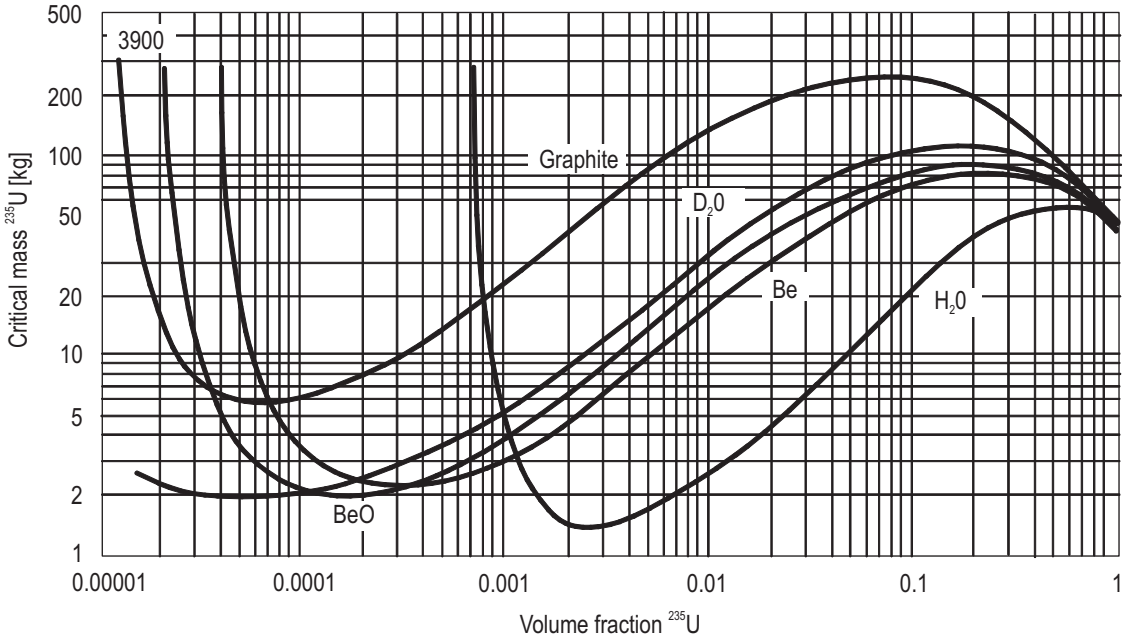


Fig. 1.17. Critical mass of a spherical core (necessary amount of ^{235}U in the core) for different moderator materials, as a function of the volume fraction of ^{235}U in the moderator (without reflector, 93 % enriched uranium).

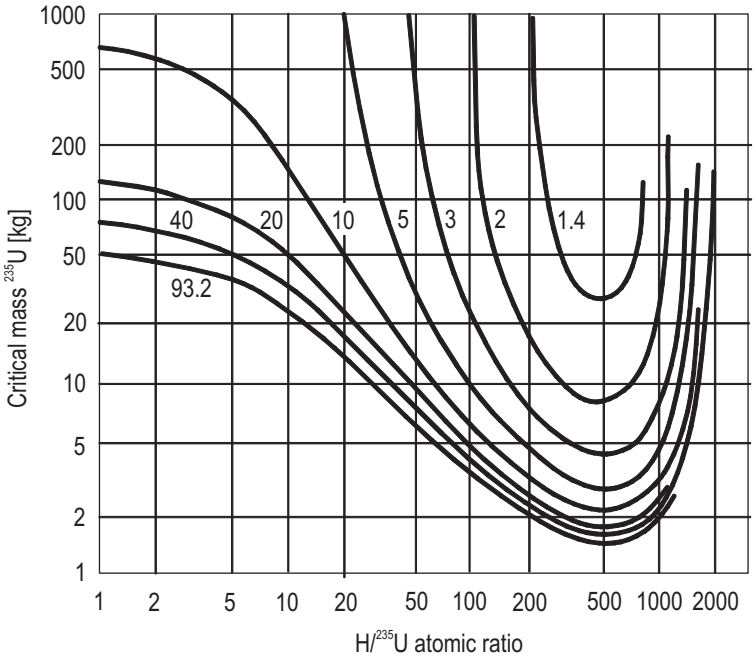


Fig. 1.18. Critical masses of unreflected reactors with variable enrichment ($a = {}^{235}U/({}^{235}U + {}^{238}U)$) and H_2 as moderator.

In technical applications the reactor core is surrounded by a very efficient reflector which reduces the critical masses and core dimensions. Some aspects are given in Sect. 1.1.6. The critical equation in the simplified form,

$$k_{\text{eff}} = k_\infty \cdot \frac{\mathrm{e}^{-B^2 \cdot \tau}}{1 + L^2 \cdot B^2} = 1 , \tag{1.118}$$

can be used to estimate the material composition of reactor cores or to find the dimension of a critical reactor. If the material composition and the inner structure of fuel elements and core are given, i.e. if the parameters k_∞, L^2 and τ are known, the material buckling, B_{m}^2, can be found from the relation

$$B_{\text{m}}^2 = \frac{k_\infty \cdot \mathrm{e}^{-B^2 \tau} - 1}{L^2} . \tag{1.119}$$

The criticality condition requires $B_{\text{m}}^2 = B_{\text{g}}^2$, so that using the expression for the cylindrical reactor,

$$B_{\text{g}}^2 = \left(\frac{\pi}{H} \right)^2 + \left(\frac{2.405}{R} \right)^2 , \tag{1.120}$$

one obtains an equation

$$f(k_\infty, \tau, L^2, H, R) = 0 , \tag{1.121}$$

which has to be solved with the condition $H/D \approx 0.9$ for an optimal reactor. By this way one finds the reactor dimensions. If the reactor dimensions D, H are given, the geometric buckling, B_{g}^2 can be calculated. From the criticality condition, $B_{\text{m}}^2 = B_{\text{g}}^2$, the values of k_∞, L^2 and τ can be estimated, if some additional parameters as the fuel element design are fixed. One very important parameter in this process of designing the reactor core, which allows to change k_∞, is the fuel enrichment.

1.1.6 Neutron balance and heat production in an LWR core

1.1.6.1 Neutron balance of the core

Design and layout of nuclear reactor cores are carried out today using large and complex computer programs. In order to gain better understanding of single effects, the discussion of the factors of the four-factor formula for a critical reactor is helpful even today. For the infinite reactor system,

$$k_\infty = \varepsilon \cdot f \cdot \eta \cdot p \tag{1.122}$$

is valid as mentioned above. Figure 1.19 shows the dependence of the single factors on important parameters, mainly on the ratio of water to fuel volume in a unit cell.

Following simplified reactor statics the thermal utilization is given by the expression

$$f = \frac{N_{\text{f}} \cdot \sigma_{\text{af}} \cdot \phi_{\text{f}} \cdot V_{\text{f}}}{N_{\text{f}} \cdot \sigma_{\text{af}} \cdot \phi_{\text{f}} \cdot V_{\text{f}} + N_{\text{M}} \cdot \sigma_{\text{aM}} \cdot \phi_{\text{M}} \cdot V_{\text{M}}} , \tag{1.123}$$

where the index f characterizes the fuel and M the moderator. Using the ratio $x = V_{\text{M}} / V_{\text{f}} = V_{\text{H}_2\text{O}} / V_{\text{UO}_2}$, one obtains the following relation (see Fig. 1.20):

$$f = \frac{1}{1 + c \cdot x} . \tag{1.124}$$

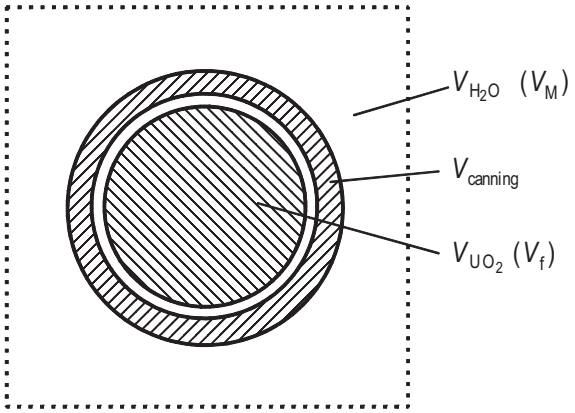


Fig. 1.19. Cross section through the unit cell of a PWR fuel element.

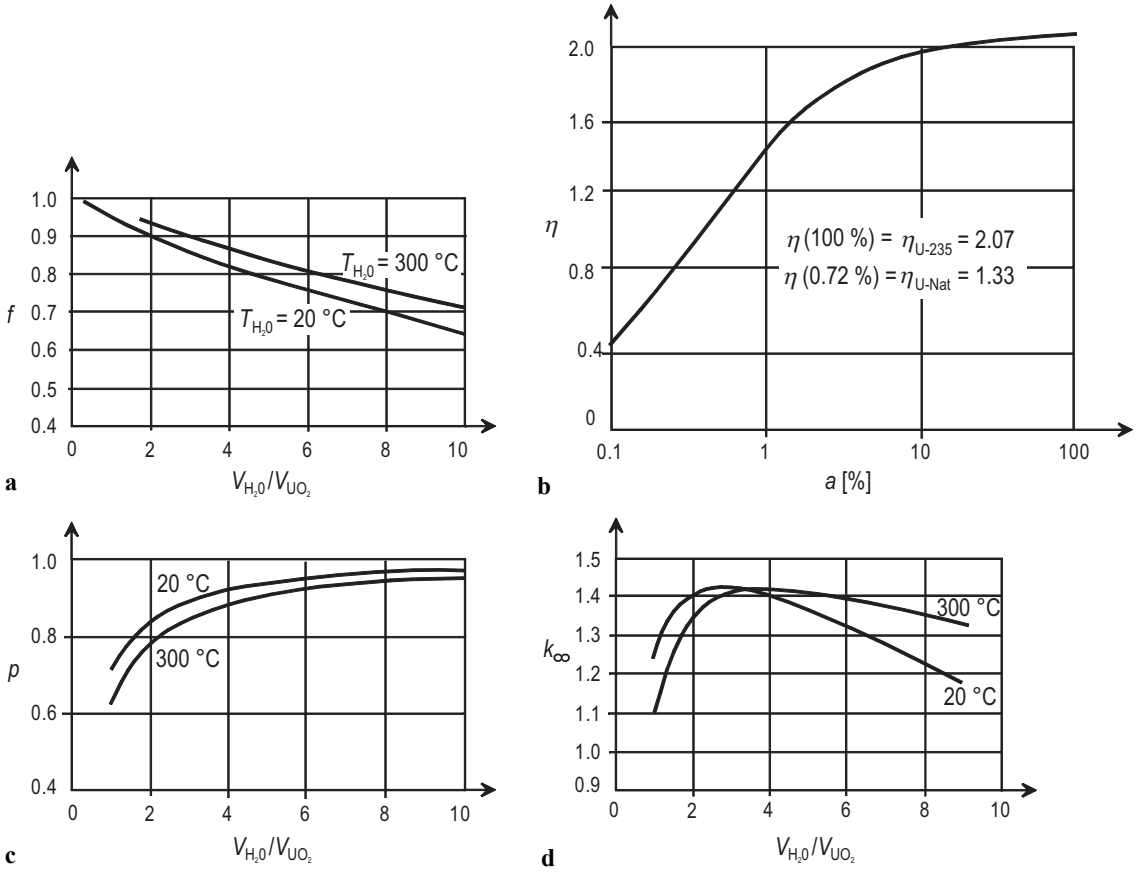


Fig. 1.20. Parameters of the four-factor formula applied to a PWR: **(a)** factor f vs ratio of moderator to fuel volume, V_{H_2O}/V_{UO_2}, **(b)** factor η vs enrichment, a, **(c)** factor p vs V_{H_2O}/V_{UO_2}, **(d)** factor k_∞ vs V_{H_2O}/V_{UO_2}.

The thermal utilization is influenced by the ratio of water to uranium oxide and by the temperature. The PWR nearly behaves like a homogeneous system related to this parameter, f increases with rising temperature. Therefore a positive moderator temperature coefficient can occur at some core conditions. The parameter ε is almost independent of x. It can be calculated from the expression

$$\varepsilon = \frac{\int_0^{\infty} \nu(E)\cdot\Sigma_\mathrm{f}(E)\cdot\phi(E)\cdot\mathrm{d}E}{\int_0^{E_\mathrm{th}} \nu(E)\cdot\Sigma_\mathrm{f}(E)\cdot\phi(E)\cdot\mathrm{d}E} \tag{1.125}$$

and depends on the neutron spectrum as well as on the geometrical conditions of the fuel elements. In LWR cores, $\varepsilon \approx 1.05$ to 1.1 is a characteristic value. This means that 5 to 10 % of the neutrons are produced by fissions in the fast part of the spectrum.

The value of η, describing the number of neutrons available per absorption in the fuel, is given by

$$\eta = \frac{\nu\cdot\Sigma_\mathrm{f}}{\Sigma_\mathrm{a}} = \frac{\nu\cdot\sigma_\mathrm{f}\cdot a}{\sigma_\mathrm{a}(\text{U-235})\cdot a + (a-1)\cdot\sigma_\mathrm{a}(\text{U-235}) + 2\cdot\sigma_\mathrm{a}(\text{O})} \tag{1.126}$$

and depends mainly on the fuel enrichment, a. For a typical thermal reactor with an enrichment of $a = 3$ %, the average value of η will be 1.8 including the influence of a thermal spectrum.

For the resonance escape factor, p, the theory explained above yields

$$p = \exp\left(-\int_{E_1}^{E_2} \frac{\Sigma_\mathrm{a}(E)}{\xi\cdot\Sigma_\mathrm{s}(E)}\cdot\frac{\mathrm{d}E}{E}\right) \tag{1.127}$$

for a homogeneous system, and similarly including aspects of heterogeneous arrangement of fuel and moderator,

$$p \approx \exp\left(-\frac{c(T,\text{geometry})}{x}\right), \tag{1.128}$$

where p mainly depends on the water/UO_2 ratio and on the fuel temperature. With the increase of temperature the value of p decreases and therefore the system has a strong negative temperature coefficient of fuel, which is fundamentally important for the safety aspects of reactor systems.

The value of k_∞ depends on x, too, following the rough approximation:

$$k_\infty(x) \approx \frac{1}{1+c\cdot x}\cdot\exp\left(-\frac{c^*}{x}\right). \tag{1.129}$$

Figure 1.21 shows this typical feature for a PWR core. It is necessary to operate the reactor with a value of x which is on the left side of the maximum of k_∞ (for example x ≈ 2), because just in this case in a loss-of-coolant accident there is a negative void coefficient of the core. It is a very important requirement of reactor safety always to have a negative void coefficient beside a strong negative temperature coefficient.

From an exact calculation of all reaction rates of the core of a real reactor one gets the neutron balance for the total core including all neutron losses in the moderator, in the structural materials of the core, in the fission products, and losses by leakage through the core surface. Table 1.12 contains the results of this analysis for a reactor with fresh, just loaded fuel and for a higher burn-up state of the core.

The leakage is relatively small, around 3.8 %, especially in a fresh core a large amount of neutrons (14.6 %) is absorbed in B-10 in the coolant and in fixed burnable poisons. This is necessary because of the burn-up compensation caused by the yearly discontinuous loading of fuel. This effect requires a high excess reactivity which is mainly compensated by the boron that is added in variable quantity to the coolant.

With rising burn-up the amount of boron inside the coolant is reduced. Naturally this variable addition of neutron poison influences the status of criticality of the core, as shown in Fig. 1.21. The adequate borating of the coolant is safety-relevant, too. Particularly the emergency cooling systems must always contain the specified amount of boron.

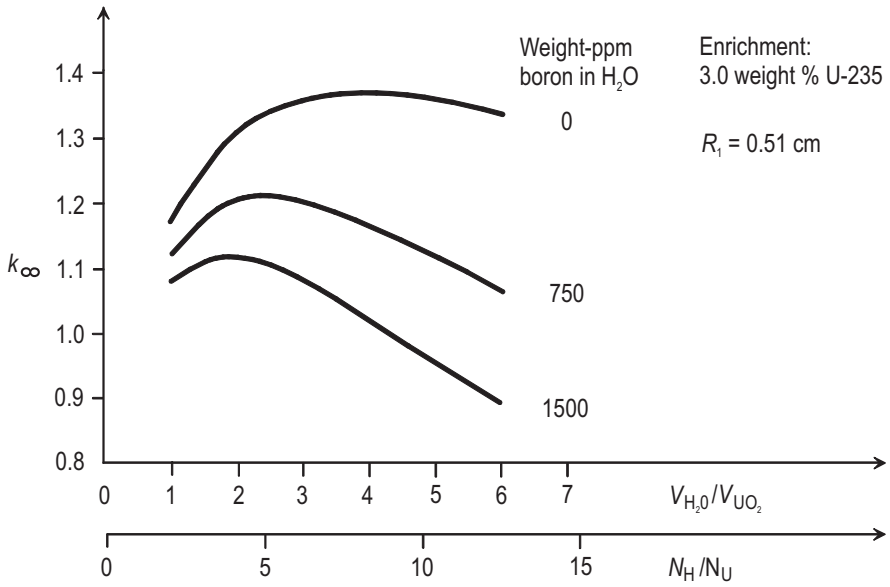


Fig. 1.21. Value of k_∞ versus the ratio of water volume to uranium-oxide volume, for different values of boron content in the coolant.

Table 1.12. Neutron balance of a PWR (1300 MW (el)) core, related to 100 fast fission neutrons formed.

	Freshly loaded reactor			Burn-up condition, 21645 MWd/t U		
Losses by	captures	fissions	total	captures	fissions	total
U-235	8.7	37.5	46.2	4.1	17.5	21.6
U-236	—	—	—	0.4	—	0.4
U-238 [a])	9.3	2.8	12.1	8.9	2.7	11.6
U-238 [b])	17.0	—	17.0	15.9	—	15.9
Pu-239	—	—	—	8.8	14.9	23.6
Pu-240	—	—	—	3.9	—	3.9
Pu-241	—	—	—	0.6	2.2	2.7
Sm-149	—	—	—	0.5	—	0.5
Xe-135	—	—	—	2.0	—	2.0
fission products	—	—	—	6.7	—	6.7
Zircaloy	1.2	—	1.0	1.3	—	1.3
steel/INCONEL	1.4	—	1.4	0.8	—	0.8
B-10 in the coolant	11.3	—	11.3	0.6	—	0.6
burnable poisons	3.3	—	3.3	—	—	—
H_2O	4.1	—	4.7	4.6	—	4.6
radial leakage losses	2.6	2.6	2.6	2.8	2.8	2.8
axial leakage losses	0.3	0.3	0.3	1.0	1.0	1.0
			100.0			100.0

[a]) Captures and fissions in the "non-resonance energy range".
[b]) Captures and fissions in the resonance energy range.

1.1.6.2 Heat production in the core

The neutron flux in the core depends on the spatial parameters and on the neutron energy. The macroscopic cross sections of fission include the distribution of fuel in the core and are energy dependent, too. Therefore the fission rate in the total core is calculated by

$$R_{\mathrm{f}} = \int\limits_{(V)} \int\limits_{E=0}^{\infty} \Sigma_{\mathrm{f}}(E, \vec{r}) \cdot \phi\left(E, \vec{r}\right) \cdot \mathrm{d}E \cdot \mathrm{d}\vec{r} \,. \tag{1.130}$$

The thermal power of the reactor is then given by

$$P_{\mathrm{th}} = \bar{E}_{\mathrm{f}} \cdot R_{\mathrm{f}} \,, \tag{1.131}$$

using the energy per fission, $\bar{E}_{\mathrm{f}} = 200$ MeV. Applying the relations $\Sigma_{\mathrm{f}}(E,\vec{r}) = \sigma_{\mathrm{f}}(E,\vec{r}) \cdot N_{\mathrm{f}}(\vec{r})$ and

$$M_{\mathrm{f}} = \frac{N_{\mathrm{f}}}{L} \cdot A \cdot V \,, \tag{1.132}$$

where M_{f} denotes the mass of fissionable material in the core, L denotes the Loschmidt number, N_{f} is the number of fissionable kernels, V is the reactor volume, and A is the atomic mass of U-235, one obtains for the thermal power of the reactor:

$$P_{\mathrm{th}} = \frac{\bar{E}_{\mathrm{f}} \cdot M_{\mathrm{f}} \cdot L}{A \cdot V} \cdot \int\limits_{(V)} \int\limits_{E=0}^{\infty} \sigma_{\mathrm{f}}\left(E,\vec{r}\right) \cdot \phi\left(E,\vec{r}\right) \cdot \mathrm{d}E \cdot \mathrm{d}\vec{r} = \frac{\bar{E}_{\mathrm{f}} \cdot M_{\mathrm{f}} \cdot L}{A} \cdot \overline{\sigma_{\mathrm{f}} \cdot \phi} \,, \tag{1.133}$$

where $\overline{\sigma_{\mathrm{f}} \cdot \phi}$ is an averaged value in the reactor including spectral effects. The real consumption of fuel is calculated by the expression

$$m_{\mathrm{spec}} = 1.26\,\mathrm{g\ U\text{-}235/1\,MWd} \,, \tag{1.134}$$

because parallel to fission there is parasitic absorption in the reactor, too. In detailed analyses the in-situ fission of Pu-239 and Pu-241, which are produced in all reactors using natural or enriched uranium, have to be taken into account. For a large PWR (3800 MW (th)) with an enrichment of 4 % and an operation time of 360 days/a, a uranium demand of 37.5 t/a would result. Including the in-situ utilization of the plutonium isotopes, the demand of fresh enriched uranium will be around 30 t/a.

1.1.7 Some aspects of reactor physics

1.1.7.1 Burn-up of fissile materials and build-up of higher isotopes

The U-235 fuel in the reactor will be burned according to the equations

$$\begin{aligned} \frac{\mathrm{d}N(\text{U-235})}{\mathrm{d}t} &= -\overline{\sigma_{\mathrm{f}} \cdot \phi} \cdot N(\text{U-235}), \\ N(\text{U-235}) &= N_0(\text{U-235}) \cdot \exp(-\overline{\sigma \cdot \phi} \cdot t). \end{aligned} \tag{1.135}$$

In order to calculate the burn-up of the fissile material, the equation

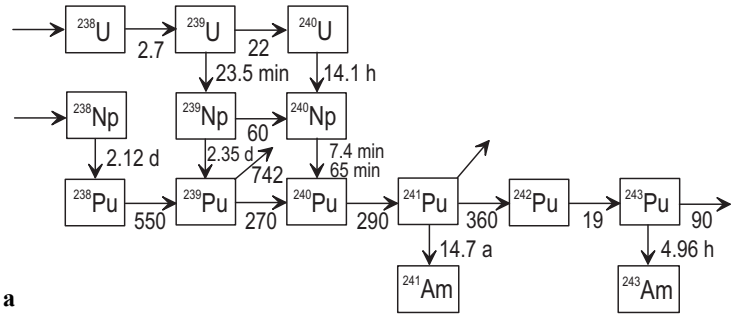
$$B = \frac{\bar{E}_{\mathrm{f}}}{\rho_{\mathrm{f}}} \cdot \int\limits_{0}^{\tau} \phi(t) \cdot \Sigma_{\mathrm{f}}(t) \cdot \mathrm{d}t = \frac{\bar{E}_{\mathrm{f}}}{\rho_{\mathrm{f}}} \cdot \bar{\phi} \cdot \bar{\Sigma}_{\mathrm{f}} \cdot \tau \tag{1.136}$$

is used, where $\bar{E}_{\mathrm{f}} = 200$ MeV, ρ_{f} is the fuel density, and τ is the time of fuel insertion. The unit is MW d/t. Practical values of burn-up realized today in commercial LWRs are 35000 to 50000 MW d/t, correspond-

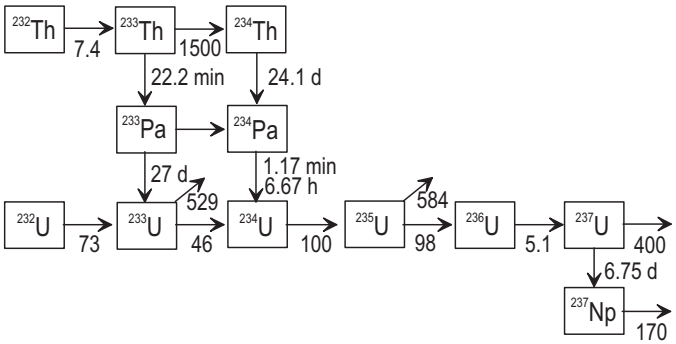
ing to an enrichment of 3 to 4.5 %. For high temperature reactors, burn-up values of more than 100000 MW d/t have been realized using higher enrichment.

The possible level of burn-up is given not only by the enrichment, but also by technical parameters of fuel design (internal pressure, neutron irradiation, embrittlement, corrosion). Inside the reactor core, parallel to the fission of U-235, there are reactions with the U-238 forming higher isotopes.

Figure 1.22 shows the chain for transformation of U-238, and in a similar way of Th-232, into higher isotopes.



a



b

Fig. 1.22. Transformation of **(a)** U-238, and **(b)** Th-232 into higher isotopes. The numbers at the horizontal arrows indicate thermal absorption cross sections in barn. The numbers at the arrows with 45 degree inclination indicate thermal fission cross sections in barn.

Corresponding to the simplified chain,

$$\underset{1}{{}^{238}_{92}\mathrm{U}} + {}^{1}_{0}\mathrm{n} \ldots \rightarrow \underset{2}{{}^{239}_{94}\mathrm{Pu}} \rightarrow \underset{3}{{}^{240}_{94}\mathrm{Pu}} \rightarrow \underset{4}{{}^{241}_{94}\mathrm{Pu}} \rightarrow \underset{5}{{}^{242}_{94}\mathrm{Pu}}, \tag{1.137}$$

plutonium and its higher isotopes are formed. The following system of differential equations, which are further simplified, describes the build-up in time of these substances:

$$\frac{\mathrm{d}N_1}{\mathrm{d}t} = -\overline{\sigma_1 \cdot \phi} \cdot N_1 , \tag{1.138}$$

$$\frac{\mathrm{d}N_2}{\mathrm{d}t} = \overline{\sigma_1 \cdot \phi} \cdot N_1 - \overline{\sigma_2 \cdot \phi} \cdot N_2 , \tag{1.139}$$

$$\frac{\mathrm{d}N_3}{\mathrm{d}t} = \overline{\sigma_2 \cdot \phi} \cdot N_2 - \overline{\sigma_3 \cdot \phi} \cdot N_3 , \tag{1.140}$$

$$\frac{\mathrm{d}N_4}{\mathrm{d}t} = \overline{\sigma_3 \cdot \phi} \cdot N_3 - \overline{\sigma_4 \cdot \phi} \cdot N_4 , \tag{1.141}$$

$$\frac{\mathrm{d}N_5}{\mathrm{d}t} = \overline{\sigma_4 \cdot \phi} \cdot N_4 . \tag{1.142}$$

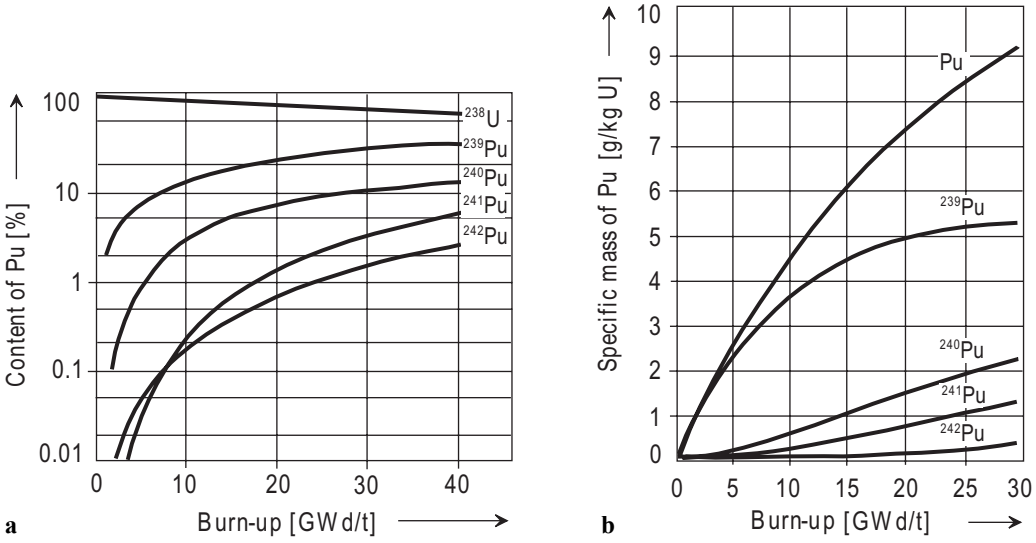


Fig. 1.23. Plutonium production in a PWR: **(a)** content of Pu versus burn-up, **(b)** specific mass of Pu related to uranium in spent fuel elements versus burn-up.

The result of a numerical evaluation of this system is given in Fig. 1.23. Overall the new fissionable isotopes Pu-239 and Pu-241 are produced in each reactor containing U-238, they are partly consumed by fission again, and partly they stay in the spent fuel elements.

Spent PWR fuel elements with a burn-up of 34 GWd/t, as an example, contain 9.3 g of plutonium per kg of uranium, around 6.5 g of this Pu is fissionable. Today already remarkable quantities of this material are recycled in LWR plants, the fresh fuel elements are inserted as MOX elements (mixed oxides), containing uranium and plutonium as fuel.

1.1.7.2 Building up of fission product inventory

In the fission process, radioactive fission products are formed. The fission products constitute the main part of radioactivity in the core with the total activity, A_t, and the following decay law for each isotope:

$$A_t = \sum_i \lambda_i \cdot N_i \,, \tag{1.143}$$

$$N_i(t) = N_i^0 \cdot e^{-\lambda_i t} \,. \tag{1.144}$$

The sum of these decay processes causes the decay heat production in all nuclear installations.

Furthermore the fission products are the source of the delayed neutrons which are necessary for the control of reactors as is explained below. Some fission products like xenon and samarium have very high absorption cross sections and therefore they influence the operation behavior of reactors significantly (see Sect. 1.1.7.3).

In (1.143) and (1.144), λ_i is the decay constant of a specific isotope. The half-life, $T_{1/2i}$, is defined by

$$T_{1/2i} = \ln 2 / \lambda_i \tag{1.145}$$

and can have values ranging from parts of seconds to 10^6 years depending on the type of isotope. The unit of activity is 1 Becquerel (= 1 decay/s).

In a nuclear reactor the production rate of fission products depends on the fuel and on the burn-up. In Fig. 1.24 a characteristic dependence is shown for LWR fuel. According to this example, around 32.5 kg of fission products are contained in 1 t of spent fuel after removal from the reactor core.

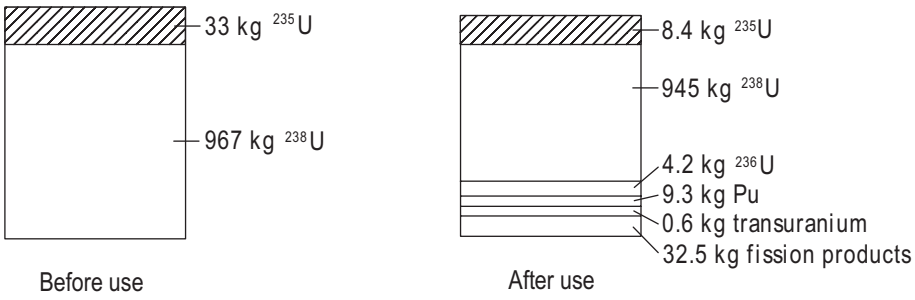


Fig. 1.24. Amount of heavy isotopes and fission products dependent on burn-up and enrichment (PWR, 3.3 % enrichment, 34 000 MW d/t burn-up).

After a longer operation time, an equilibrium value of radioactivity of around $8\times10^{16}\,\mathrm{Bq/MW(th)}$ has accumulated in the core of a large PWR. The retention of this large amount of radioactivity inside the reactor system in normal operation and in all accidents is the main requirement of reactor safety. In the field of waste disposal, it is the main requirement, too, to avoid unallowed release of radioactivity to the biosphere.

The time-dependent nuclide densities in the reactor core can be calculated for each fission product by solving differential equations of the following type:

$$\frac{\mathrm{d}N_i}{\mathrm{d}t}=\sum_k \gamma_i\cdot\int_0^\infty \phi\cdot\sigma_{\mathrm{f}_k}\cdot N_k\cdot\mathrm{d}E+\int_0^\infty \phi\cdot\sigma_{i-1}\cdot N_{i-1}\cdot\mathrm{d}E$$
$$+\sum_j \lambda_j\cdot N_j-\lambda_i\cdot N_i-\int_0^\infty \phi\cdot\sigma_{\mathrm{a}_i}\cdot N_i\cdot\mathrm{d}E-L-D. \tag{1.146}$$

The first term on the right describes the generation of fission products by fission in k fissionable materials (U-235, U-233, Pu-239, Pu-241). The second term is caused by neutron capture in the isotope $i-1$, the third term includes production by α- and β^--decay, the fourth the decay of the isotope i itself. Furthermore, losses by absorption, by leakage L and by deloading of spent fuel from core, D, are sinks for the isotope i.

Today more than hundred fission products in balance equations are included in calculations regarding reactor physics and fission products inventories. Table 1.13 shows data of some important fission products in a PWR core.

The fission product inventory is built up according to the equation mentioned above. After shutdown of the reactor, or if the fuel elements are removed from the reactor, the inventory is reduced by radioactive decay. Characteristic equations for the energy released by β^-- and γ-decays are:

$$E_\beta(t)=1.26\cdot t^{-1.2}\ \mathrm{MeV/s},$$
$$E_\gamma(t)=1.40\cdot t^{-1.2}\ \mathrm{MeV/s}. \tag{1.147}$$

Adding up these functions delivers the well-known decay heat function, $P_\mathrm{D}(t)$:

$$P_\mathrm{D}(t)=P_\mathrm{th}\cdot0.06\cdot\left(t^{-0.2}-(t+t_0)^{-0.2}\right)\quad(t,t_0\ \text{in s}), \tag{1.148}$$

with t_0 as the time the reactor has been in operation before shutdown.

Table 1.13. Data of some important fission products (PWR core, 3800 MW (th); 1 Ci = 3.7×10^{10} Bq).

Nuclide	Inventory [10^6 Ci]	Radiation type	Energy [MeV]	Half-life ($T_{1/2}$)
Krypton-88	82	γ	1.55	2.77 h
Strontium-89	113	β^-	1.46	51 d
Strontium-90	—	β^-	0.5	28.5 a
Tellurium-132	144	β^-, γ	—	77 h
Antimony-129	40	β^-	—	4.2 h
Iodine-131	102	β^-, γ	0.61/0.8/0.36/0.64	8.0 d
Iodine-132	144	β^-, γ	—	2.4 h
Iodine-133	204	β^-, γ	—	20.5 h
Iodine-135	180	γ	1.8	6.75 h
Cesium 134	9	β^-	0.6	2.2 a
Cesium-137	6	β^-, γ	0.51/1.7/0.6	30 a
Barium-140	192	β^-	1	12.8 d
Lanthanum-140	192	γ	1.6	1.68 d

Figure 1.25 shows the time dependence of the fission product content of spent uranium, characterized by decay heat and activity after a long time. After some decades the isotope Cs-137 and Sr-90 (with half-lives of around 30 years) dominate the activity together with the actinides.

These data are important for the intermediate storage, later conditioning or reprocessing, and for the final storage of spent fuel elements. Beside keeping subcriticality and retaining radioactivity, reliably removing the decay heat and limiting the fuel temperature below allowable values is one of the main safety requirements in all storage systems.

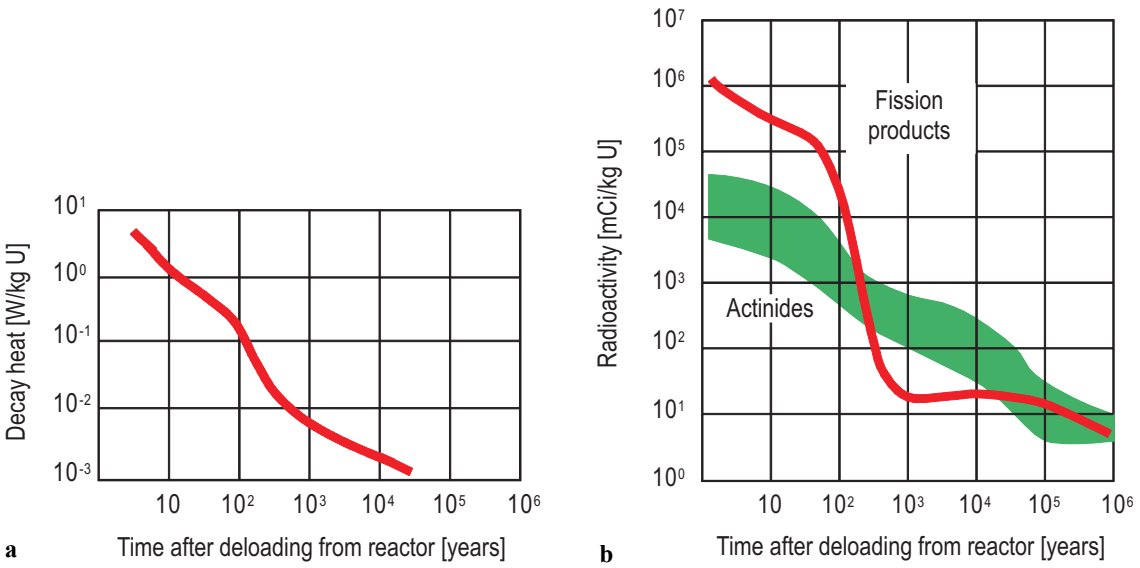


Fig. 1.25. **(a)** Decay heat and **(b)** long-term activity of spent fuel elements (PWR, burn-up 35 000 MW d/t).

1.1.7.3 Xenon and samarium poisoning

There are two fission products which influence the neutron balance in the reactor very much, xenon-135 and samarium-149. They have relatively high absorption cross sections in thermal reactors (see Fig. 1.26) and they decay with half-lives in the order of hours. Therefore they can change the reactivity state of reactors corresponding to those time schedules.

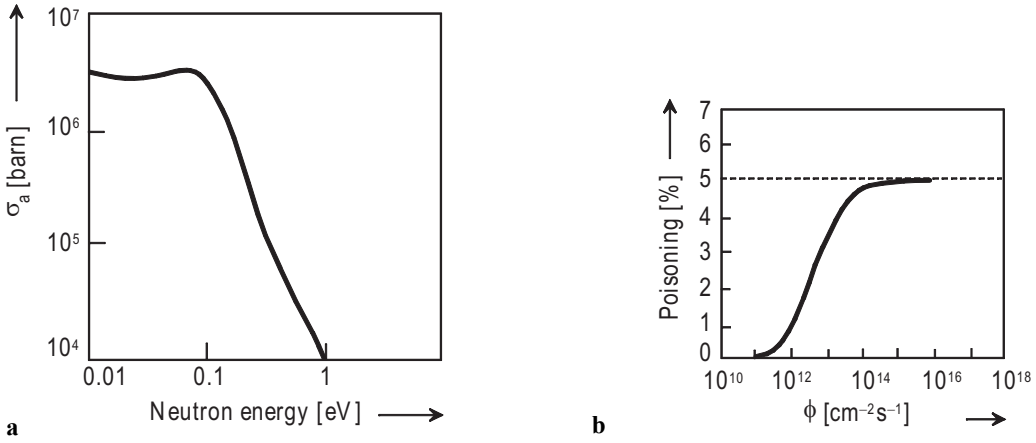


Fig. 1.26. Xenon poisoning in thermal reactors: **(a)** cross section for parasitic absorption by Xe-135 versus neutron energy, **(b)** equilibrium values of Xe-135 poisoning during reactor operation as a function of the neutron flux, ϕ.

Xenon-135 is an intermediate decay product of the following fission product chain:

$$^{135}\mathrm{Te} \underset{18\,\mathrm{s}}{\rightarrow} {}^{135}\mathrm{I} \underset{6.61\,\mathrm{h}}{\rightarrow} {}^{135}\mathrm{Xe} \underset{9.1\,\mathrm{h}}{\rightarrow} {}^{135}\mathrm{Cs} \underset{2\times10^6\,\mathrm{a}}{\rightarrow} {}^{135}\mathrm{Ba}. \qquad (1.149)$$

1 *2*

Samarium-149, a stable end product, is formed by the chain starting from neodymium:

$$^{149}\mathrm{Nd} \underset{1.73\,\mathrm{h}}{\rightarrow} {}^{149}\mathrm{Pm} \underset{53.1\,\mathrm{h}}{\rightarrow} {}^{149}\mathrm{Sm}. \qquad (1.150)$$

The absorption cross section of Xe has a value of around 3×10^6 barn at thermal energies and the fission yield in case of U-235 fuel is 0.3 %. As the isotope additionally decays with a half-life of 9.2 h and is built up by decay of the fission product iodine-135 with half-life 6.7 h (the fission yield of I-135 is 6.1 %), the neutron balance is strongly influenced by the xenon in the core.

The coupled differential equations of the time dependence of I and Xe in (1.149) are:

$$\frac{\mathrm{d}N_1}{\mathrm{d}t} = \gamma_1 \cdot \Sigma_\mathrm{f} \cdot \phi - \lambda_1 \cdot N_1, \qquad (1.151)$$

$$\frac{\mathrm{d}N_2}{\mathrm{d}t} = \lambda_1 \cdot N_1 + \gamma_2 \cdot \Sigma_\mathrm{f} \cdot \phi - \sigma_{\mathrm{a}_2} \cdot N_2 \cdot \phi - \lambda_2 \cdot N_2 . \qquad (1.152)$$

For the stationary state one gets

$$N_1^* = \frac{\gamma_1 \cdot \Sigma_\mathrm{f} \cdot \phi}{\lambda_1}, \qquad (1.153)$$

$$N_2^* = \frac{(\gamma_1 + \gamma_2) \cdot \Sigma_f \cdot \phi}{\lambda_2 + \sigma_{a_2} \cdot \phi} . \tag{1.154}$$

The stationary poisoning, V, of the core by xenon is defined by the expression

$$V = \frac{\sigma_{a_2} \cdot N_2^*}{\Sigma_f} = \frac{(\gamma_1 + \gamma_2) \cdot \phi}{\phi + \lambda_2 / \sigma_{a_2}} , \tag{1.155}$$

where V depends on the neutron flux in the core, as shown in Fig. 1.27.

In the core mainly the value of thermal utilization, β, is influenced by the xenon poisoning. For the reactivity change one approximately finds the relation

$$\rho = -f \cdot \frac{\eta}{\nu} \cdot V = -f \cdot \frac{\eta}{\nu} \cdot \frac{(\gamma_1 + \gamma_2) \cdot \phi}{\phi + \lambda_2 / \sigma_{a_2}} . \tag{1.156}$$

The maximal value of ρ for $\phi \to \infty$ is around $\rho = -3.1$ % in a typical thermal reactor ($\nu = 2.44$, $\eta = 1.32$, $\gamma_1 + \gamma_2 = 0.064$, $f = 0.9$).

After shutdown of the reactor the neutron flux vanishes ($\phi = 0$). The differential equations to be solved are now:

$$\frac{dN_1}{dt} = -\lambda_1 \cdot N_1 , \tag{1.157}$$

$$\frac{dN_2}{dt} = \lambda_1 \cdot N_1 - \lambda_2 \cdot N_2 , \tag{1.158}$$

with the starting condition ($t = 0$):

$$N_1^0 = \frac{\gamma_1 \cdot \Sigma_f \cdot \phi}{\lambda_1} , \tag{1.159}$$

$$N_2^0 = \frac{(\gamma_1 + \gamma_2) \cdot \Sigma_f \cdot \phi}{\lambda_2 + \sigma_{a_2} \cdot \phi} . \tag{1.160}$$

The solutions of the above differential equations are:

$$N_1(t) = \frac{\gamma_1 \cdot \phi \cdot \Sigma_f}{\lambda_1} \cdot e^{-\lambda_1 \cdot t} , \tag{1.161}$$

$$N_2(t) = \frac{(\gamma_1 + \gamma_2) \cdot \Sigma_f \cdot \phi}{\lambda_2 + \sigma_{a_2} \cdot \phi} \cdot e^{-\lambda_2 \cdot t} + \frac{\gamma_1 \cdot \Sigma_f \cdot \phi}{\lambda_2 - \lambda_1} \cdot \left(e^{-\lambda_1 \cdot t} - e^{-\lambda_2 \cdot t} \right) . \tag{1.162}$$

For the time-dependent xenon poisoning, V^*, one now obtains

$$V^* = \frac{\sigma_{a_2} \cdot N_2(t)}{\Sigma_f} = V^*(\phi, t) . \tag{1.163}$$

The function $V^*(\phi, t)$ is shown in Fig. 1.27, indicating the large influence of the flux, ϕ, on the time-dependent behavior.

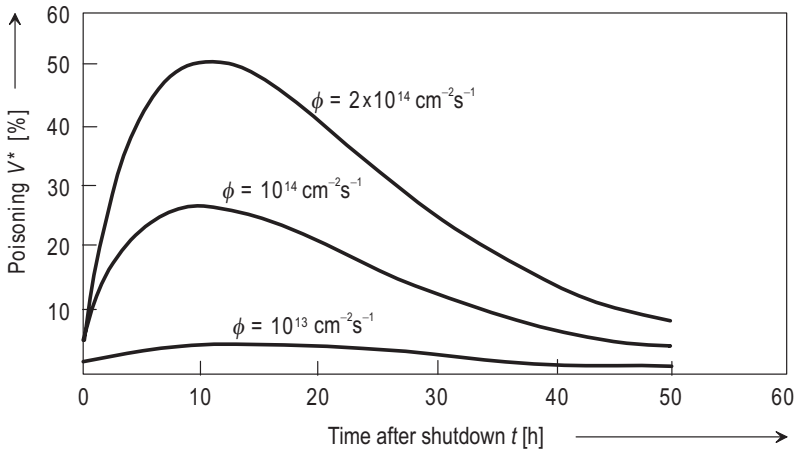


Fig. 1.27. Time-dependent xenon poisoning, V^*, with ϕ as parameter.

After shutdown of the reactor the xenon poisoning, V^*, is rising up, because xenon is still produced by decay of iodine, whereas no xenon disappears by absorption of neutrons. The maximum of the xenon "hill" appears after 11 hours. The reactor can be started again if an excess reactivity corresponding to the xenon poisoning, V^*, is available in the core. This value must indeed be compensated by the control and shutdown system. As an example, a reactor with a neutron flux of 3×10^{13} cm^{-2}s^{-1} requires around 5 % excess reactivity if it should be started at any time after shutdown.

Alternately the reactor would have to be restarted soon after the shutdown, because in this case the xenon would not already build up until the maximal value of poisoning. The xenon effect is characteristic just for thermal reactors because the absorption cross section of in xenon is very high just at thermal energies. Intermediate or fast spectrum reactions do not suffer from this aspect. In Fig. 1.28 some characteristic transients in a large PWR core are shown displaying the influence of xenon poisoning.

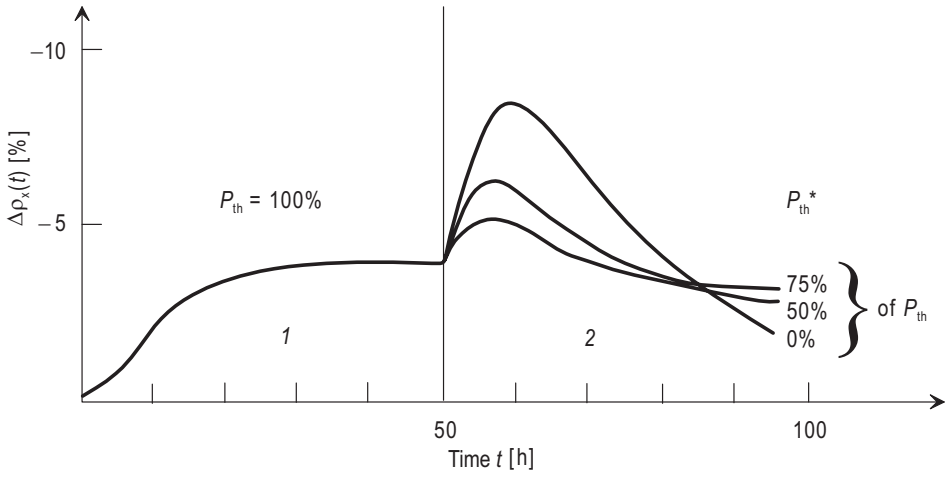


Fig. 1.28. Time-dependent change of xenon reactivity in the core of a PWR after change of load (P_{th} = 100 %, P_{th}* at variable level).

In reactors with very large core dimensions, spatial xenon oscillations can occur, which have to be controlled by the control systems.

The stable fission product samarium-149 is built up from the fission product neodymium-149 with promethium-149 as an intermediate decay product:

$$\underset{1}{^{149}_{60}\mathrm{Nd}} \xrightarrow[1.73\,\mathrm{h}]{\beta^-} \underset{2}{^{149}_{61}\mathrm{Pm}} \xrightarrow[5.31\,\mathrm{h}]{\beta^-} \underset{3}{^{149}_{62}\mathrm{Sm}} \ \text{(stable)}. \tag{1.164}$$

As the half-life of Nd is short compared to that of Pm, the latter can be considered as the originally produced fission product ($\gamma_1 = 0.0113$, $\gamma_2 = 81$). The fission yield of Sm is very small. For the isotopes one obtains the equations:

$$\frac{\mathrm{d}N_2}{\mathrm{d}t} = \gamma_2 \cdot \Sigma_\mathrm{f} \cdot \phi - \lambda_2 \cdot N_2 , \tag{1.165}$$

$$\frac{\mathrm{d}N_3}{\mathrm{d}t} = \lambda_2 \cdot N_2 - \sigma_{\mathrm{a}_3} \cdot \phi \cdot N_3 . \tag{1.166}$$

For the equilibrium the number of nuclei is given by

$$N_2^* = \frac{\gamma_2 \cdot \Sigma_\mathrm{f} \cdot \phi}{\lambda_2} , \tag{1.167}$$

$$N_3^* = \frac{\gamma_2 \cdot \Sigma_\mathrm{f}}{\sigma_{\mathrm{a}_3}} . \tag{1.168}$$

The reactivity loss caused by samarium can be calculated roughly from

$$\rho = -\frac{f \cdot \sigma_{\mathrm{a}_3} \cdot N_2}{\sigma_{a\,\mathrm{fuel}} \cdot N_\mathrm{fuel}} = -f \cdot \frac{\eta}{\nu} \cdot \gamma_2 . \tag{1.169}$$

The equilibrium concentration of samarium occurs in a relatively short time, after some days the core reaches a value of $\rho = -0.046$ (see Fig. 1.29). After shutdown ($\phi = 0$), the equations

$$\frac{\mathrm{d}N_2}{\mathrm{d}t} = -\lambda_2 \cdot N_2 , \tag{1.170}$$

$$\frac{\mathrm{d}N_3}{\mathrm{d}t} = \lambda_2 \cdot N_2 \tag{1.171}$$

have to be considered. With the starting conditions for $t = 0$, $N_2 = N_2^0$, $N_3 = N_3^0$, one has:

$$N_2(t) = N_2^0 \cdot \mathrm{e}^{-\lambda_2 \cdot t} , \tag{1.172}$$

$$N_3(t) = N_2^0 \cdot \left(1 - \mathrm{e}^{-\lambda_2 \cdot t}\right) + N_3^0 . \tag{1.173}$$

The rising burn-up of the fuel causes a reduction of reactivity in the core, as shown in Fig. 1.30. The reason is the growing poisoning of the core by neutron-absorbing fission products. This effect causes a limitation of the burn-up level that is achievable for reactors with discontinuous loading. In case of reactors with continuous loading and deloading this limitation is not given. These reactors can be operated almost without excess reactivity for burn-up compensation. The high temperature reactor is a typical example for this feature, the CANDU and RBMK reactors are using nearly continuous loading. To avoid excess reactivity is a very important safety feature in connection with severe reactivity accidents. Higher burn-ups naturally require higher enrichment of fuel.

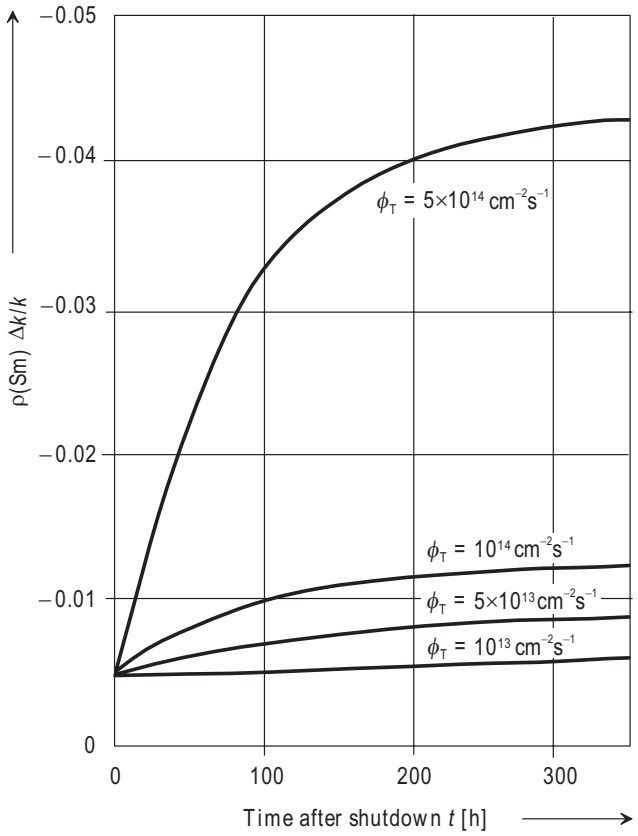


Fig. 1.29. Reactivity value caused by samarium poisoning after shutdown of the reactor, the parameter being the thermal flux in the operation time before shutdown.

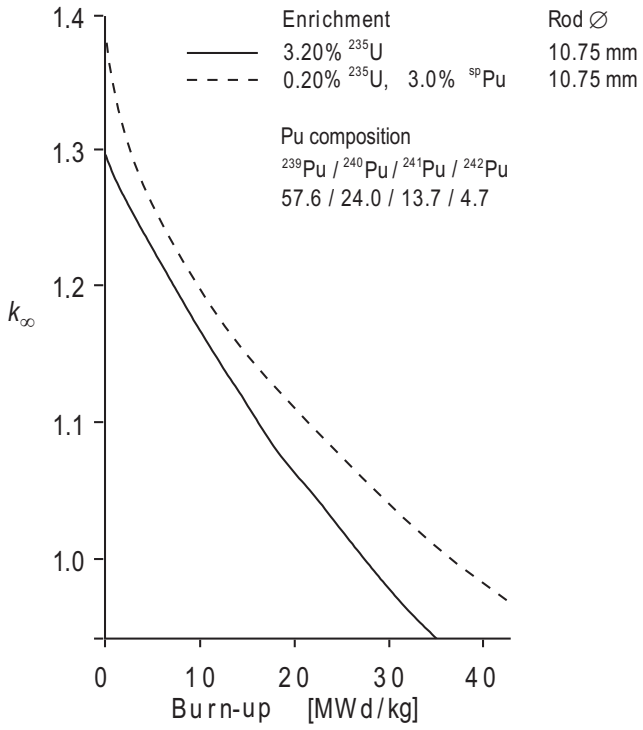


Fig. 1.30. Criticality factor dependent on burn-up of fuel; parameter is the type of fuel.

1.1.7.4 Reactivity coefficients

The reactivity of the core, $\rho = \Delta k$, can be changed by many different reasons. By movements of control elements, by changing the state of the coolant, especially by changes of densities and temperatures, as well as of the fuel, k_{eff} is changed. In a simplified consideration for a finite reactor arrangement one obtains if, for instance, T is changed:

$$k_{\mathrm{eff}} = \varepsilon \cdot p \cdot f \cdot \eta \cdot W_{\mathrm{th}} \cdot W_{\mathrm{f}} = \prod_i \xi_i \,, \tag{1.174}$$

$$\frac{1}{k_{\mathrm{eff}}} \cdot \frac{\mathrm{d}k_{\mathrm{eff}}}{\mathrm{d}T} = \sum_i \frac{1}{\xi_i} \cdot \frac{\partial \xi_i}{\partial T} \,. \tag{1.175}$$

Changes of reaction rates can be calculated, too:

$$R = \sigma \cdot \phi \cdot N \,, \tag{1.176}$$

$$\frac{1}{R} \cdot \frac{\mathrm{d}R}{\mathrm{d}T} = \frac{1}{\sigma} \cdot \frac{\mathrm{d}\sigma}{\mathrm{d}T} + \frac{1}{\phi} \cdot \frac{\mathrm{d}\phi}{\mathrm{d}T} + \frac{1}{N} \cdot \frac{\mathrm{d}N}{\mathrm{d}T} \,. \tag{1.177}$$

As a very important example the fuel temperature coefficient can be considered. If the reactor power increases, the fuel temperature rises as well. The resonance absorption in the futile materials, U-238 or Th-232, becomes larger by this effect, because the resonances are broadened by the thermal movement of the absorbing nuclei (see Fig. 1.31).

The absorption rate,

$$R_{\mathrm{a}} = \int \sigma_{\mathrm{a}} \cdot \phi(E) \cdot N \cdot \mathrm{d}E \,, \tag{1.178}$$

becomes larger if the temperature rises.

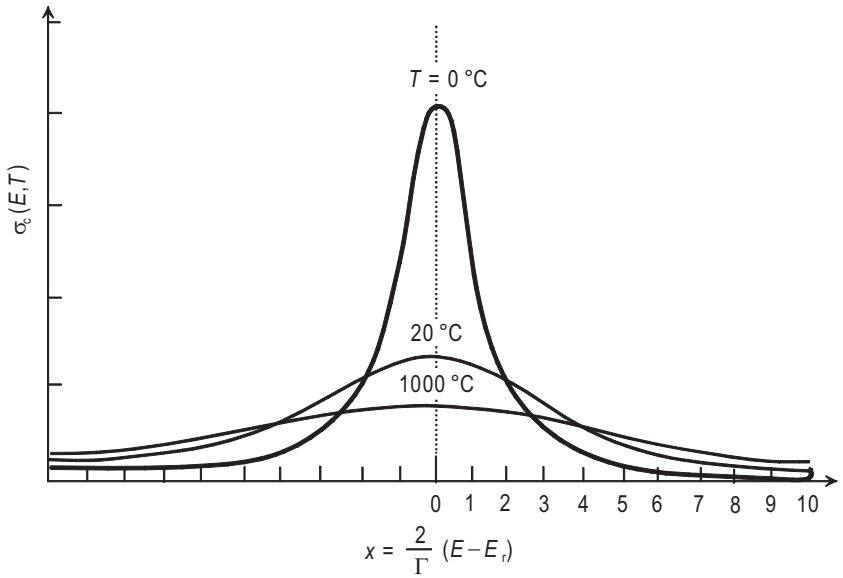


Fig. 1.31. Broadening of a characteristic resonance line of U-238 with rising temperature (Doppler broadening).

The higher parasitic absorption rate causes a reduction of the criticality constant, k. The total effect results in a self-acting stabilization of power and temperature in case of rising reactor power. This inherent mechanism of regulation is fundamentally important for the safety of nuclear reactors. The temperature coefficient of fuel is defined by the relations:

$$\rho_{\text{fuel}} = \int_{T_{f_1}}^{T_{f_2}} \Gamma_f(T_f) \cdot dT_f \approx \Gamma_f \cdot \left(T_{f_2} - T_{f_1}\right), \tag{1.179}$$

$$\Gamma_f \approx \frac{\partial \rho_f}{\partial T_f}. \tag{1.180}$$

Exact calculations of these temperature coefficients are carried out today with the help of detailed computer programs. Characteristic values of the temperature coefficient of PWR fuel are between -3×10^{-5} °C^{-1} (zero power) and -1.6×10^{-5} °C^{-1} (full power). Under theses conditions a reactivity change of 1 ‰ during normal operation corresponds to a temperature rise in the fuel of around 60 °C. In short-time load changes the fuel temperature coefficient is mainly relevant for the behavior of reactor power and fuel temperature. In case of quasi-stationary changes a power coefficient can be defined. Furthermore a moderator coefficient can be defined (ρ_M). This coefficient is relevant if the moderator temperature or density are changed.

Higher temperatures in the reactor cause a reduction of the cross section of fissionable material ($1/v$-law). Therefore the reactivity change is negative. In addition, at rising temperature the density of the moderator is reduced, which leads to reduced parasitic absorption of neutrons in the moderator. This is the reason for a positive contribution to ρ_M. In total the moderator effect can be described by:

$$\rho_M = \int_{T_{M_1}}^{T_{M_2}} \Gamma_M(T_M) \cdot dT_M \approx \Gamma_M \left(T_{M_2} - T_{M_1}\right), \tag{1.181}$$

$$\Gamma_M \approx \frac{\partial \rho_M}{\partial T_M}. \tag{1.182}$$

In case of light water reactors, ρ_M depends on the concentration of boron acid in the coolant, too, and is changed during operation corresponding to progressing burn-up. Figure 1.32 shows the coolant temperature coefficient as a function of the boron concentration.

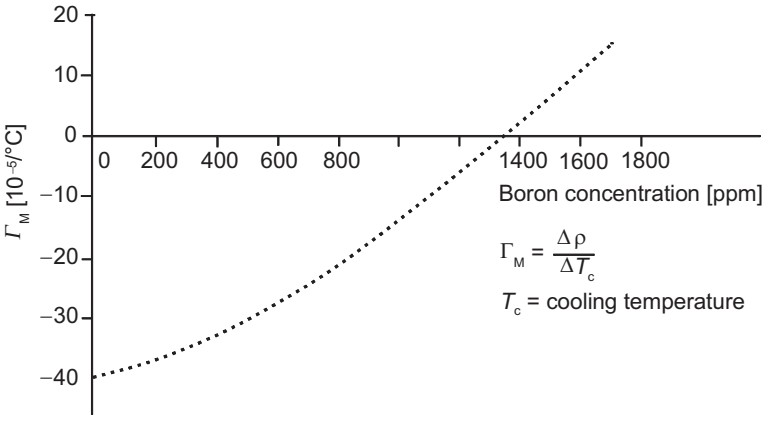


Fig. 1.32. Coolant temperature coefficient of reactivity as a function of boron concentration (PWR, T_{core} = 290 °C).

If a large amount of boron is available in the coolant a reduction of density can cause a loss of absorber in the core, and therefore a slightly positive coolant coefficient is possible for a freshly loaded core which has a high content of boron.

Void coefficients of reactors are fundamentally important for the safety behavior of water-moderated reactors. They must always be negative, as already explained in Sect. 1.1.6.1. The operation point (OP) of reactors has to be chosen on the left side of the maximum of the curve in Fig. 1.33.

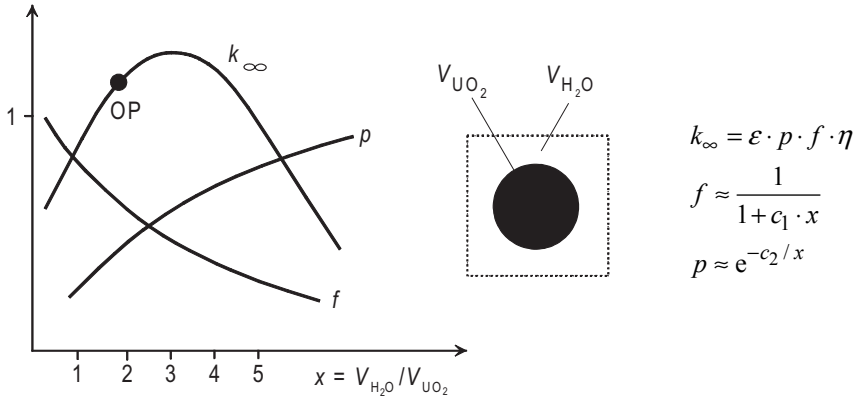


Fig. 1.33. Criticality constant, k_∞, as well as parameters p and f, versus ratio of water to fuel. In addition, a schematic of PWR-type fuel is shown.

Practically the ratio V_{H_2O}/V_{UO_2} in a large PWR is around 2, plutonium in the core (MOX fuel) shifts this ratio to higher numbers because of differences in cross sections and spectra. An always negative void coefficient guarantees that the core of LWRs will be shut down immediately in loss-of-coolant accidents. Values of the void coefficient of $\Gamma_{void} \approx -0.01/\%$ (related to the percentage of water volume lost from the core) are typical for large PWR cores.

1.1.7.5 Time behavior of reactors, kinetic equations

The time-related behavior of a reactor is fundamentally influenced by the delayed neutrons in the reactor. To identify the impact of delayed neutrons on the dynamic processes in a reactor, first of all a consideration without delayed neutrons shall be carried out.

The reactor shall be even critical at time $t = 0$, so that $k = 1$. With a slight alteration of k the reactor then becomes super-critical. The life-span of prompt neutrons is dnoted τ_P. Since in thermal systems the diffusion time, τ_{Diff}, is much longer than the slowing-down time, the quantity τ_{Diff} can be used to decribe the time-dependent behavior in the reactor (for this first approximation without delayed neutrons). In thermal systems, for instance,

$$\tau_P \approx \tau_{Diff} \approx \frac{1}{\Sigma_a \cdot v_{th}} . \tag{1.183}$$

For successive generations of neutrons the relation

$$n(t+\tau_P) = k \cdot n(t) \tag{1.184}$$

applies. With the help of a Taylor series one obtains

$$n(t+\tau_P) \approx n(t) + \frac{dn}{dt} \cdot \tau_p + \dots , \tag{1.185}$$

so that the resulting differential equation for the number of neutrons is

$$\frac{dn}{dt} = \frac{k-1}{\tau_P} \cdot dt . \tag{1.186}$$

With $n(0)$ denoting the initial number of neutrons, the solution of (1.186) is

$$n(t) = n(0) \cdot \exp(t/\tau) , \tag{1.187}$$

where

$$T = \frac{\tau_P}{k-1} \tag{1.188}$$

is the so-called reactor period.

For a water-moderated reactor system, $\tau_P \approx 10^{-4}$ s shall apply. With $k = 1.001$, the reactor period, T, attains the value 0.1 s. Thus, the number of neutrons increases immensely, e.g. in one second by a factor of

$$n(1\,\mathrm{s})/n(0) \approx \mathrm{e}^{10} \approx 2.2 \times 10^4 . \tag{1.189}$$

Controlling a technical system with these features would be impossible. Fortunately, however, not all neutrons in nuclear fission emerge promptly, but there is a fraction of delayed neutrons, too. This leads to a much longer reactor period and allows controlling with maintainable technical operating expense.

Some nuclear fission products decay with delay under the emission of neutrons. For U-235, e.g., six groups of delayed neutrons are known, as shown in Table 1.14.

Table 1.14. Groups of delayed neutrons for fission of U-235.

Group	$T_{1/2}$ [s]	Average lifetime, $\overline{T}_i$ [s]	Decay constant, λ_i [s^{-1}]	Proportion related to fission neutrons, β_i [%]
1	55.7	80.2	0.0124	0.0215
2	22.7	32.7	0.0305	0.1424
3	6.2	8.9	0.111	0.1274
4	2.3	3.3	0.301	0.2568
5	0.61	0.88	1.14	0.0748
6	0.23	0.33	3.01	0.0273

The total proportion of delayed neutrons in the neutron balance,

$$\beta = \sum_{i=1}^{6} \beta_i , \tag{1.190}$$

is 0.6502 % for U-235. For the other fissile materials this value is different, e.g. $\beta = 0.2$ % for Pu-239. The fraction $1-\beta$ emerges as prompt neutrons from the fission process. The average lifetime of a delayed neutron in group i shall now be τ_i. Hence, the average lifetime of all delayed neutrons can be estimated to

$$\tau_{Del} = \frac{1}{\beta} \cdot \sum_{i=1}^{6} \beta_i \cdot \tau_i , \tag{1.191}$$

with a value of about 13 s for U-235.

The average lifetime of all neutrons in the reactor can be composed of the proportions of both prompt and delayed neutrons, according to the relation

$$\tau_{tot} = (1-\beta) \cdot \tau_P + \beta \cdot \tau_{Del} = (1-\beta)\,\tau_P + \sum_{i=1}^{6} \tau_i \cdot \beta_i . \tag{1.192}$$

Simplified it could be assumed that the average lifetime of all neutrons is actually determined by the delayed neutrons only, since

$$(1-\beta) \cdot \tau_P << \sum_{i=1}^{6} \beta_i \cdot \tau_i . \tag{1.193}$$

Thus, for the total lifetime the following relation holds approximately:

$$\tau_{tot} \approx \beta \cdot \tau_{Del} . \tag{1.194}$$

With $\beta = 0.0065$ and $\tau_{Del} = 15$ s the characteristic time, τ_{tot}, is around 0.1 s. A change in reactivity with $k = 1.001$ leads to a reactor period,

$$T = \tau_{tot} / (k-1), \tag{1.195}$$

of $T \approx 100$ s, so that the change in the number of neutrons in one second is calculated to be

$$n(1s)/n(0) \approx e^{0.01} \approx 1.01. \tag{1.196}$$

Such small changes within long reactor periods can be controlled and carried out quite easily. It must be pointed out once more that the neutron action in the reactor is mainly determined by the delayed neutrons. Therefore a reactor should never be brought to the state of prompt criticality in accidents, as the power could rise extremely in short time.

For an infinitely extended multiplying medium, the change in the number of neutrons can be described by the balance equation,

$$\frac{dn}{dt} = k_\infty \cdot (1-\beta) \cdot \phi \cdot \Sigma_a + \sum_{i=1}^{6} \lambda_i \cdot C_i - \phi \cdot \Sigma_a. \tag{1.197}$$

The first term describes the source of those prompt neutrons that emerge by fission, while the second term refers to the neutron production through nuclear decay of isotopes which send out delayed neutrons. The third term describes neutron losses through absorption.

For the balancing of delayed neutrons, six balance equations can be applied for the alterations of nuclear number density, C_i, of nuclear isotopes which are carriers of delayed neutrons:

$$\frac{dC_i}{dt} = k_\infty \cdot \beta_i \cdot n \cdot \upsilon \cdot \Sigma_a - \lambda_i \cdot C_i, \tag{1.198}$$

with υ as velocity of thermal neutrons and λ_i as decay constants for the six groups of delayed neutrons. The first part on the right-hand side again refers to the emerging from fission, the second term describes the nuclear decay of the mother isotopes emitting delayed neutrons.
With the relations

$$\phi = n \cdot \upsilon, \qquad \rho = \frac{k_\infty - 1}{k_\infty}, \qquad \tau_P = \frac{1}{\Sigma_a \cdot \upsilon} \tag{1.199}$$

and

$$k_\infty \cdot \beta_i \cdot n \cdot \upsilon \cdot \Sigma_a = \frac{k_\infty \cdot \beta_i \cdot n}{\tau_p}, \tag{1.200}$$

one gains the kinetic equations of the following form:

$$\frac{dn}{dt} = k_\infty \cdot \frac{\rho - \beta}{\tau_P} \cdot n + \sum_{i=1}^{6} \lambda_i C_i, \tag{1.201}$$

$$\frac{dC_i}{dt} = k_\infty \cdot \frac{\beta_i}{\tau_P} \cdot n - \lambda_i \cdot C_i \qquad (i = 1 \ldots 6), \tag{1.202}$$

which can be solved if the starting conditions, $n(0) = n_0$ and $C_i(0) = C_i^0$, as well as the change of reactivity, $\rho(t)$, are given. Naturally in practical applications the spatial distributions of neutron flux and isotope concentrations, reactivity feedbacks and temperature distributions have to be included in extended computer programs. The system consisting of seven coupled differential equations is solved by the functions

$$n(t) = A \cdot e^{s \cdot t}, \qquad C_i(t) = C_i \cdot e^{s \cdot t}. \tag{1.203}$$

Insertion into the system delivers:

$$\rho_0 = \frac{s \cdot \tau_P}{s \cdot \tau_P + 1} + \frac{1}{s \cdot \tau_P + 1} \cdot \sum_{i=1}^{6} \frac{s \cdot \beta_i}{s + \lambda_i} = \rho(s) \,. \tag{1.204}$$

This equation determines the decay constants s for a given reactivity ρ. Figure 1.34 gives a graphical solution of the equation, which is also known as "inhour equation". The reason for this nomination is that in the beginning of nuclear technology the reactivity was often expressed as "inverse hours". This was defined as the amount of reactivity which would be necessary to realize a reactor period of 1 hour. Often the unit of reactivity changes is 1 \$ corresponding to the value of β ($\hat{=}$ 0.65 %), 1 ct corresponds to 6.5×10^{-3} %.

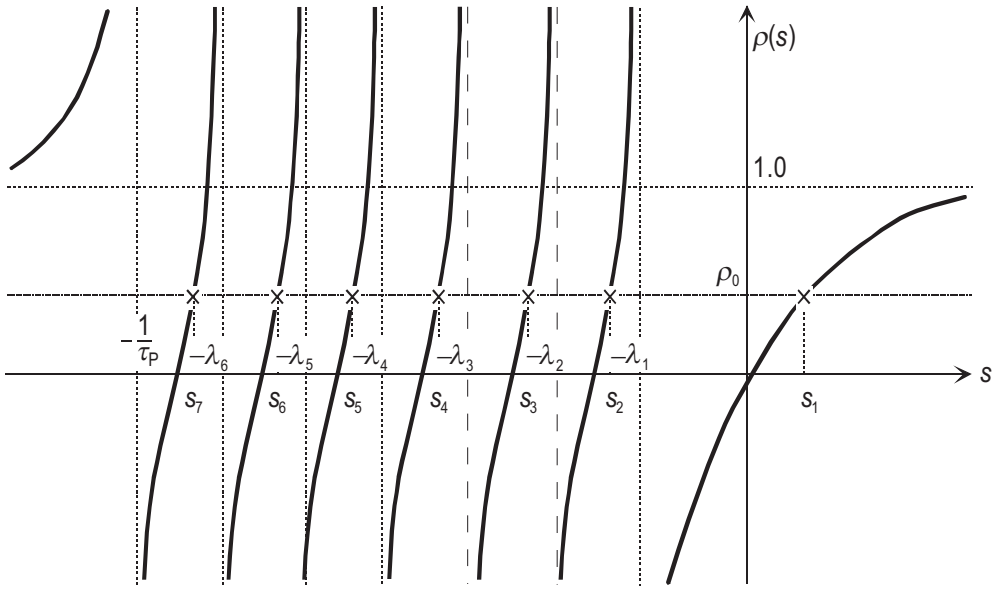


Fig. 1.34. Graphical determination of the routes s_i of the inhour equation.

For a given change of reactivity, ρ_0, the routes are s_1 until s_7. The total solution is

$$n(t) = \sum_{j=1}^{7} n_j \cdot e^{s_i \cdot t} \,. \tag{1.205}$$

The route on the right side of Fig. 1.34 defines the reactor period,

$$T = \frac{1}{s_1} \,. \tag{1.206}$$

Only this route is positive, the other routes are negative, they correspond to transient parts of the solution which would decay fast if a reactivity ρ_0 was added to the reactor core. For the value of ρ_0 the limits

$$-\infty < \rho = \frac{k-1}{k} < 1 \tag{1.207}$$

are valid. The following cases can be defined for transients:

$\rho_0 = 0,$	$s_1 = 0,$	critical reactor,
$\rho_0 = 1,$	$s_1 \to \infty,$	supercritical reactor,
$\rho_0 \to -\infty,$	$s_1 = -\lambda_1,$	subcritical reactor.

The last case is interesting because it shows that a nuclear reactor cannot be shut down faster than with a period $T = 1/\lambda_1$, independent of the negative change of reactivity.

If small changes in reactivity are considered, as it is realized in normal operation ($\rho_0 << \beta$), one obtains for the reactor period:

$$T = \frac{1}{s_1} \approx \frac{1}{\rho_0}\left(l + \sum_{i=1}^{6} \beta_i / \lambda_i\right) \approx \frac{\bar{l}}{\rho_0}, \tag{1.208}$$

with

$$\bar{l} = (1-\beta)\cdot\tau_P + \sum_{i=1}^{6} \beta_i / \lambda_i \ . \tag{1.209}$$

Figure 1.35 shows the reactor period as a function of the reactivity for different values of the lifetime of the prompt neutrons.

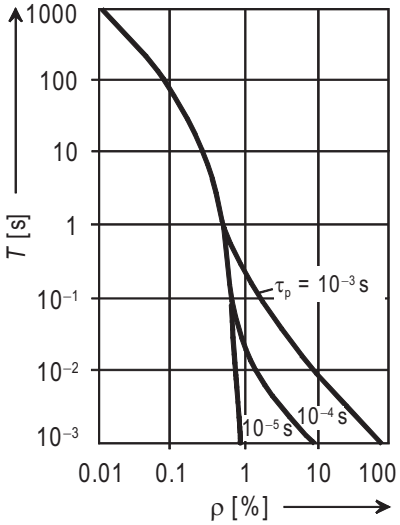


Fig. 1.35. Reactor period, T, as a function of the added reactivity, ρ, for different lifetimes of the prompt neutrons.

For thermal reactors, e.g., with lifetimes of $\tau_p = 10^{-4}$ s and reactivity changes of 1 ‰, reactor periods are typically around 80 s. This allows the safe operation of technical control systems.

Very large positive changes of reactivity ($\rho > \beta$) can be analyzed as follows; $s_1 >> \lambda_i$ is valid, therefore (with $\tau_p = l/e$):

$$\rho_0 \approx \frac{s_1}{s_1 + 1/l} + \frac{1/l}{s_1 + 1/l}\cdot\sum_{i=1}^{6}\beta_i = \frac{s_1 + \beta/l}{s_1 + 1/l}. \tag{1.210}$$

For the reactor period, one finds:

$$T = \frac{1}{s_1} \approx \frac{1}{k\cdot(\rho_0 - \beta)} \approx \frac{l}{k-1}. \tag{1.211}$$

This is the well-known result if the influence of the delayed neutrons is completely neglected. For very large positive changes of reactivity the kinetic behavior of a reactor is destined just by the prompt neutrons. Normally, especially in LWR plants, this condition of prompt criticality must be avoided under all circumstances to prevent destruction of fuel elements and of the reactor core.

1.1.7.6 Dynamic equations

To analyze the dynamic behavior of the total system of a nuclear power plant, all equations that describe the transport of energy from the fission process to the electrical grid have to be formulated and solved. Changes inside the core have consequences for the production of electricity and the delivery to the grid and, vice versa, there are changes inside the core caused by changes initiated from the grid. Changes during operation as start and shutdown, part load or accidents like fast shutdown, failure of pumps, loss of house load, failures on components of cooling systems or auxiliary systems are analyzed to gain knowledge on loads of components and systems.

In a very simplified description some dynamic equations are given as follows – for the neutron behavior the kinetic equations are valid:

$$\frac{1}{\upsilon}\cdot\frac{\partial\phi}{\partial t}=\frac{\rho-\beta}{\tau_{\mathrm{P}}}\cdot\frac{\phi}{\upsilon}+\sum_{i=1}^{6}\lambda_i\cdot C_i\,, \tag{1.212}$$

$$\frac{\mathrm{d}C_i}{\mathrm{d}t}=\frac{\beta_i}{\tau_{\mathrm{P}}}\cdot\frac{\phi}{\upsilon}-\lambda_i\cdot C_i\quad(i=1\ldots6)\,. \tag{1.213}$$

The thermal behavior of the fuel can be approximated by the time-dependent differential equation for the fuel temperatures,

$$\rho_{\mathrm{f}}\cdot c_{\mathrm{f}}\cdot\frac{\partial T_{\mathrm{f}}}{\partial t}=\dot{q}'''_{\mathrm{f}}(\vec{r},t)+\lambda_{\mathrm{f}}\cdot\Delta T_{\mathrm{f}}\,. \tag{1.214}$$

The index f relates to the fuel, $\dot{q}'''_{\mathrm{f}}$ is the power density in the fuel zone, Δ denotes the Laplacian operator. From the fuel the heat is transferred to the coolant according to the expression

$$\dot{q}''(z,t)=\alpha(z,t)\cdot\left[T_{\mathrm{f}}(r,z,t)-T_{\mathrm{c}}(z,t)\right], \tag{1.215}$$

with $\dot{q}''$ as the heat flux on the surface of the fuel canning, α as the heat transfer number, and T_{c} as the coolant temperature. For the power density in the core, $\dot{q}'''_{\mathrm{C}}$, the relation

$$\dot{q}'''_{\mathrm{C}}=\bar{E}_{\mathrm{f}}\cdot\Sigma_{\mathrm{f}}\cdot\phi(\vec{r},t) \tag{1.216}$$

is valid. For reactors with rod-type fuel elements there is a simple connection between $\dot{q}'''_{\mathrm{C}}$, $\dot{q}'''_{\mathrm{f}}$ and $\dot{q}''$:

$$\dot{q}'''_{\mathrm{f}}\cdot\pi R^2\approx\dot{q}''\cdot2\pi R,\qquad\dot{q}'''_{\mathrm{f}}\cdot V_{\mathrm{f}}=\dot{q}'''_{\mathrm{C}}\cdot(V_{\mathrm{f}}+V_{\mathrm{M}})\,, \tag{1.217}$$

with R as the radius of fuel rods, V_{f} as the fuel volume and V_{M} as the moderator volume. To simplify the explanation the heat transport from the fuel surface through the gap to the canning is left out here. It is easy to take into account the gap and the canning by another simple equation.

The heat transport in the coolant can be approximately described by

$$\rho_{\mathrm{c}}\cdot c_{\mathrm{c}}\cdot\frac{\partial T_c}{\partial t}=\dot{q}'''_{\mathrm{c}}(z,t)+\frac{2\pi R}{A_{\mathrm{channel}}}\cdot q''(z,t)-\bar{\upsilon}(z,t)\cdot\rho_{\mathrm{c}}\cdot c_{\mathrm{c}}\cdot\frac{\partial T_{\mathrm{c}}}{\partial z}\,. \tag{1.218}$$

In this equation, $\dot{q}'''_{\mathrm{c}}$ is the volumetric heat production in the coolant itself, A_{channel} is the coolant channel area for each rod, $\bar{\upsilon}$ represents the average velocity of coolant in the channel. In addition to this system of differential equations there is a relation for the changes of reactivity,

$$\rho(t)=\rho_{\mathrm{f}}+\rho_{\mathrm{M}}+\rho_{\mathrm{CS}}=\Gamma_{\mathrm{f}}\cdot\Delta T_{\mathrm{f}}+\Gamma_{\mathrm{M}}\cdot\Delta T_{\mathrm{M}}+\rho_{\mathrm{CS}}\,, \tag{1.219}$$

with the indices f for fuel, M for moderator and CS for the control system. These reactivity changes can be correlated with temperature changes using temperature coefficients. This function of reactivity changes can depend on time in a complex way in case of procedures of normal operation and of accidents.

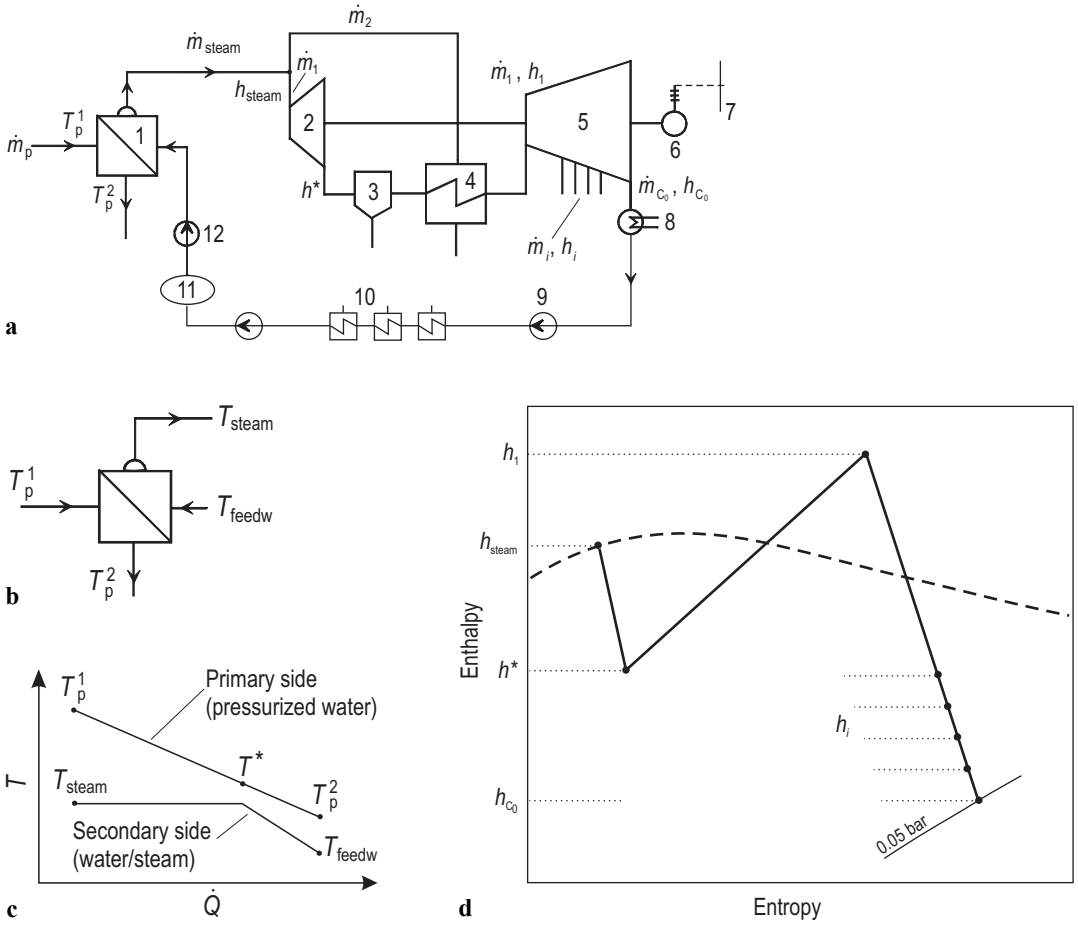


Fig. 1.36. **(a)** Balance of the steam turbine process (example PWR), 1: steam generator, 2: high-pressure turbine, 3: water separator, 4: intermediate reheater, 5: low-pressure turbine, 6: generator, 7: electrical grid, 8: condenser, 9: condensate pump, 10: preheating system, 11: feed water storage tank, 12: feed water pump. **(b)** Heat transfer in the steam generator (example PWR), principle of component. **(c)** T-$\dot{Q}$ diagram for the steam generator. **(d)** Enthalpy-entropy diagram (qualitative) for the expansion process in the steam turbine.

The primary circuit in most types of reactors contains a steam generator. The primary and the secondary side of this component have to be balanced (see Fig. 1.36),

$$\dot{Q}_{\mathrm{SG}} = P_{\mathrm{th}} \cdot \eta_{\mathrm{SG}} = \dot{m}_{\mathrm{p}} \cdot c\left(T_{\mathrm{p}}^{1} - T_{\mathrm{p}}^{2}\right) \cdot \eta_{\mathrm{SG}} = \dot{m}_{\mathrm{sec}} \cdot \left(h_{\mathrm{steam}} - h_{\mathrm{feedw.}}\right) = \sum_{i} k_i \cdot A_i \cdot \Delta T_{\log_i} , \qquad (1.220)$$

where $\dot{Q}_{\mathrm{SG}}$ is the heat load taken by the steam generator, η_{SG} is the efficiency of the steam generator (which normally has a value near 1 in nuclear reactors), and h_i are the relevant enthalpies of the steam turbine process, as indicated in Fig. 1.36. They can be taken from the enthalpy-entropy diagram for the steam expansion in the turbine (Fig. 1.36d).

The mass flow of steam leaving the steam generator is partly used to reheat the steam for the low-pressure turbine. The overall heat-transfer numbers for the steam generator are denoted by k_i (for the economizer and the evaporator section), A_i are the corresponding heat exchanger surfaces, and $\Delta T_{\log_i}$ are the logarithmic temperature differences for the heat transfer in both sections,

$$\Delta T_{\log_1} = \frac{(T_\mathrm{p}^1 - T_\mathrm{steam}) - (T^* - T_\mathrm{steam})}{\ln\left((T_\mathrm{p}^1 - T_\mathrm{steam})/(T^* - T_\mathrm{steam})\right)}, \tag{1.221}$$

$$\Delta T_{\log_2} = \frac{(T^* - T_\mathrm{steam}) - (T_\mathrm{p}^2 - T_\mathrm{steam})}{\ln\left((T_\mathrm{p}^2 - T_\mathrm{steam})/(T^* - T_\mathrm{steam})\right)}. \tag{1.222}$$

All parameters like mass flows, temperatures, powers, and the heat-transfer numbers are time-dependent in case of transients. Analyzing dynamic processes, the total differential of the equations given above has to be used. The steam turbine and the total secondary side of the plant have to be balanced, too, corresponding to Fig. 1.36:

$$P_\mathrm{T} = \eta_\mathrm{th} \cdot P_\mathrm{th} \cdot \eta_\mathrm{SG} = \dot{m}_1 \cdot \left(h_\mathrm{steam} - h^*\right) + \dot{m}_1 \cdot h_1 - \sum_i \dot{m}_i \cdot h_i - \dot{m}_{\mathrm{C}_0} \cdot h_{\mathrm{C}_0}, \tag{1.223}$$

where $P_\mathrm{T}(t)$ is the mechanical power of the steam turbine system, and η_th is the efficiency of the steam cycle.

The values h_i for the extraction of steam from the low-pressure turbine for feeding the preheating section are taken from the exact analysis of the steam turbine process.

Finally the generator is included with the help of the equation

$$P_\mathrm{Gen}(t) = \eta_\mathrm{gen.} \cdot \eta_\mathrm{mech.} \cdot P_\mathrm{T}(t), \tag{1.224}$$

where $\eta_\mathrm{gen.}$ is the efficiency of the generator, and $\eta_\mathrm{mech.}$ is the mechanical efficiency of the steam turbine. A fraction of the electrical power is used as house load (some percent, depending on the reactor type, characterized by $\eta_\mathrm{deliv.}$), so that the net power given to the electrical grid is:

$$P_{\mathrm{el}_\mathrm{net}}(t) = \eta_\mathrm{SG} \cdot \eta_\mathrm{th} \cdot \eta_\mathrm{mech.} \cdot \eta_\mathrm{gen.} \cdot \eta_\mathrm{deliv.} \cdot P_\mathrm{th}(t). \tag{1.225}$$

Thus, changes in the core finally cause changes in the net electricity production, and vice versa.

Naturally, to get a detailed dynamic model of the whole plant, all the other components like pipes, pumps, cooling systems, preheaters have to be included, too. All the equations mentioned above and others which have not been given in detail here form the total system of equations to describe the dynamic behavior of the plant. Complex computer programs are available today for different types of nuclear power plants to analyze all types of dynamic transients and accidents.

As an important example, in Fig. 1.37 there is a short explanation of the power loss of the steam turbine caused by the loss of the electrical grid.

If the generator cannot deliver the electrical energy to the grid, in a very short time the power of the turbine is reduced to the house load. The steam valve at the inlet of the turbine is closed, a valve at the condenser is opened and the steam is condensed in the turbine condenser. If this heat sink fails, the steam can blow off via safety valves. The time-dependent values of mass flow, pressure, coolant temperature and volume in the pressurizer are shown in Fig. 1.37. After around 100 s an equilibrium is established again.

All transients which could occur during power operation and accidents that could happen in the design area are analyzed today using complex computer programs, and the loads on the components of the total plant are calculated to show that the components are strong enough and can fulfill their purpose.

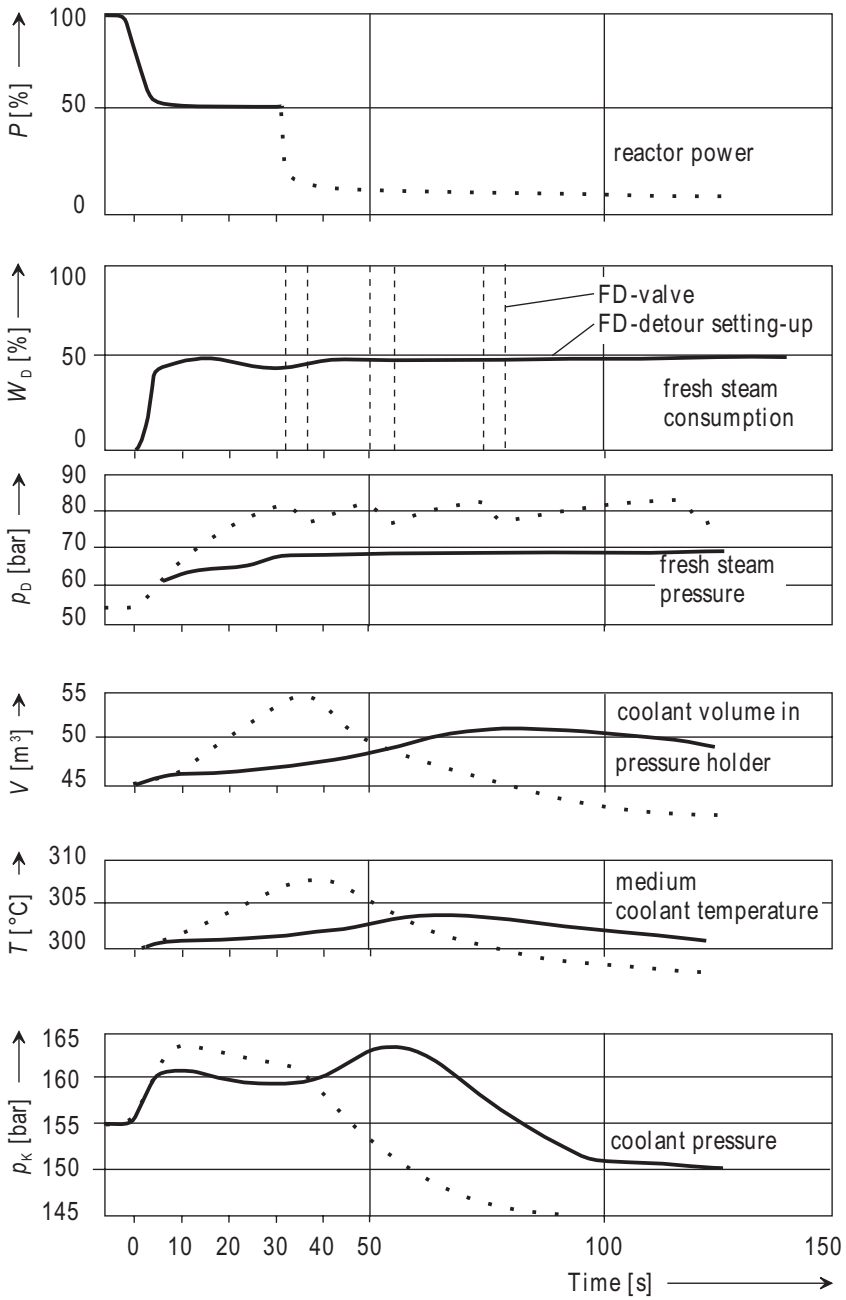


Fig. 1.37. Behavior of a PWR plant in case of loss of the electrical grid (P: reactor power, W_D: mass flow of fresh steam, p_D: pressure of fresh steam, V: volume of coolant in pressurizer, T: average value of coolant temperature, p_K: pressure of primary coolant).

1.1.7.7 On the importance of fission products in nuclear technology

Fission products are of great importance for the operation and safety of nuclear reactors as well as of all steps of waste handling and disposal behind the reactor. With (1.146) a simplified balance equation to calculate the time-dependent inventory of single isotopes in a nuclear reactor was given above. Table 1.13 contains some important fission products with regard to radioactive inventories. The relevant aspects are as follows (see Fig. 1.38):

– The large inventory of fission products in the reactor core – around 10^6 Ci/MW (th) in normal operation – is the source term in case of accidents. There must be a reliable and independent system of barriers to retain the fission products inside the core during all types of accidents.
A similar requirement of fission product retention in the plants holds for all other steps in the nuclear fuel cycle: compact storage of spent fuel elements, intermediate storage of spent fuel elements, direct final storage of fuel elements in geological depositories, reprocessing, production and intermediate storage of glass containers loaded with high-level radioactive waste, and final storage of glass containers. For future plants for partitioning and transmutation the requirement of retention of fission products in the plants will be valid, too.
In case of severe accidents in nuclear reactors, in the first phase fission products like iodine-131 ($T_{1/2} \approx 8$ d) are important for the radioactive burden. Later on, contamination of land becomes very important as a consequence of very severe accidents. Then isotopes like cesium-137 ($T_{1/2} \approx 30$ a) and strontium-90 ($T_{1/2} \approx 29$ a) cause long-term contamination of land. Higher levels of radiation for the population under extreme conditions cause resettlement of people.

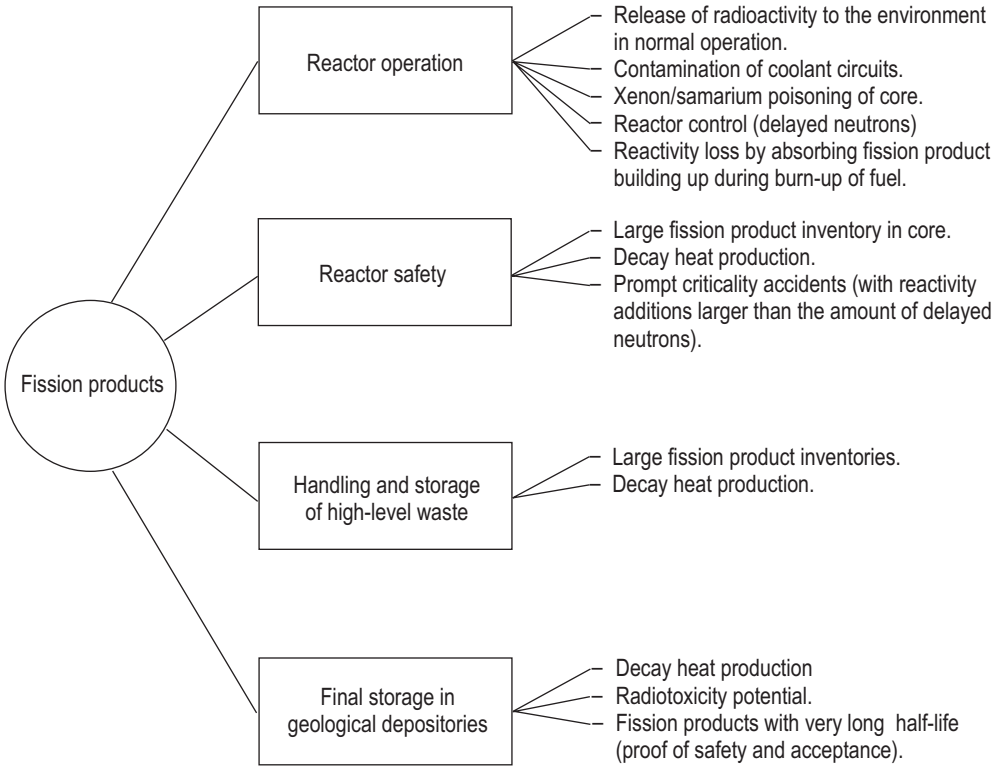
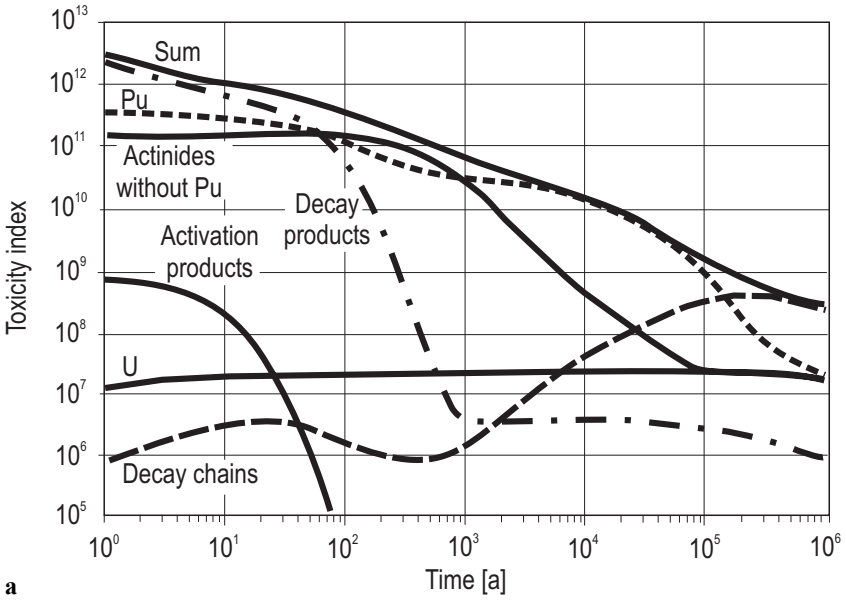


Fig. 1.38. Important aspects of fission products in nuclear technology.

– The decay of fission products produces the decay heat of fuel.
  The save removal of this decay heat is a key question of reactor safety. Additionally this requirement is important in all stages of waste handling and waste storage.
  A total loss of active cooling and of coolant in LWR plants, for instance, would result in a melting of the core and in the release of large amounts of radioactivity to the environment. This kind of accident must be made very improbable by a highly diversified and redundant decay-heat removal system.
  In case of modular high temperature reactors (HTR) with relatively small power, decay heat can be removed from the core just by heat conduction, heat radiation, and natural convection without reaching too high fuel temperatures. Melting of the core is physically excluded.
– Some fission products (Br-87, I-137, Br-88, I-138, …) emit neutrons with very long half-life (the average value being 13 s for U-235). The amount of these delayed neutrons is just 0.65 % in U-235 fuel, but the average lifetime of these neutrons is extremely long compared to the prompt lifetime of neutrons after diffusion (around 10^{-4} s in LWRs).
  The delayed neutrons therefore govern the kinetic behavior of the core in normal operation. Without this fraction of the neutrons the control of reactors would be impossible. In severe accidents, in which the reactivity gain would be larger than the amount of delayed neutrons, prompt criticality could occur, where the behaviour of the rector is just influenced by the prompt neutrons, with very high production of power and destruction of fuel elements. These accidents must be avoided in any case.
  Only in case of coated-particle fuel with specific conditions as in modular HTRs, even such hypothetical prompt excursions could be tolerated, and the core and fuel elements would stay intact.
– Some fission products are very strong neutron absorbers in thermal reactors (xenon-135, samarium-149). The build-up by formation by fission and decay, and the reduction of these isotopes by absorption and decay (xenon) has a strong influence on the dynamic behavior of reactors in case of changes of load as well as after shutdown. Large xenon effects sometimes cause the installation of additional excess reactivity in the core, which has to be compensated for in normal operation. This can initiate disturbances in the reactivity balance with the consequence of reactivity accidents.
– Additionally many other fission products which absorb neutrons are built up in LWRs and other types of reactors with discontinuous loading and discharging of fuel elements. Therefore the reactivity of the core is reduced. To maintain criticality during the operation time, large excess reactivity is installed in the core. Poisoning the cooling water with variable boron concentration is necessary for PWRs. The boron content is reduced during proceeding burn-up. Disturbances of this boronation can cause severe reactivity accidents, too.
  In HTRs with pebble-bed fuel a continuous loading and discharging of fuel elements is realized, therefore no excess reactivity for burn-up is necessary and no accidents can be caused due to this reason.
– Radioactive waste contains some radioactive isotopes with very long half-lives (see Table 1.15). The resulting activity is very low; these isotopes, however, determine the radiotoxicity (see Fig. 1.39) in the final geological depository after very long times (10^6 years storage time and longer). Their contribution to the overall fission product content of spent fuel is small (less than 4 kg/t U), but it is very difficult to perform the proof of safety of final storage over such a long period.

Table 1.15. Content of fission products with long half-lives in high-level radioactive waste (PWR, burn-up: 40000 MWd/t U), Cs-137 and Sr-90 being included for comparison.

Isotope	Half-life [a]	Activity [Ci/t U]
Zr-93	1500000	2.4
Tc-99	210000	17
I-129	17000000	0.04
Cs-135	3000000	2.4
Cs-137	30	125000
Sr-90	28	90000



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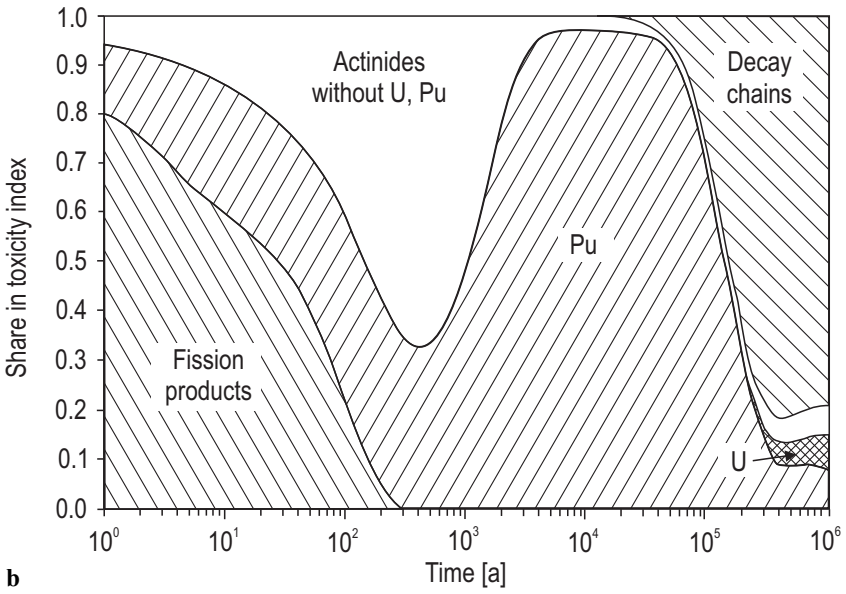

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Fig. 1.39. Radiotoxicity of spent fuel (1 t uranium, burn-up 40 000 MWd/t, PWR fuel): **(a)** toxicity index (ingestion), **(b)** share of different isotopes in toxicity index.

1.2 Nuclear power plants

1.2.1 Overview of different reactor types

The core of a nuclear reactor contains the fuel elements, the moderator, the coolant and structural materials. Additionally control and shutdown elements are installed. The heat is generated by fission processes inside the fuel elements. Moderators in thermal systems act to slow down the neutrons from fission energies to thermal energies. This is advantageous because the cross sections for fission are much higher in the thermal region of the neutron spectrum than at very high energies. Suitable moderator substances with technical importance today are H_2O, D_2O and graphite (Fig. 1.40). In case of fast reactors a moderator is nearly totally avoided. The reason is that η, the number of neutrons produced per fission, is higher at very high energies compared to thermal energies.

The coolant transports the heat from the core to the steam generator or directly to a turbine. Important coolants are H_2O, D_2O, CO_2 and helium; liquid metals (Na, Pb) are suited for fast reactors (Fig. 1.40). From the beginning in nuclear technology many reactor concepts have been developed, but only a few remained which where technically feasible, economically attractive and interesting for future application. Table 1.16 contains these concepts and some of their main aspects.

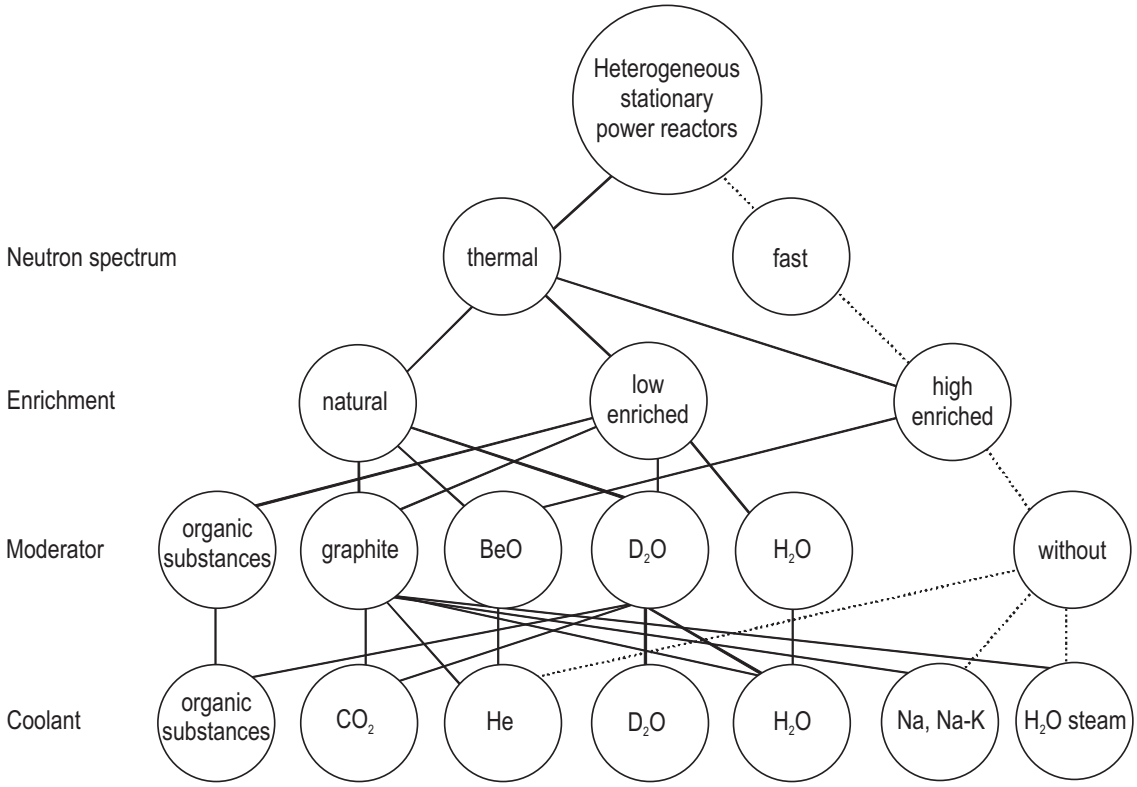


Fig. 1.40. Overview of possible combinations of reactor concepts.

Table 1.16. Important reactor concepts and some of their main aspects.

	PWR	BWR	RBMK	CANDU	AGR	HTR	LMFR
Moderator	H_2O	H_2O	H_2O/C	D_2O	C	C	—
Neutron spectrum	thermal	thermal	thermal	thermal	thermal	thermal	fast
Fuel	UO_2, PuO_2	UO_2, PuO_2	UO_2, PuO_2	UO_2, PuO_2	UO_2	UO_2, PuO_2	UO_2, PuO_2
Shape of fuel elements	rods	rods	rods	rods	rods	pebbles, blocks	rods
Coolant	H_2O	H_2O	H_2O	D_2O	CO_2	He	Na
State of coolant	liquid	liquid/ steam	liquid/ steam	liquid	gas	gas	liquid
Special aspects	Zircaloy canning	Zircaloy canning	pressure tubes (Zr)	pressure tubes (Zr)	steel canning	coated-particle fuel	breeder

Today the main interest focuses on the following systems:

- Pressurized water reactors (PWR) [74Old, 83Zie, 92Kni, 83Kes]: H_2O in liquid state serves as moderator and coolant. The high pressure of the primary system (160 bar) is coupled to the maximal temperature of the coolant, because the coolant must stay in liquid state. This system just produces saturated steam, which allows a net efficiency of 33 %. The steam generator allows for a strong separation between the primary and secondary circuit. Worldwide nearly 200 large PWR plants are presently operational.
- Boiling water reactors (BWR) [00GKN, 86Boh, 69Sau, 92Led, 88Ton]: They work with a direct cycle with a steam turbine directly coupled to the reactor. Water is evaporated inside the core, the steam is dried inside the reactor pressure vessel and directly enters the turbine. The steam state and the net efficiency are similar to the corresponding parameters of PWRs. Today around 100 large BWR plants are in operation worldwide.
- RBMK reactors [91Ull, 86IAE]: These reactors use a large number of vertically arranged pressure tubes with fuel rods inside these tubes to form a reactor core. The neutronic coupling between the single tubes is realized by graphite blocks. This technology allows to operate power plants with large thermal power without large reactor pressure vessels. Inside the tubes water is evaporated, similar as in the core of a BWR. An outside steam drum is used to give off saturated steam for the turbine plant. The thermodynamic data are similar to those of LWRs. In the countries of the former USSR 13 plants of this type are still in operation today. After the catastrophic accident in Chernobyl (1986) no further RBMK reactors have been built.
- CANDU reactors [75Mor, 87Can, 77EPR]: This type, mainly realized in Canada, uses heavy water for moderation and cooling. The fuel rods are arranged inside horizontal pressure tubes. The neutronic coupling is realized by heavy water in a calandria tank which contains a large number of pressure tubes. Because of use of D_2O the CANDU reactor can work with natural uranium or low enriched material. The steam parameters are below those of LWRs, therefore the efficiency is lower, too. Altogether 30 CANDU plants are in operation in Canada and were exported to some other countries.
- Advanced gas-cooled reactors (AGR) [92Mod, 80Den]: Carbon dioxide acts as coolant, and graphite is the main structural material in the core. Steel is used as canning material for the fuel rods. Corrosion between CO_2 and graphite restricts the coolant temperature to maximal values of 650 °C, which is sufficiently high to produce hot steam of 530 °C. Therefore the net efficiencies of these power plants are high (40 %). This type of plant is realized and demonstrates successful operation since decades in Great Britain (15 plants). The total primary system is integrated in a prestressed concrete reactor vessel.
- High temperature reactors (HTR) [90AVR, 89Kug, 84Mel, 84KWU]: Helium is the coolant and graphite the main structural material. Fuel and fertile material are used in very finely dispersed form. These coated particles with three very thin layers of graphite/silicon carbide/graphite are very well suited to retain the fission products. Developed are spherical fuel elements and hexagonal blocks. The

coolant temperature is above 700 °C, and this allows the combination to steam cycles, gas turbines or combined processes. The efficiencies are between 40 and 48 %. This type of reactor promises a high degree of inherent safety for future application, too, because the fuel can never melt in loss-of-coolant accidents. The HTR is still under development.

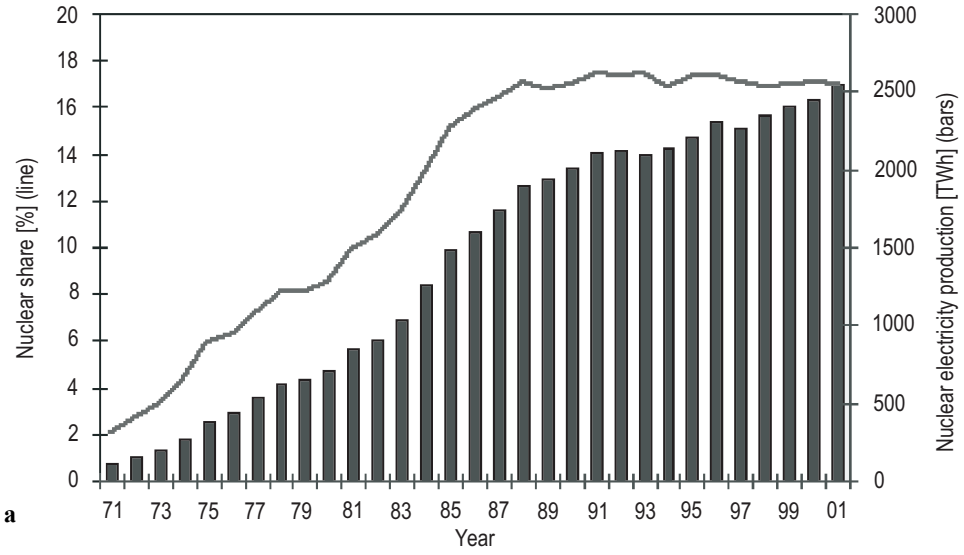
– Liquid-metal fast reactors (LMFR) [72Bed, 81Wal, 76Bra, 71Gra, 95Mic]: The neutron spectrum of these reactors is fast, because moderators are avoided as far as possible. The coolant is sodium or lead. The reactor contains fuel elements with plutonium and fertile elements with depleted uranium. Principally breeding is possible. The system uses an intermediate circuit consisting of sodium for safety reasons, the steam conditions are almost identical to those of conventional power plants. Therefore the efficiency is high (40 %). The technology of fast breeding reactors is still under development as a future option for long-term fuel supply. Breeding would allow much more efficient use of uranium compared to current LWRs, a factor of 40 would be feasible using intensive reprocessing.

The numbers in Table 1.17 characterize some technical conditions of the core and plant layout of the different reactor types for comparison. Being the most important reactor type today – constituting more than 60 % of the nuclear power capacity installed worldwide – the PWR is explained in more detail in Sect. 1.2.2. In Sect. 1.2.6 some additional information is given on the other reactor types mentioned above.

Figure 1.41a shows the development of nuclear electricity production in the world. Today already 17 % of the electricity is produced by nuclear power plants. The shares of the different reactor types are given in Fig. 1.41b. Some countries are already heavily dependent on nuclear energy (see Fig. 1.41c).

Table 1.17. Some important parameters of different reactor types.

		PWR	BWR	RBMK	CANDU	AGR	HTR	LMFR
Typical enrichment	[%]	3...5	3...5	2	< 1.5	2	8	10
Core power density	[MW/m³]	100	50	4	15	2	3	400
Coolant temperature	[°C]	290…325	200…285	200…285	200…305	250…650	250…700 (900)	380…540
Coolant pressure	[bar]	160	≈ 70	70	95	40	60	10
Steam pressure	[bar]	65	70	70	43	180	180	170
Steam temperature	[°C]	280	285	285	255	530	530	500
Efficiency	[%]	33	33	32	32	40	40	40
Typ. thermal power	[MW]	3800	3800	3000	1500	1500	200…600	750
Special aspects					natural uranium possible		gas turbine application (900 °C)	breeding



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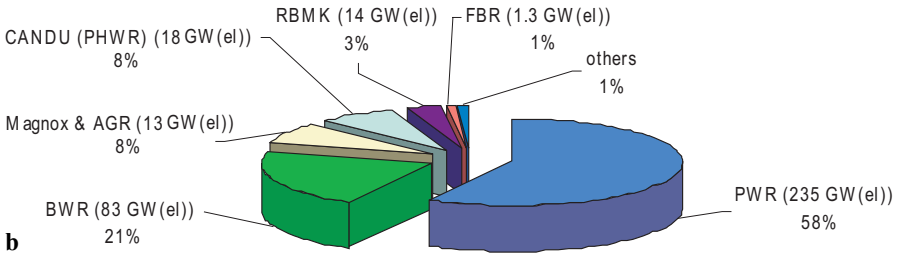

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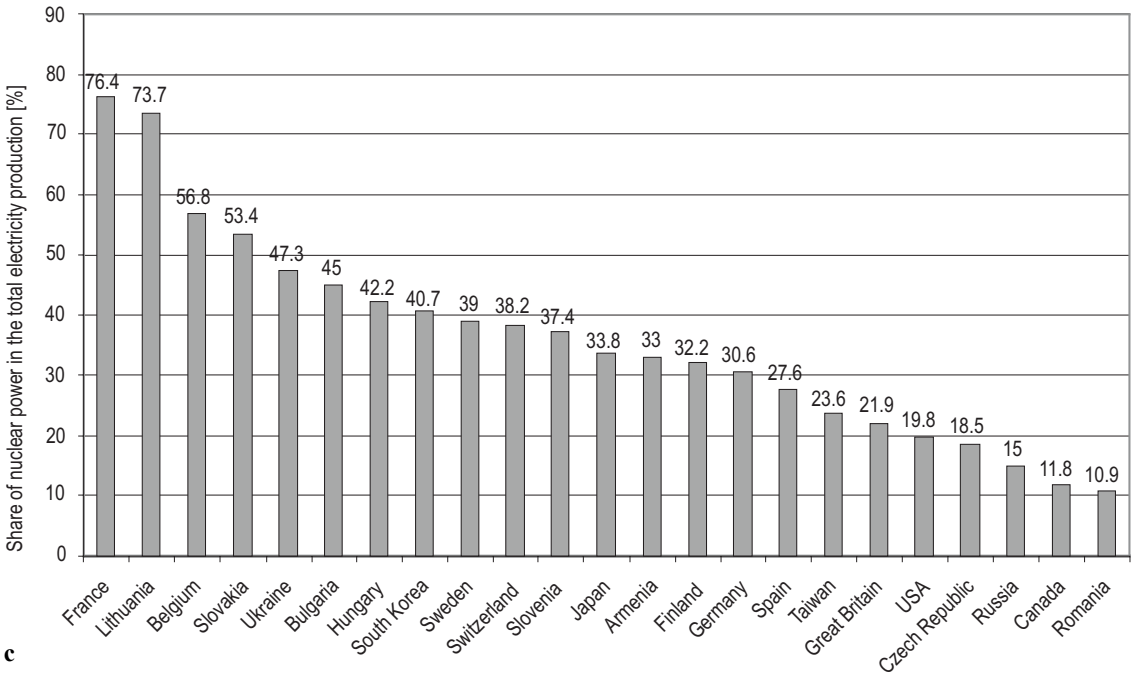

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Fig. 1.41. Development of nuclear energy: **(a)** growth of nuclear electricity production, **(b)** shares of different reactor types (total numbers: 364 GW (el), 435 plants), **(c)** dependence of different countries on nuclear energy.

1.2.2 Pressurized water reactors

1.2.2.1 Plant overview

Pressurized water reactors [74Old, 83Zie, 92Kni, 83Kes] use water as moderator and coolant at a pressure of 160 bar and, inside the core, at a temperature between 291 °C and 328 °C (see Fig. 1.42a). In order to avoid boiling of the cooling medium a pressurizer is connected to the primary circuit. The pressurizer contains an electrical heater to raise the pressure and a spraying system for water in order to reduce the pressure. If the pressure exceeds the allowed values the primary system will be depressurized and the steam will enter the containment building.

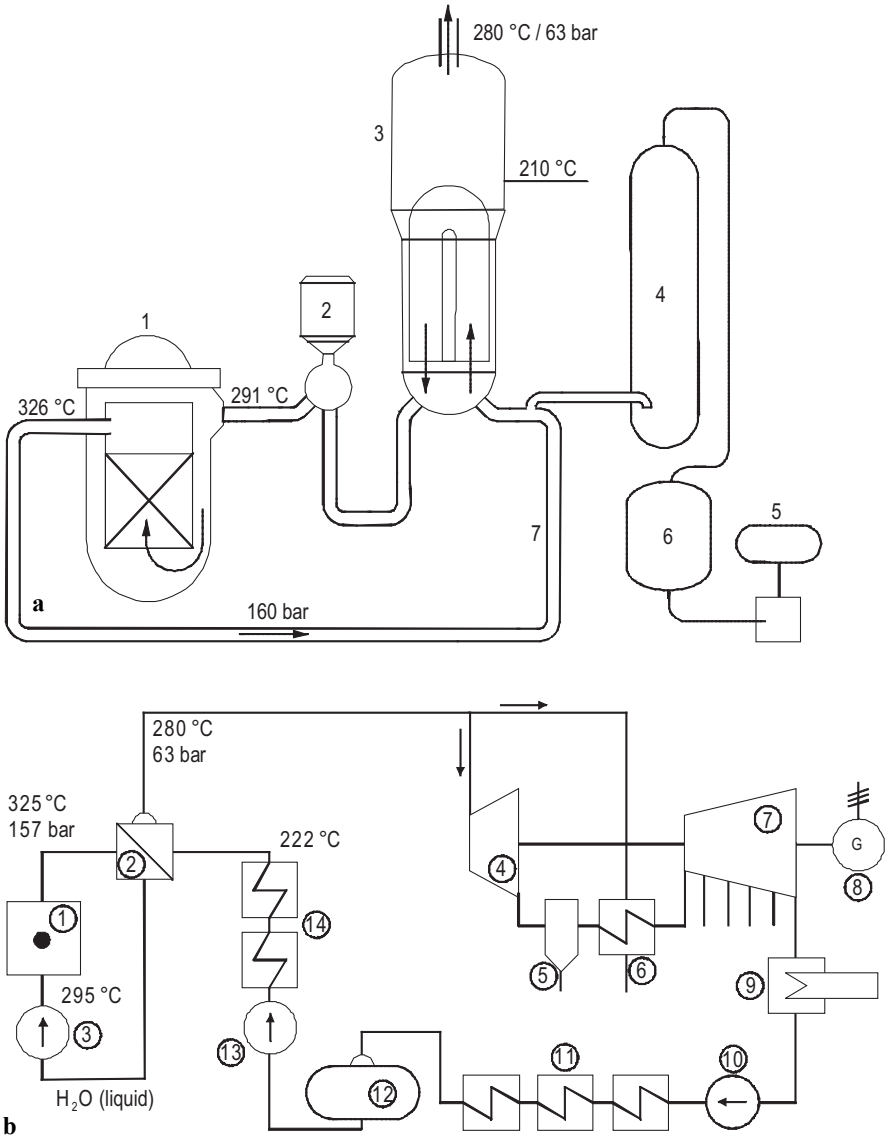
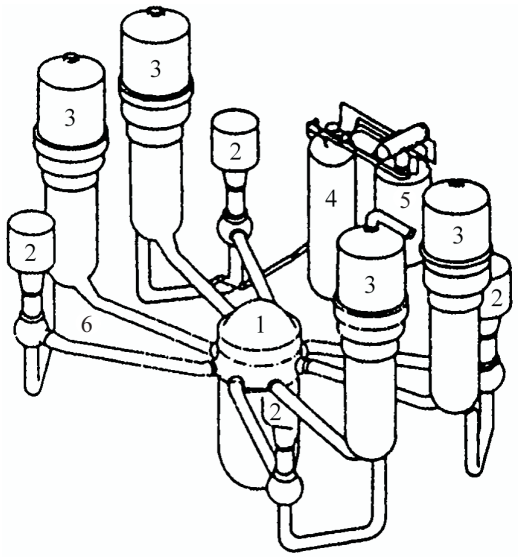


Fig. 1.42. PWR overview: **(a)** schematic flow sheet of the primary system (1: nuclear reactor, 2: pump, 3: steam generator, 4: pressurizer, 5: cooler for deloading system, 6: vessel for deloading, 7: primary pipes for coolant), **(b)** schematic flow sheet of the total plant (1: reactor, 2: steam generator, 3: pump, 4: high-pressure turbine, 5: water separator, 6: reheater, 7: compressure turbine, 8: generator, 9: condenser, 10/13: feed water pump, 11/14: preheater, 12: feed water storage).

In normal operation the hot coolant heats the steam generator, and saturated steam (280 °C, 63 bar) is produced on the secondary side of the steam generator. In a steam cycle with reheat (see Fig. 1.42b) the steam is used to drive a turbine system and to generate electricity. The condenser is cooled with the aid of a wet cooling tower. Several preheaters serve for preheating the working fluid to a temperature of around 200 °C before entering the steam generator. The reheat is made with the aid of a bypass stream of fresh steam behind the steam generator. Todayt the net efficiency of such a cycle is 33 % using wet cooling towers as heat sink.

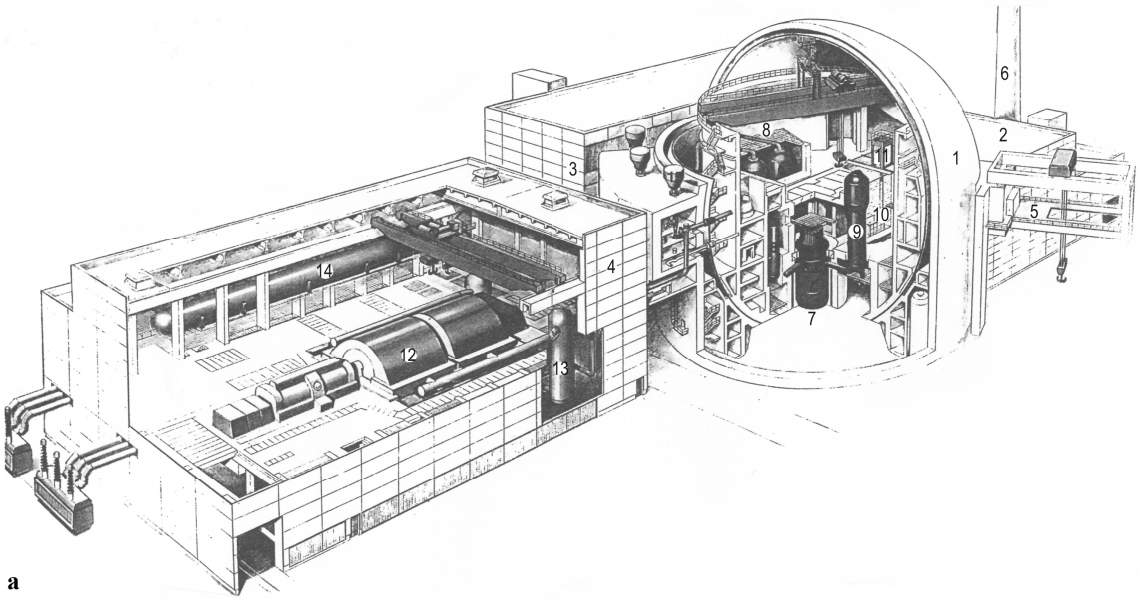
A large modern PWR (1300 MW (el)) contains four loops including steam generator, primary pumps and connecting pipes (see Fig. 1.43). All components of the primary circuit are arranged inside the reactor containment (see Fig.1.44a). This building contains a dense steel shell, which withstands a pressure of around 8 bar in accidents, and in new plants a thick-walled concrete structure of nearly two meters is realized around this steel shell. This wall thickness is designed with respect to airplane crashes (Phantom military airplanes). Details of the design are explained below. Especially in German plants a large compact wet storage system for spent fuel elements is installed inside the containment, too, for intermediate storage of around 3 years.

Additional buildings, which contain e.g. the steam turbine cycle and the turbo machinery including the generator and various auxiliary systems, are further parts of the total plant (see Fig. 1.44) These systems are: the volume control system as a link between the hot, high-pressure reactor coolant system and the reactor auxiliary systems at lower pressure; the chemical control system, which serves for adjusting the coolant chemistry of the reactor cooling water to specified values; the coolant purification system, which is used to remove corrosion products and fission products; different cleaning systems, which ensure that gaseous, liquid and solid radioactive releases are kept below specified limits and that the environment is protected; water supply systems and diesel engines for the purpose of emergency core cooling and residual-heat removal.



Thermal power	3867 MW
Electrical power (gross)	1365 MW
Electrical power (net)	1269 MW
Efficiency (net)	32.7 %
Primary coolant pressure	158 bar
Primary coolant temperature	291.7 °C…325.6 °C
Secondary circuit steam pressure	63.5 bar
Secondary circuit feed-water temp.	218 °C
Secondary circuit steam temp.	279.3 °C
Number of loops	4

Fig. 1.43. PWR: arrangement of primary components in a four-loop plant (1300 MW (el)), 1: reactor, 2: coolant pumps, 3: steam generators, 4: pressurizer, 5: blow-off system, 6: primary coolant pipes. In addition, some technical facts are given.



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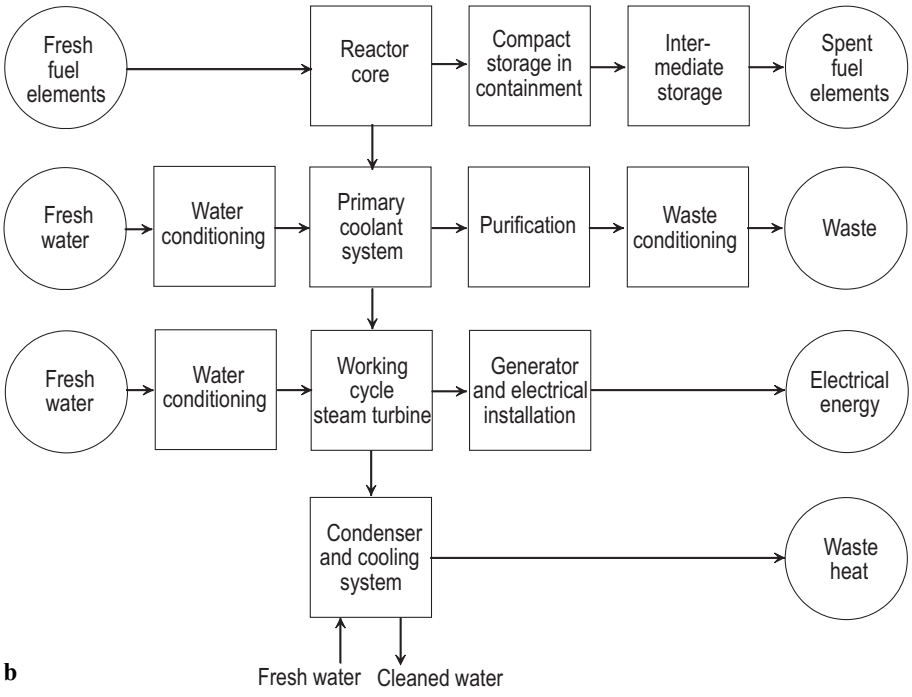


b

Fig. 1.44. PWR: **(a)** arrangement of the total nuclear power plant, 1: reactor containment, 2: reactor auxiliary building, 3: switchgear building, 4: turbine hall, 5: lift frame, 6: stack, 7: reactor, 8: pressurizer, 9: steam generator, 10: compact storage for spent fuel elements, 11: fuel loading machine, 12: steam turbine, 13: water separator/reheator, 14: feed-water storage tank. (Courtesy of Framatome ANP GmbH). **(b)** Process chain.

1.2.2.2 Components of the core

The main components of the reactor core are the fuel elements. In case of the PWR they consist of fuel rods (see Fig. 1.45a). These rods contain the fuel (UO_2, PuO_2) in form of pellets, the canning consists of Zircaloy. Some data of fuel rods and fuel elements are included in Table 1.18. The free space inside the gas-tight canning is filled with helium, during the burn-up in the reactor this space acts as a storage for gaseous fission products which are released from the pellets. Depending on the type of fuel element, 300 of these rods are arranged to form a fuel element (see Fig. 1.45b) This is a fuel element with 18 × 18 positions, some of which are empty and used to guide a finger-type control element. The rods are assembled to form a fuel element with the help of head and foot parts. Spacers between the rods are intended to avoid vibrations. A large number of these fuel elements forms the core of a PWR (193 fuel elements in a 3800 MW (th) core), as shown in Fig. 1.45c. The fuel elements are fixed in the core by an upper and a lower grid plate.

Figure 1.46 includes some further details regarding the arrangement of fuel elements in the core. Different fuel enrichments are employed to achieve a power distribution as flat as possible in the radial direction of the core and to realize an optimal utilization of fuel (Fig. 1.46c). Apart from the upper and lower grid plate, there is an outer core barrel which also serves to guide the coolant.

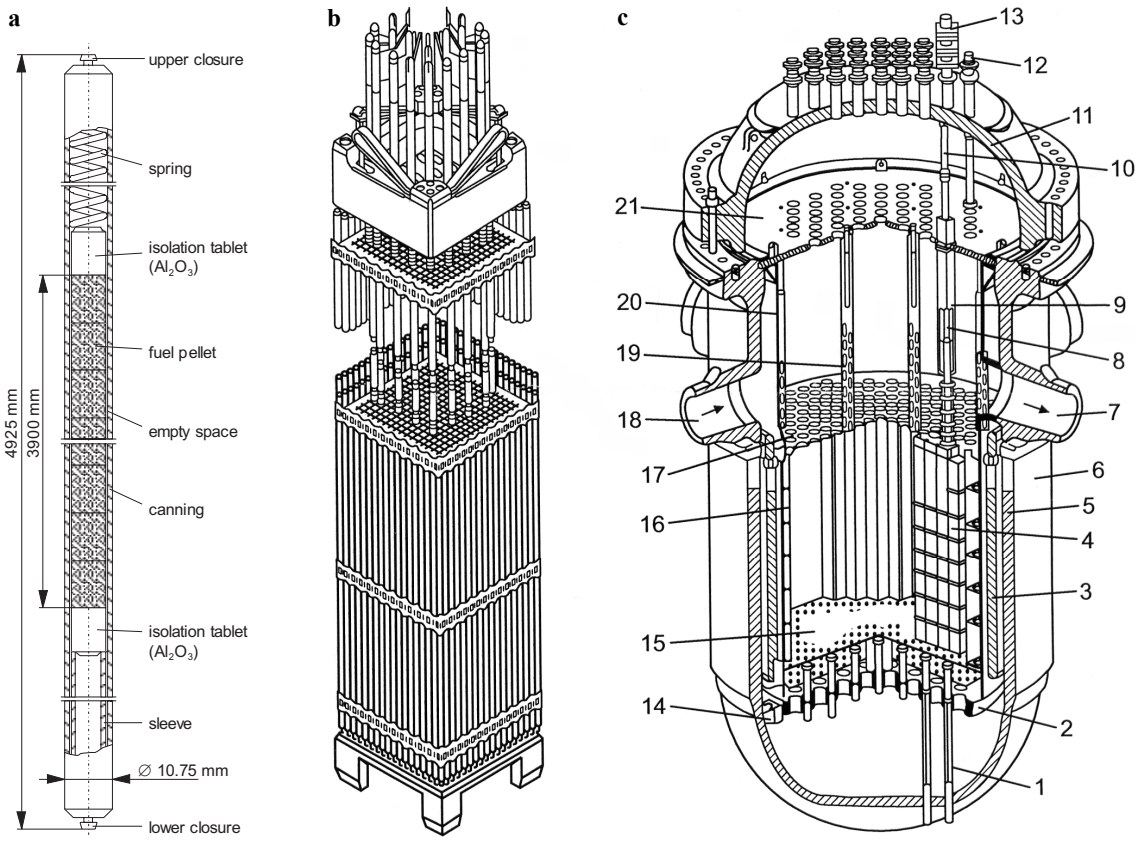
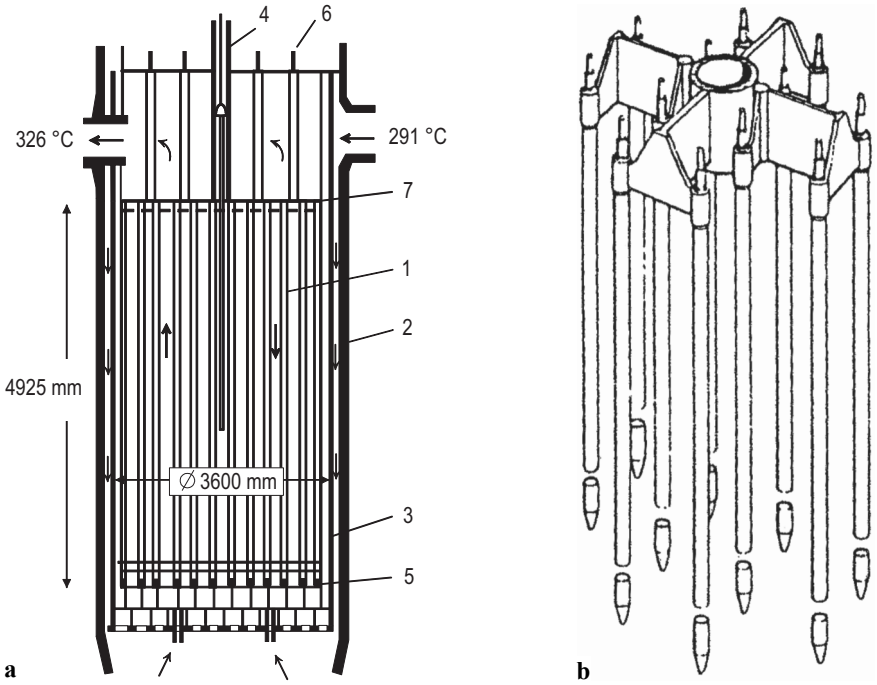


Fig. 1.45. PWR fuel elements: **(a)** fuel rod, **(b)** fuel element, **(c)** arrangement of fuel elements in the core (1: instrumentation guide tube, 2: bottom support structure, 3: outer thermal shield, 4: fuel assemblies, 5: reactor pressure vessel, 6: surface of reactor pressure vessel (insulation), 7: outlet nozzle, 8: control-rod shaft, 9: shroud, 10: connecting shaft, 11: closure head, 12: control-rod penetration, 13: control-rod drive mechanism, 14: radial support, 15: bottom support plate, 16: inner thermal shield, 17: upper core plate, 18: inlet nozzle, 19: ducting tube, 20: shroud, 21: upper support plate). (Courtesy of Framatome ANP GmbH).

Table 1.18. Some important parameters of PWR fuel elements.

Thermal power of the core	3867 MW
Number of fuel elements	193
Geometry of fuel elements (number of positions of absorber fingers)	18 × 18 (24)
Active length of fuel rod	3900 mm
Core diameter	≈ 4000 mm
Rod diameter	9.5 mm
Spacing	12.7 mm
Weight of a fuel element (UO_2)	533.7 kg
Average linear power of rod	166.6 W/cm
Average core power density	95.3 MW/m³
Fuel enrichment	max. 5 %
Average burn-up	45 000 MW d/t



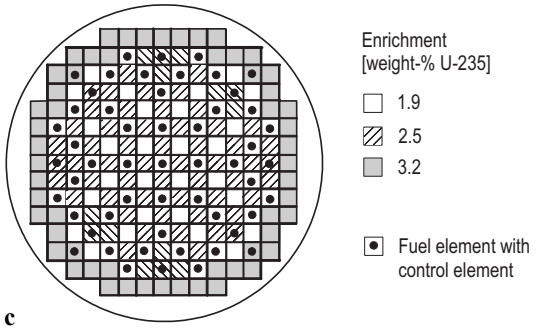


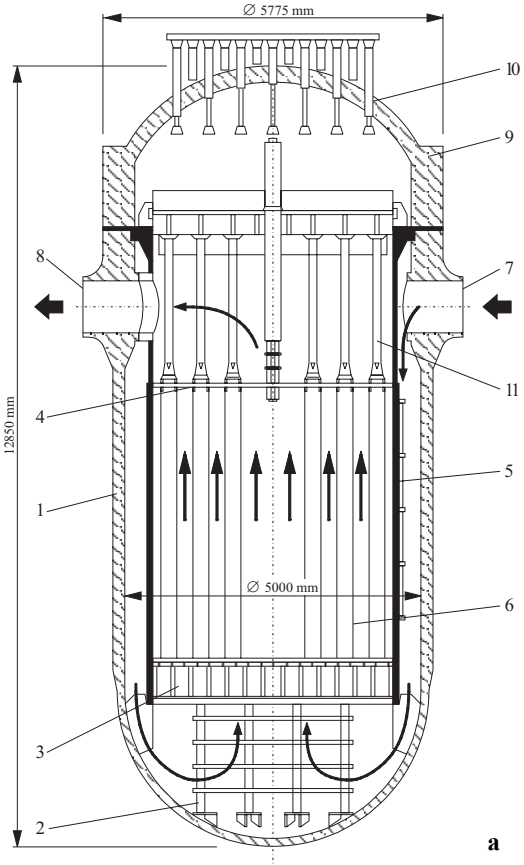
Fig. 1.46. PWR core: **(a)** component arrangement and coolant flowt (1: fuel element, 2: pressure vessel, 3: core-barrel, 4: control-rod drive system, 5: lower grid plate, 6: control-rod system, 7: upper grid plate), **(b)** finger-type control rods, **(c)** arrangement of fuel elements with different enrichment and positions of control elements.

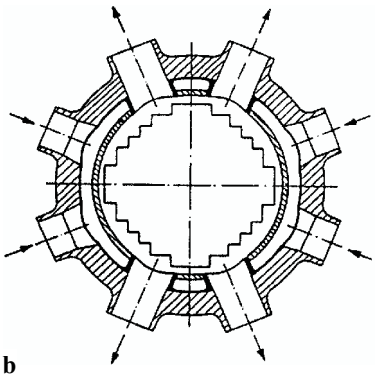
1.2.2.3 Components of the primary system

The reactor core is arranged inside the reactor pressure vessel (RPV). The cylindrical vessel has a bottom head and a closure, which is fixed by bolts and can be removed once a year for changing fuel elements. In the upper part of the cylindrical vessel there are 8 penetrations, 4 each for the inlet and outlet, respectively, of the primary coolant (Fig. 1.47). The RPV is positioned inside the concrete structures of the containment in a central position.

Because of the large diameter and the high operational pressure, a large wall thickness is necessary (see Fig. 1.47c). The RPV is totally plated on the inside by a layer of austenitic steel. This is necessary because of the strong corrosion by boron acid inside the coolant. Modern RPVs are fabricated from rings of forged steel, which are connected with the help of special welding processes.

During operation the vessel is stressed by internal pressure, heat stresses and irradiation with fast neutrons. A special, very ductile steel is used, in which the growth of cracks is practically avoided. This is guaranteed by repeated inspections. Because of the reduction of ductility by irradiation effects, the repeated integral pressure tests are carried out at elevated temperatures. If all prescriptions of design, manufacture, operation and testing are fulfilled, it is assumed that a catastrophic failure of the RPV, i.e. bursting of the vessel, is very unprobable. In safety analyses today there are estimations that the value for bursting should be below 10^{-8}/year. Bursting of the vessel would mean that the core could not be cooled and would melt. Therefore the safety of the RPV is of central importance for reactor safety.





b

Operation temperature	325 °C
Operation pressure	158 bar
Inner diameter	5000 mm
Wall thickness (cyl. wall)	250 mm
Wall thickness of austenitic plating	6 mm
Wall thickness (ring for penetr.)	400 mm
Total weight	≈ 500 t
Material	20MnMoNi55

c

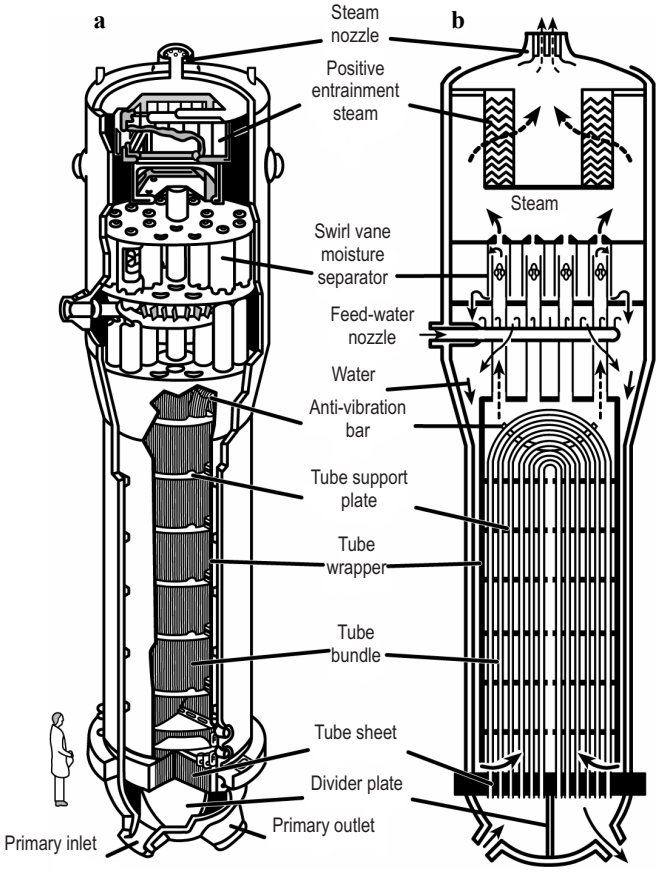
Fig. 1.47. Reactor pressure vessel of a PWR: **(a)** vertical cross section (1: pressure vessel, 2: core support, 3: lower grid plate, 4: upper grid plate, 5: core barrel, 6: fuel element, 7: inlet nozzle, 8: outlet nozzle, 9: reactor vessel head, 10: control-rod drive mechanics, 11: control-rod drive shaft), **(b)** horizontal cross section, **(c)** technical parameters.

For future concepts of nuclear reactors prestressed RPVs are proposed, they are designed to be burst proved. The heat of the fission process is used in the steam generator. In most PWR plants worldwide, steam generators as the one shown in Fig. 1.48 are employed. The primary coolant flows into a chamber at the bottom of the component.

From here the coolant is guided through U-type steam generator tubes, and the heat is transferred to the secondary side. In the bottom part of the steam generators the feed water is preheated up to the boiling point, and in the upper part the formation of saturated steam takes place. Above the tube bundle a separator for water droplets is arranged and the steam is dried to a humidity of less than 1 % water (liquid). Because of the good heat transfer capabilities of pressurized water and water/steam mixtures the heat fluxes are high. Average values of 180 kW/m^2 are realized using relatively low temperature differences between primary and secondary side for the heat transfer.

The U-type tubes of the bundle are fixed in the tube plate at the bottom of the component. The tubes themselves and the plate are fabricated from a corrosion-resistant austenitic material. Spacers between the tube avoid vibrations. The steam generator acts as an important barrier between the contaminated primary circuit and the secondary side of the working fluid. Besides the U-type steam generator described here, in some countries steam generators with straight tubes or with horizontal U-tubes and internal samples are in operation. In the first case the steam temperature can be somewhat higher, in the second case the amount of water on the secondary side is higher. This aspect is sometimes important from the standpoint of reactor safety, to achieve longer gracetimes.

In the PWR a large mass flow of water through the core must be guaranteed to remove the heat generated by the fission processes. Radial pumps with one stage are in use to compensate the total pressure drop of around 9 bar in the primary system and to keep the water circulating.



c

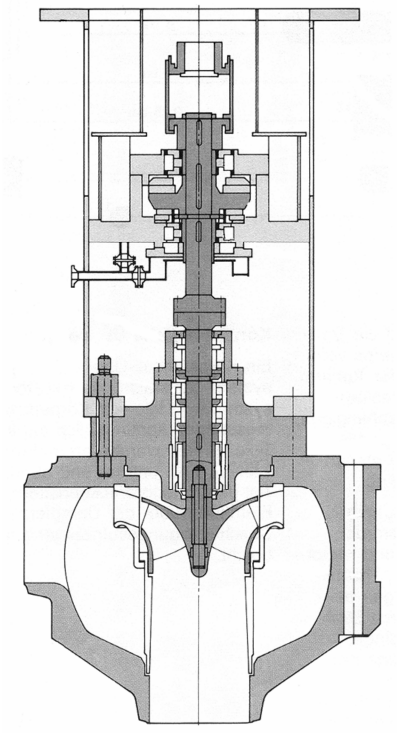
Number of steam generators	4
Power of one steam generator	967 MW
Diameter	3.7 m
Height	21.5 m
Weight	490 t
Heat transfer area	5400 m^2
Number of tubes	4118
Material of tubes	Incoloy 800

Fig. 1.48. PWR steam generator: **(a)** overview, **(b)** vertical cross section, **(c)** parameters. (Courtesy of Framatome ANP GmbH).

A specific technology is used to get the rotating system tight between water of 160 bar and the environment; a blocking system using water is suited for this purpose. It consists of several stages. Figure 1.49 shows a vertical cross section through a coolant pump and lists some technical data. The power of these pumps is large, around 6 MW are used for each loop, corresponding to 2 % of the electrical power that can be produced from the thermal power of the loop.

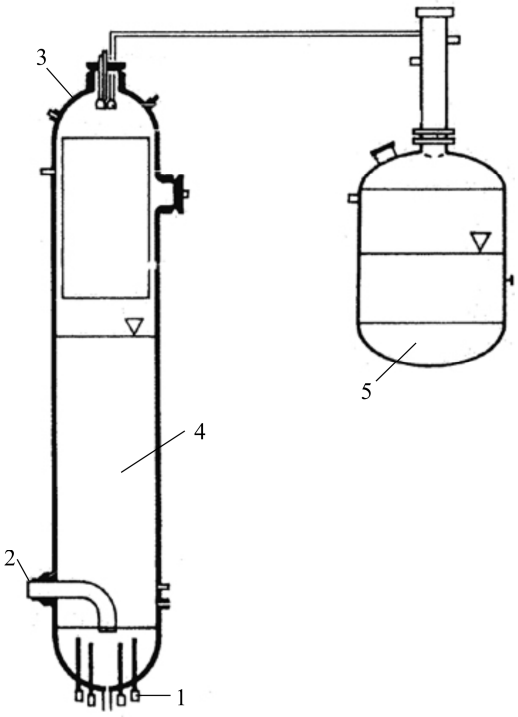
The coolant in the primary circuit must always stay in the liquid state. The primary circuit is therefore connected to a pressurizer. In the lower part of the component there is a water storage, above which there is steam. In the pressurizer an inner electrical heater is arranged to raise the pressure, and a spray system to reduce the pressure. For safety reasons the pressurizer is connected to a blow-off system. If the pressure in the primary system exceeds 179 bar the valve opens and reduces the pressure. Figure 1.50 contains the flow sheet of a pressurizer.

The different components of the primary system are connected by pipes (diameter around 750 mm), which are designed to tolerate thermal expansions, vibrations, transient loads and loads from earthquakes.



Plant	3867 MW with four loops
Number of coolant pumps	4
Mass flow	4700 kg/s
Pressure drop	9 bar
Power (max.)	7.4 MW
Drive	electrical motor

Fig. 1.49. Coolant pump: vertical cross section and technical facts. (Courtesy of Framatome ANP GmbH).



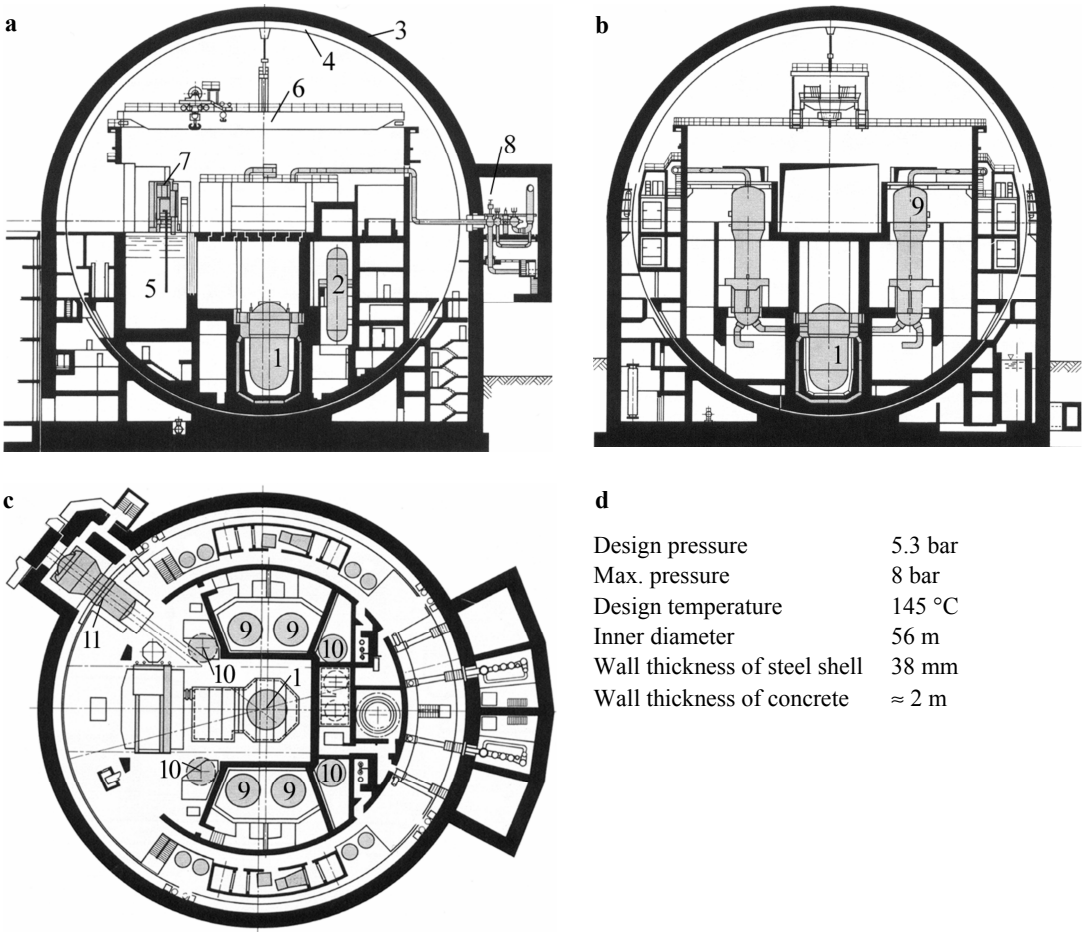
Volume	65 m^3
Power of heater	2.1 MW
Diameter	2.5 m
Height	12 m

Fig. 1.50. Pressurizer and blow-off vessel: flow sheet (1: electrical heater, 2: connection to primary circuit, 3: water injection, 4: water, 5: blow-off vessel) and technical facts. (Courtesy of Framatome ANP GmbH).

1.2.2.4 Reactor containment

All components of the primary circuit are arranged inside the reactor containment. This building (German plants) contains an inner spherical steel shell of around 60 m diameter. This inner shell is tight and is designed to withstand a maximum inner pressure of 6 to 8 bar in accidents. The outside of the containment of modern plants consists of a concrete wall with almost 2 m thickness. This outer shell acts as a protection against airplane crashes, pressure from gas-cloud explosions and against sabotage. New plants are designed to withstand relatively strong earthquakes, too. Details of a PWR containment building are shown in Fig. 1.51.

All penetrations of the containment are leaktight. Lock systems for persons and components are installed and allow to change even complete steam generators. Inside the containment a compact intermediate storage for spent fuel elements is installed. This is a water storage system, in which the spent fuel elements stay for three years. The decay heat is removed by redundant cooling systems, the arrangement of fuel elements is always subcritical and the cannings of the fuel elements act as a tight barrier for the fission products. In the ring-shaped space around the containment there are various installations of the redundant decay-heat removal and emergency core-cooling systems, as well as systems for water purification and a control system for the boration of the primary coolant. Intermediate nuclear cooling systems are arranged in this building, too.



d

Design pressure	5.3 bar
Max. pressure	8 bar
Design temperature	145 °C
Inner diameter	56 m
Wall thickness of steel shell	38 mm
Wall thickness of concrete	≈ 2 m

Fig. 1.51. PWR containment building: **(a)**, **(b)** vertical cross sections, **(c)** horizontal cross section (1: core, 2: pressurizer, 3: concrete containment, 4: reactor safety containment. 5: fuel element compact storage, 6: crane, 7: loading machine, 8: fresh steam armature, 9: steam generator, 10: storage corridor, 11: lock), **(d)** technical parameters. (Courtesy of Framatome ANP GmbH).

Specific airconditioning systems guarantee a lower pressure in the containment relative to the outer environment, therefore radioactive leaking can be sucked back into the inner containment space. Large areas of the containment can be entered by persons during operation. Different designs and design philosophies for PWR containments are realized in other countries. Especially other types of reactors, like boiling water reactors, require different technologies for this component.

1.2.2.5 Reactor safety systems

The very large radioactive inventory inside the core (around 4×10^{15} Bq/MW (th) in equilibrium during operation) must be retained inside the fuel elements in normal operation and in case of accidents. This has to be done with the help of reliable, redundant an diversified safety systems. Beside the retention of radioactivity the safe shutdown of the reactor from any operational state, as well as an effective decay heat removal and thereby cooling of the core are further important safety requirements. Parallel to accidents due to internal reasons, accidents due to outer reasons have to be governed, too. A primary reactor safety protection system triggers a fast reactor shutdown in case of all relevant accidents like larger leaks in the reactor cooling circuit, leaks in the feed-water or live-steam lines, or in case of uncontrolled removal of a shutdown rod from the core. Signals to actuate this system can be coolant temperature or pressure, neutron flux, height of water in the steam generators or in the pressurizer, or transient values of T, p, ϕ.

Figure 1.52 gives a schematic overview on the main systems like RESA (fast shutdown system), emergency cooling systems, lock of containment, which are controlled by the reactor safety protection system.

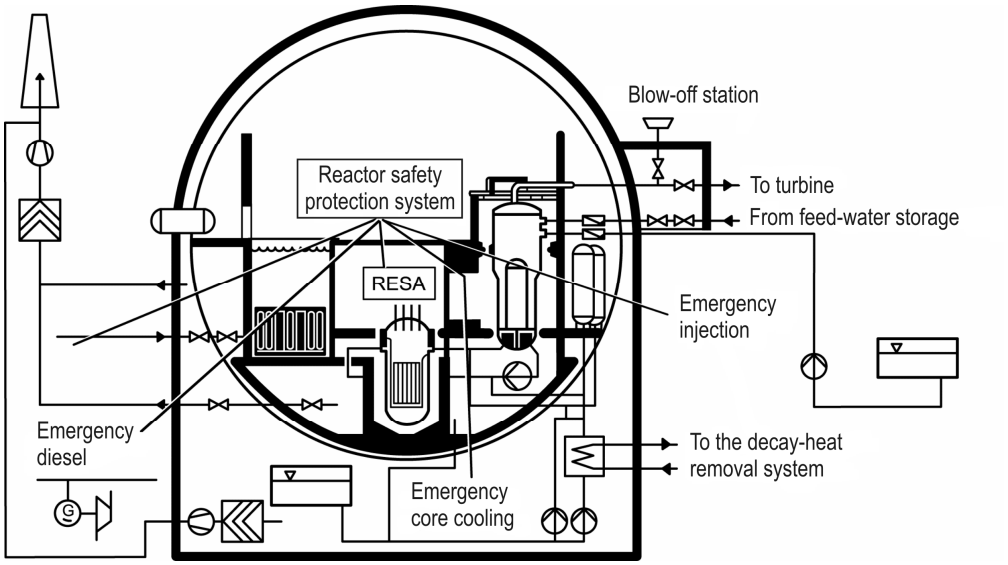


Fig. 1.52. Overview on safety systems of a PWR, which are controlled by the reactor safety protection system. (Courtesy of Framatome ANP GmbH).

Beside the shutdown by very reliable active shutdown systems, which is supported by an always strongly negative reactivity coefficient (void and temperature), the decay heat removal is of extended importance in all nuclear reactors. The β- and γ-decays of fission products produce decay heat in the reactor still after shutdown of the nuclear reactions. Directly after stopping the chain reactions this is around 6 % of the nominal power, after 1 hour nearly 1 % and after 1 day roughly 3 ‰.

In LWRs it is necessary to have systems which remove this heat very reliably. A total failure of these systems would cause a melting down of the core and eventually the release of large amounts of radioactivity to the environment (see Sect. 1.4.1). Therefore the reliability of core cooling systems is of great importance for the safety of a PWR. A modern PWR contains several different cooling systems, which are carried out in a redundant and diversified way.

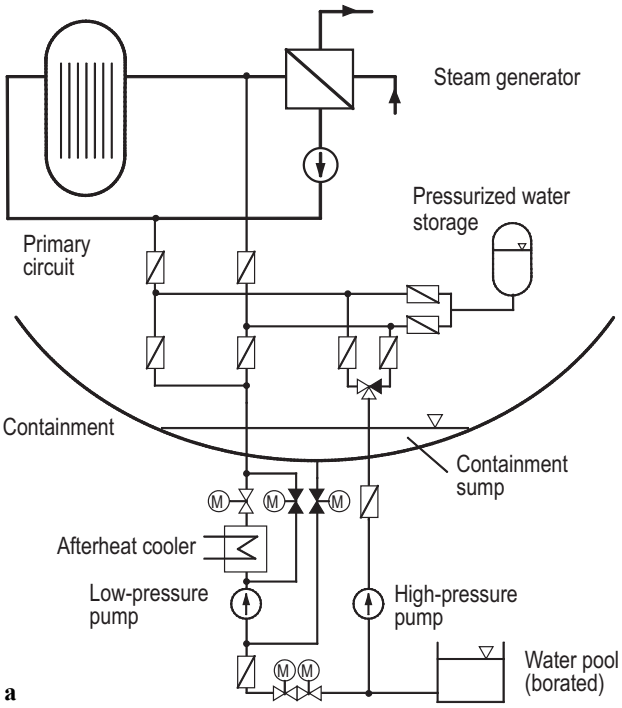
The following systems are available for decay heat removal:

- the normal cooling loops (4 times), each consisting of steam generator and coolant pump,
- high-pressure water injection systems (4 times),
- low-pressure water injection systems (4 times),
- a cooling system pumping the water from the sump of containment back to the primary circuit.

In Fig. 1.53 the principal flow sheet of this total concept is depicted. The injection systems are inserted if leaks in the primary circuits have occurred. Figure 1.54 contains some details of the decay-heat removal system showing the assignment to the different loops of the primary system.

A chain of the cooling system always contains heat exchangers, pumps, pipes, valves, cooling towers and systems for electricity supply. Hence there is a small chance of non-availability for these complex systems. If there is continuous supervision, maintenance and repair, a failure rate for loss of active cooling of around 10^{-6}/year is estimated in probabilistic safety analyses for large modern PWRs.

An important aspect in connection with the availability of decay heat removal systems is the spatially separated arrangement of decay-heat removal systems in the containment and inside the plant. Safe supply with water and electricity is one of the very important aspects.



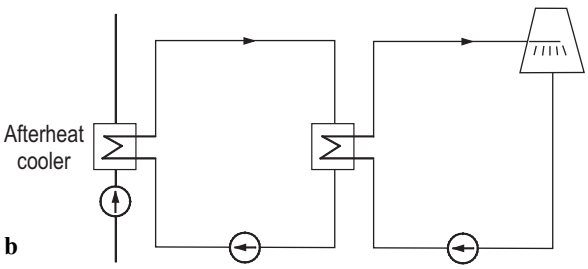


Fig. 1.53. Schematic representation of **(a)** decay-heat removal systems in modern PWRs, and **(b)** heat removal from the afterheat cooler.

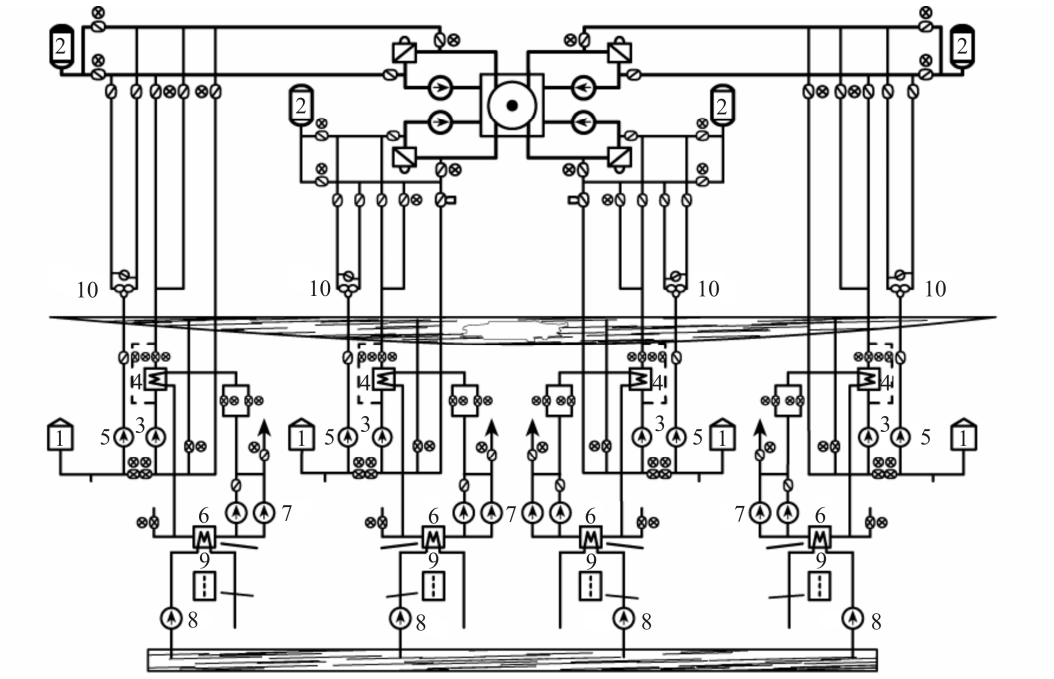


Fig. 1.54. Decay-heat removal system of a modern PWR, 1: flood tank, 2: pressure storage, 3: after-cooler, 4: decay heat pump, 5: safety feed injection, 6: intermediate cooler, 7: intermediate cooling pump, 8: compensation cooler, 9: secured intermediate cooler, 10: three-way valve. (Courtesy of Framatome ANP GmbH).

1.2.2.6 Auxiliary systems

Nuclear reactors need different auxiliary systems for normal operation. Important examples in case of the PWR are the water purification systems and the volume control system, as well as the chemical injection and additional coolant purification system (Figs. 1.55 and 1.56).

Furthermore extensive nuclear intercooling systems and airconditioning systems are installed. To deliver the necessary electrical energy in case of accidents emergency diesel engines are available. Installations for decontamination and laboratories to handle radioactive substances are part of the normal equipment of every nuclear power plant. The supervision of the primary circuit and of the total power cycle requires extensive measurements of various parameters like neutron flux, temperatures, pressures, mass flows, concentrations of different substances and of many radioactive isotopes in air, water or in solid wastes.

1.2.2.7 Steam turbine plant

The primary circuit and the secondary power cycle of a PWR are separated by the steam generator. Therefore the steam cycle process can be optimized separately. The saturated-steam cycle is arranged as displayed in Fig. 1.57. The pressure of live steam in this example is 54 bar with a humidity of 0.25 %. For the generation of electrical energy a one-shaft condensation turbine with 1500 rotations/min is used. A three-phase asynchronous generator is coupled to the turbine. The turbine has four housings, the first one is a double-flow high-pressure part, the three others each contain a double-flow low-pressure part of the turbine. Behind the high-pressure turbine the steam passes a water separator and is reheated by a bypass of live steam. The condensor behind the turbine is designed to cool down the whole amount of live steam in case of failure of the electrical system.

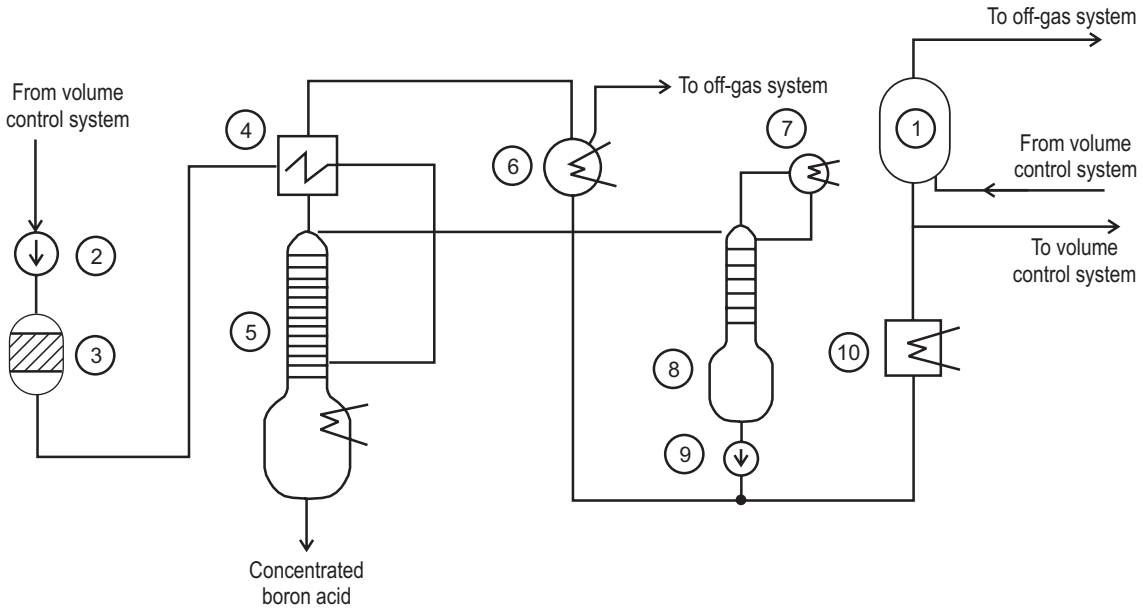


Fig. 1.55. Processing and storage of coolant, 1: storage for coolant, 2: pump for the evaporator, 3: ion exchanger, 4: preheater, 5: evaporator, 6: condenser, 7: reflux condenser, 8: degasser, 9: degasser removal pump, 10: cooler.

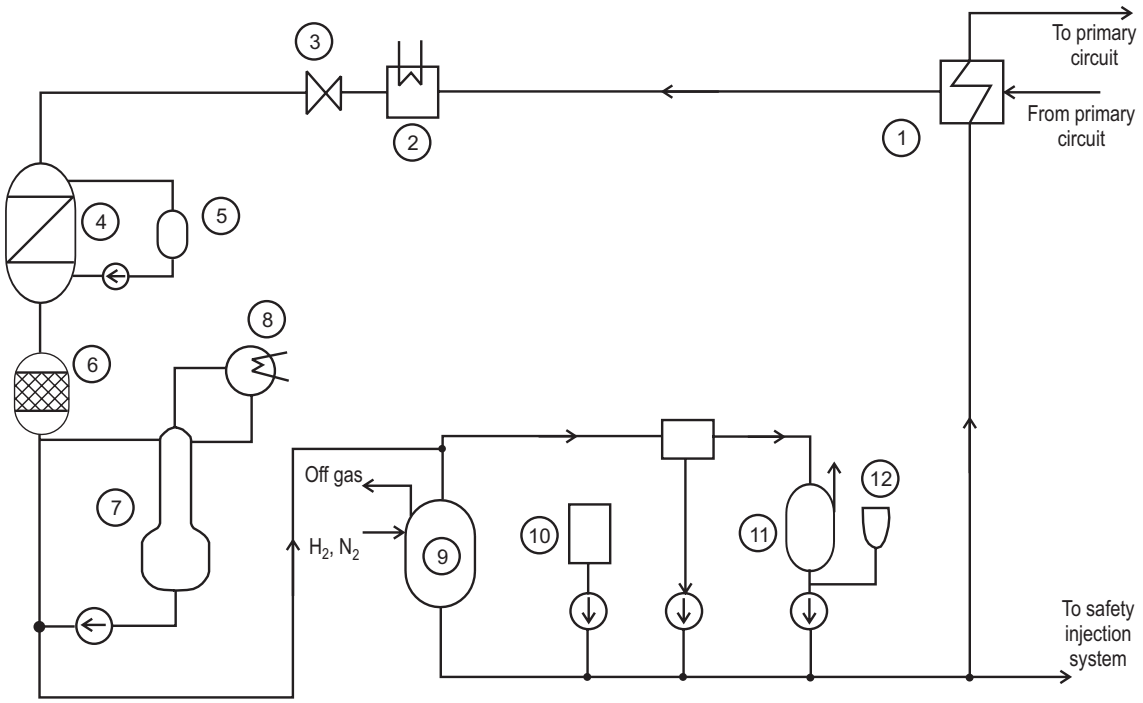


Fig. 1.56. Volume control system, chemical injection and coolant purification, 1: recuperator, 2: high-pressure cooler, 3: high-pressure reducing station, 4: mixed bed filter, 5: resin waste storage, 6: mechanical filter, 7: degasser for coolant, 8: condensor, 9: volume equalization storage, 10: chemicals mixing vessel, 11: storage and mixing vessel for boron acid, 12: boron acid storage. (Courtesy of Framatome ANP GmbH).

The regenerative preheating of feed water contains a 6-stage preheating of the condensate before the latter enters the steam generator again. The main condensate pumps and the main feed water pumps are usually designed with sufficient redundancy, for example 3 × 50 %. The generator operates at a voltage of 27 kV and feeds into the electrical grid of 380 kV via two parallel transformers. Large nuclear power plants in Germany are operated with wet cooling towers as heat sink and therefore the condenser vacuum is restricted to values of around 0.08 to 0.09 bar. The *H-S* diagram of the process indicates the expansion lines (Fig. 1.58). Because of the low pressure and temperature of the live steam (54…60 bar (saturated)) the efficiency of LWR plants today is limited to 33 %.

For hot-steam processes efficiencies of more than 40 % are possible. This cycle is used in AGR, HTR and LMFR plants.

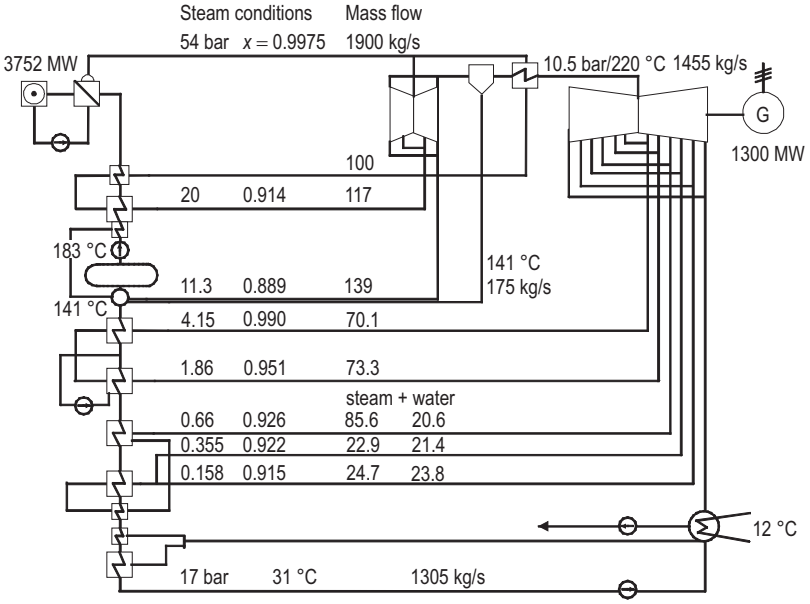
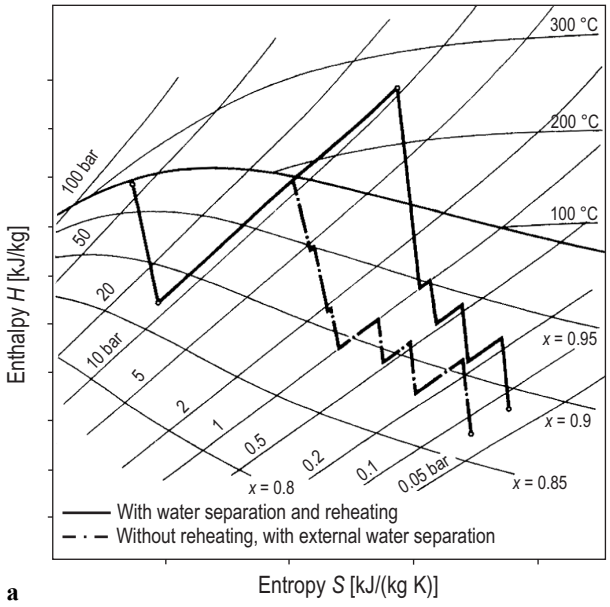


Fig. 1.57. Simplified steam cycle of a PWR. The steam content in the two-phase mixture is denoted by *x*. (Courtesy of Framatome ANP GmbH).



a

Thermal power	3752 MW
Electrical power	1300 MW
Steam generator loops	4
High-pressure turbines	1
Low-pressure turbines	4
Frequency of turbine	3000 min^{-1}
Waste heat removal	wet cooling tower
Pressure in condenser	0.04 bar

b

Fig. 1.58. **(a)** Enthalpy versus entropy (*H-S* diagram) for a saturated steam cycle, and **(b)** some important parameters of the steam cycle.

1.2.3 Thermohydraulic aspects of the core and of the fuel elements

The discussion of thermohydraulic aspects of the core and the fuel elements includes the power density in the core, the heating up of the coolant and structures in the core, as well as aspects of flow, heat transfer in the fluid and pressure drops [57Bon, 63Gla, 77Wei]. The power density inside the reactor core is defined by the expression

$$\dot{q}'''(\vec{r}) = \int_0^\infty \bar{E}_{\mathrm{sp}} \cdot \Sigma_{\mathrm{f}}(\vec{r}, E) \cdot \phi(\vec{r}, E) \cdot \mathrm{d}E \,, \tag{1.226}$$

where Σ_{f} is the macroscopic cross section for fission; if several different fissionable isotopes are in the core (U-235, Pu-239, Pu-241), the integral includes the sum of all fission reaction rates.

The nuclide densities of fissionable materials and the neutron flux have a non-uniform spatial distribution in the core, therefore the thermal power of the core is calculated by an integration over the whole core. For a cylindrical core (see Fig. 1.59a) one obtains:

$$P_{\mathrm{th}} = \int_{(V_R)} \dot{q}'''(\vec{r}) \cdot \mathrm{d}V = \bar{E}_{\mathrm{sp}} \cdot \int_{(V_R)} \Sigma_{\mathrm{f}}(\vec{r}) \cdot \phi_0 \cdot \cos\left(\frac{\pi \cdot z}{H}\right) \cdot J_0\left(\frac{2.405 \cdot r}{R}\right) \cdot \mathrm{d}V \tag{1.227}$$

With suitable average values the thermal power can be approximately calculated by

$$P_{\mathrm{th}} = \bar{E}_{\mathrm{sp}} \cdot \Sigma_{\mathrm{f}} \cdot \phi_{\mathrm{th}} \cdot V_R = \bar{\dot{q}}''' \cdot V_R \,. \tag{1.228}$$

The average power density inside the core, $\bar{\dot{q}}'''$, with the unit MW/m^3 (= kW/l), is a main parameter of the core which influences the design of the fuel elements, the core dimensions, the magnitude of the neutron flux, the pressure drop in the core, and especially the safety characteristics of the reactor in accidents with loss of coolant and cooling. Average core power densities realized in nuclear reactors operating today are summarized in Table 1.19.

Table 1.19. Average core power density of different reactor types.

Reactor type	Coolant	Average core power density [MW/m^3]
Magnox	CO_2	0.5
AGR	CO_2	2
HTR	He	2…6
PWR	H_2O (liquid)	100
BWR	H_2O	50
CANDU	D_2O	50
RBMK [a])	H_2O	50
FBR	Na	300…500

[a]) Related to fuel channels.

Figure 1.59b shows a measured axial distribution of the power density in a PWR core. The magnitude of the core power density, which is required to be technically feasible, economically reasonable and acceptable from the safety point of view, results from a complex optimization procedure for each reactor type.

The coolant is heated up during its flowing through the reactor core. In Fig. 1.60 the unit cell, the axial power density and the coolant temperature of a water-cooled reactor (PWR) are shown qualitatively. The power density inside a fuel pellet, $\dot{q}'''_{\mathrm{f}}$, the heat flux on the surface of the pellet, $\dot{q}''$, and the linear power of the rod, $\dot{q}'$, are correlated as follows:

$$\dot{q}'(z) = 2\pi R \cdot \dot{q}''(z) = \pi R^2 \cdot \dot{q}'''_{\mathrm{f}}(z)\,. \tag{1.229}$$

Characteristic average values in a large PWR are $\bar{\dot{q}}'''_{\mathrm{f}} \approx 300$ W/cm^3, $R = 0.5$ cm, $\bar{\dot{q}}'' = 75$ W/cm^2, $\bar{\dot{q}}' \approx 220$ W/cm.

For the coolant channel the energy balance results in

$$\dot{m}^* \cdot c \cdot \mathrm{d}T = \dot{q}'_0 \cdot \cos\left(\frac{\pi z}{H}\right) \cdot \mathrm{d}z\,, \tag{1.230}$$

with $\dot{q}'_0$ as the maximum linear power of the rod,

$$\dot{q}'_0 = \pi R_1^2 \cdot \bar{E}_{\mathrm{sp}} \cdot \Sigma_{\mathrm{f}} \cdot \bar{\phi}_0\,. \tag{1.231}$$

For the coolant temperature in z-direction one obtains

$$T(z) = \frac{\dot{q}'_0 \cdot H}{\dot{m}^* \cdot c} \cdot \frac{1}{\pi}\left[\sin\left(\frac{\pi z}{H}\right) + 1\right] + T_{\mathrm{i}}\,, \tag{1.232}$$

with T_{i} being the inlet temperature for the reactor core. For the total heating up in the core one finds the simple expression

$$T_{\mathrm{o}} - T_{\mathrm{i}} = \frac{\dot{Q}_{\mathrm{rod}}}{\dot{m}^* \cdot c}\,, \tag{1.233}$$

with $\dot{Q}_{\mathrm{rod}}$ denoting the power of one rod, T_{o} the outlet temperature from the channel, and $\dot{m}^*$ denoting the mass flow in the channel.

As the mass flow through the whole core is $\dot{m} = \dot{m}^* \cdot z$, with z denoting the number of rods, the total power of the core is $P_{\mathrm{th}} = z \cdot \dot{Q}_{\mathrm{rod}}$.

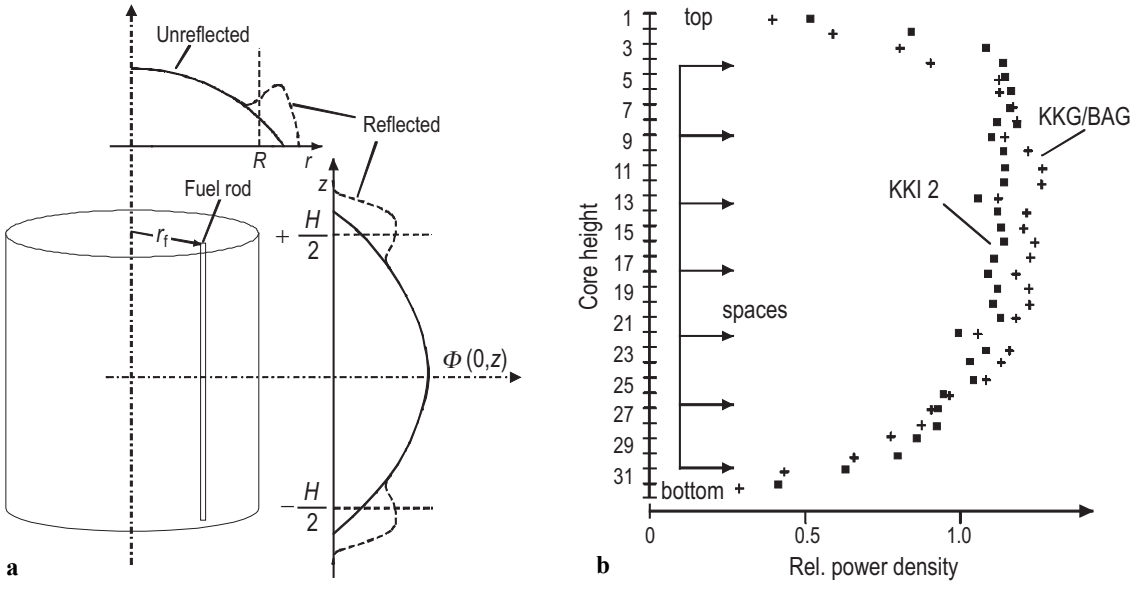


Fig.1. 59. Spatial distribution of the core power density in nuclear reactors: **(a)** theoretical distribution in case of a cylindrical core, **(b)** measured axial power density (relative values) in a large German (squares) and Swiss (crosses) PWRs. Top and bottom denote the upper and lower bound, respectively, of the fuel elements.

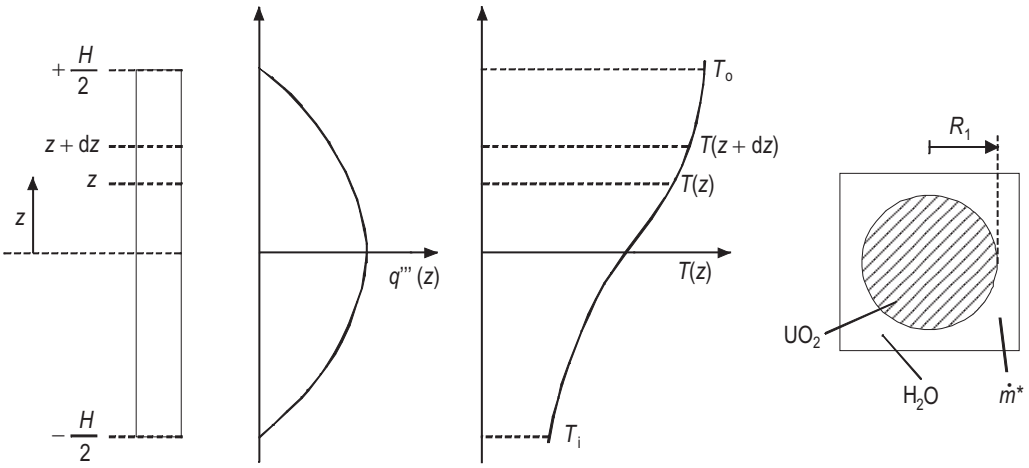


Fig. 1.60. Simplified model to estimate the heating up of coolant in a channel of a PWR.

For a large PWR some characteristic values are as follows: $P_{th} = 3800$ MW, $z \approx 50000$, $\dot{Q}_{rod} \approx 76$ kW; the total mass flow through the core is around 18800 kg/s, whereas the mass flow of one coolant channel is 0.38 kg/s.

Some characteristic parameters of other types of coolant circuits are summarized in Table 1.20.

The temperature distributions in the fuel elements depend on the power density, the coolant temperature, the heat transfer number in the coolant, and on from the geometric data of the fuel elements.

Certain parameters of the canning (λ, s) influence the temperature in this part of the fuel element, too. Figure 1.61 shows the qualitative dependence of the temperatures of coolant, canning and center of fuel for a nuclear reactor core, e.g. of a PWR.

Table 1.20. Characteristic parameters of coolant loops in different reactors.

Parameter		PWR	BWR	AGR	HTR	LMFR
Thermal power	[MW]	3800	3800	1500	200	750
Coolant		H_2O (liquid)	H_2O	CO_2	He	Na
Pressure of coolant	[bar]	160	70	40	60	10
Inlet temperature in core	[°C]	290	215	287	250	380
Outlet temperature from core	[°C]	325	285	651	700	550
Mass flow	[kg/s]	18800	14300	3687	85	3400

Principally in each channel of the core the coolant temperature can be described by the equation

$$T_{cool}(z) = T_i + \int_{-H/2}^{z} A \cdot \dot{q}'''(z')\,dz'. \tag{1.234}$$

For the canning (inside or outside) the following relation is valid:

$$T_{cann}(z) = T_{cool}(z) + \dot{q}''' \cdot \psi(\lambda_i, r_i, \alpha, \text{geometry}), \tag{1.235}$$

where ψ is a function describing the relevant heat transfer inside the fuel element. For the fuel temperature on the center line one finds

$$T_f(z) = T_{cool}(z) + \dot{q}''' \cdot \phi(\lambda_i, r_i, \alpha, \text{geometry}), \tag{1.236}$$

with a further function ϕ for the overall heat transfer inside the fuel element. The calculation of the functions ϕ and ψ requires a detailed analysis of heat transport in the fuel elements.

As an example for cylindrical rod geometry, from the stationary partial differential equation of heat conduction,

$$\mathrm{div}\left(\lambda\,\mathrm{grad}T\right)+\dot{q}_{\mathrm{f}}'''\left(\vec{r}\right)=0\,, \tag{1.237}$$

one obtains the radial temperature profile shown in Fig. 1.62, which depends on both the channel considered and the vertical position along the rod.

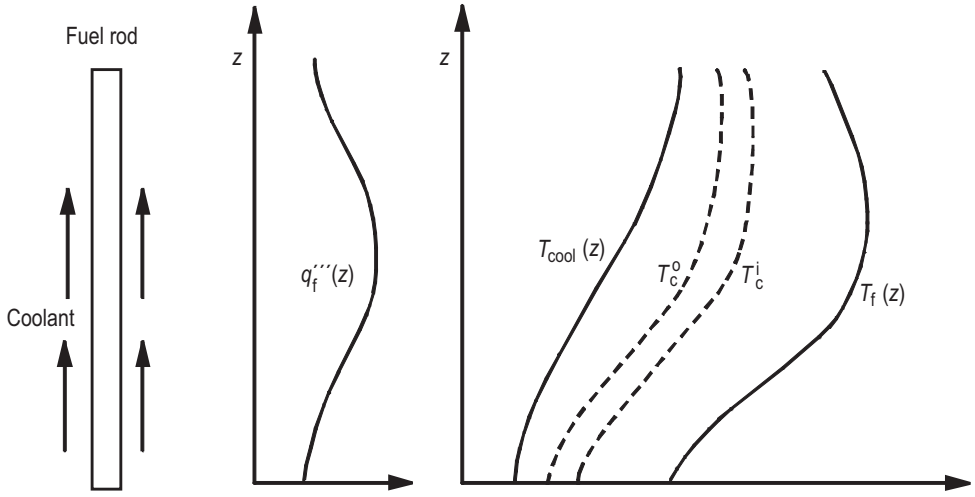


Fig. 1.61. Qualitative dependence of coolant temperature, canning temperature, and central fuel temperature on the core height.

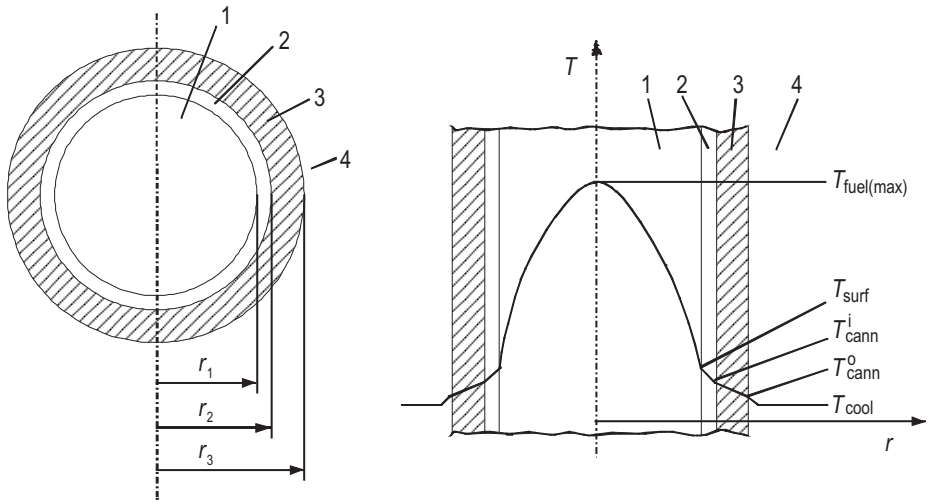


Fig. 1.62. Radial temperature profile in a cylindrical-rod fuel element; 1: fuel pellet, 2: gap, 3: canning, 4: coolant.

Applying the solutions of the differential equation for the different parts of the fuel elements the temperature difference between surface and center of the fuel pellet is given by

$$\Delta T_{\mathrm{f}} = T_{\mathrm{center}} - T_{\mathrm{surf}} = \frac{\dot{q}_{\mathrm{f}}''' \cdot r_1^2}{4\lambda_1} . \tag{1.238}$$

For the gap between the fuel and the inner side of the canning the result is

$$\Delta T_{\mathrm{gap}} = T_{\mathrm{surf}} - T_{\mathrm{cann}}^{\mathrm{i}} = \dot{q}_{\mathrm{f}}''' \cdot \frac{r_1^2}{2} \cdot \frac{\ln(r_2 / r_1)}{\lambda_2} , \tag{1.239}$$

whereas in the canning a temperature difference,

$$\Delta T_{\mathrm{cann}} = T_{\mathrm{cann}}^{\mathrm{i}} - T_{\mathrm{cann}}^{\mathrm{o}} = \dot{q}_{\mathrm{f}}''' \cdot \frac{r_1^2}{2} \cdot \frac{\ln(r_3 / r_2)}{\lambda_3} , \tag{1.240}$$

exists. Finally, with α denoting the relevant heat transfer number of the coolant, for the temperature difference between the outside of the canning and the coolant one finds

$$\Delta T_{\mathrm{cool}} = T_{\mathrm{cann}}^{\mathrm{o}} - T_{\mathrm{cool}} = \dot{q}_{\mathrm{f}}''' \cdot \frac{r_1^2}{2 \cdot r_3} \cdot \frac{1}{\alpha} . \tag{1.241}$$

When designing PWR fuel elements, three main conditions must always be fulfilled:

- In the hottest rod, i.e. the rod with the highest power, the center fuel temperature must stay below the melting temperature of UO_2 (2850 °C).
- The canning temperatures must stay below allowable material temperatures ($T_{\mathrm{cann}} < 450$ °C).
- The coolant must not be evaporated in normal operation (i.e., for the heat flux, $\dot{q}'' < \dot{q}''_{\mathrm{crit}}$ is always required).

Particularly the first requirement limits the linear power rating of a rod. The conductivity integral,

$$\chi\left(T^*\right) = \int_{T^*}^{T_{\mathrm{c}}} \lambda(T) \cdot \mathrm{d}T \approx \overline{\lambda}_1 \cdot \left(T_{\mathrm{c}} - T^*\right), \tag{1.242}$$

must stay below allowable limits. Figure 1.63 presents the linear power rating as a function of the burn-up, including the parameter χ. Maximum values of around 200 W/cm are used for the linear power rating in practical operation.

To fulfill the requirement that the fluid should not be evaporated in a channel of a PWR, the safety factor in the thermohydraulic design of the fuel elements sufficiently high. Figure 1.64 presents the heat flux as a function of the difference between surface and saturation temperature. This curve is often called Nukijama curve or curve of pool boiling.

The critical heat flux is around 10^6 W/m^2. The maximum heat flux in operation should be less than $10^6/1.3$ W/m^2 in the core to avoid damage of the fuel element surfaces.

In the core there are around 50 000 rods in case of a large PWR. For the hottest rod and rods with an average heat production there exists a hot-channel factor, F_q, defined by the expression

$$F_q = \frac{\dot{q}'''_{\max}}{\overline{\dot{q}'''}} = \frac{\dot{q}''_{\max}}{\overline{\dot{q}''}} = \frac{\dot{q}'_{\max}}{\overline{\dot{q}'}} . \tag{1.243}$$

This factor, which has a value of around 2 in large PWRs (see Fig. 1.65) depends on several parameters: the spatial power distribution in the core, deviations in the geometrical data of fuel elements and in material parameters, as well as deviations in the homogeneity of the mass flow. During burn-up the value will be slightly reduced.

For the heating up of the coolant in the core a similar hot-channel factor is defined:

$$F_{\Delta h} = \frac{\Delta h_{\max}}{\Delta \bar{h}}, \tag{1.244}$$

with the condition that net evaporation in a channel of a PWR should not occur. Therefore the increase in temperature in all channels is limited. Values of $F_{\Delta h}$ near 1.5 are characteristic for current PWR designs.

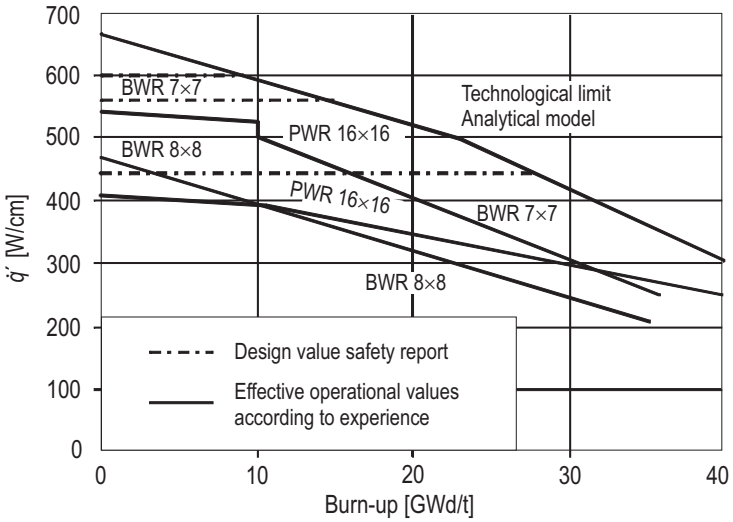


Fig. 1.63. Linear power rating of PWR and BWR fuel rods versus burn-up of the fuel.

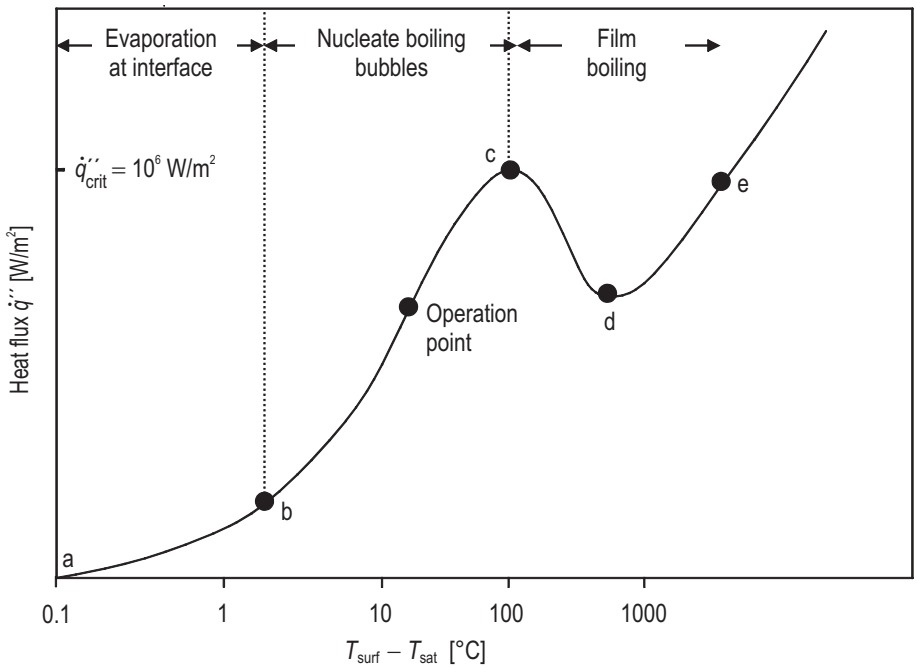


Fig. 1.64. Heat flux versus temperature difference between surface and liquid in case of pool boiling.

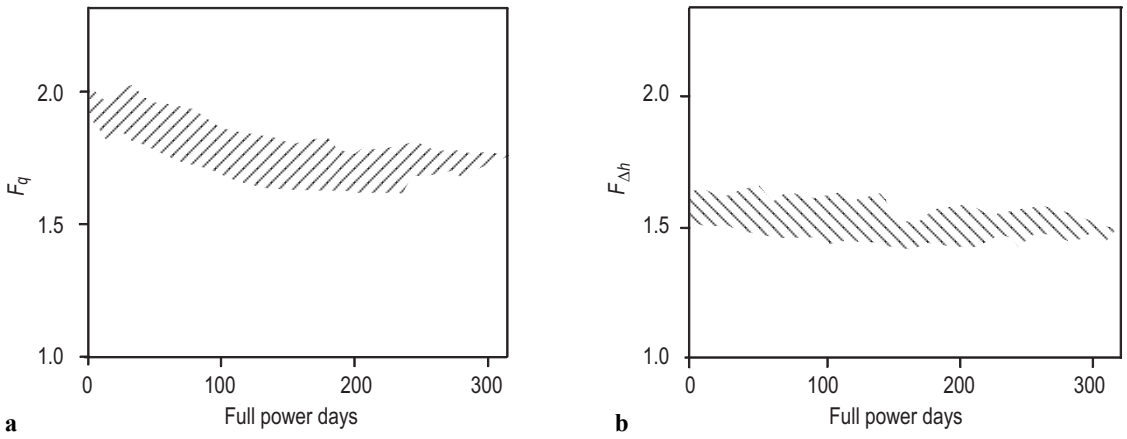


Fig. 1.65. Hot-channel factors **(a)** for the power density, F_q, and **(b)** for the heating of the coolant, $F_{\Delta h}$, in case of a large PWR core.

Naturally just a small part of the rods has hot-channel factors near 2, as being displayed by the statistical distributions in Fig. 1.66.

The thermohydraulic aspects of a large core PWR might be characterized by the following data. A core with a thermal power of 3800 MW contains around 5×10^4 fuel rods. With a diameter of 1 cm and a length of nearly 4 m this represents a total surface of 6000 m^2 inside the core. Therefore the average heat flux is nearly 600 kW/m^2, and the linear power rating of the rods is 190 W/cm. Using a heat transfer number of 10000 W/m^2K in the pressurized water the characteristic canning temperature is 380 °C, the center temperature in the fuel pellets in a normal channel would be 1200 °C. In the hottest channels, with $F_q \approx 2$, the maximum fuel temperatures can exceed 2200 °C.

This design gives sufficient certainty for normal operation. Therefore the number of failures of fuel cannings is very small in modern reactors. In case of accidents the conditions of power production and cooling are changed, in some cases dramatically, and therefore fuel elements can be damaged (see Sects. 1.2.5 and 1.4).

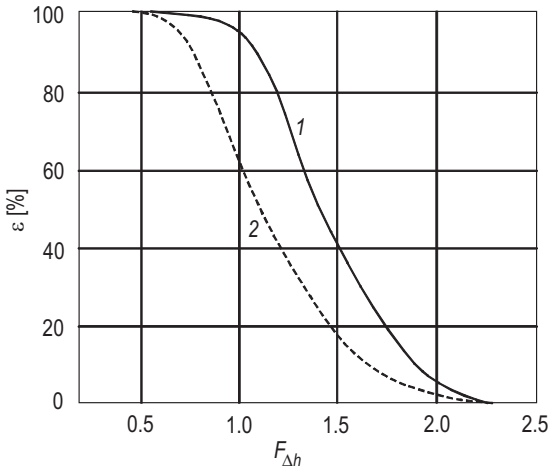


Fig. 1.66. Percentage of rods (curve 1) and cannings (curve 2) that have hot-channel factors larger than $F_{\Delta h}$.

1.2.4 Operating experience

1.2.4.1 Availability of power plants

A very important parameter of a successful operation of nuclear power plants is the availability of work, V_a, defined by the expression

$$V_a = \int_0^{1a} \frac{P_{el}(t) \cdot dt}{P_{el}^0 \cdot 8760\,h} = \frac{T}{8760\,h}, \tag{1.245}$$

where T is the integral time of full-power operation in one year. This number depends on the conditions of the grid, the costs of the fuel cycle and, naturally, on the plant operation conditions. During the last decades this number became much better in countries operating nuclear power plants. The respective results of German power plants in the year 2001 [01Jah] are presented in Fig. 1.67.

An average value of $V_a \approx 8000$ h/year is possible today. This is important for economical analysis. The time availability, V_t, i.e. the percentage of time in which the plant is available for operation, is important as well:

$$V_t = \frac{T^*}{8760\,h}, \tag{1.246}$$

with T^* denoting the number of hours the plant is ready for operation; T^* is larger than T. Excellent plants achieve time availabilities of 98 % and working availabilities of 96 %. In the last years not only the availability has been raised worldwide by technical progress and experience, but also the number of fast shutdowns has been drastically reduced. The reduction of personnel doses is an additional indicator for progress in the operation of nuclear power plants.

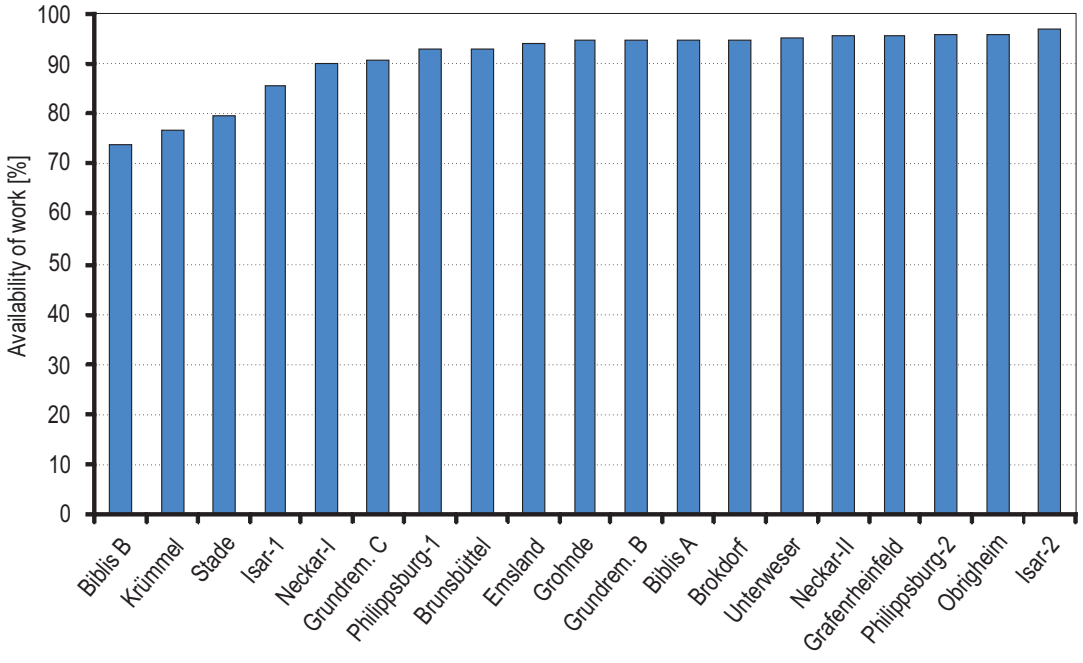


Fig. 1.67. Availability of work in German nuclear power plants in 2001.

1.2.4.2 Release of radioactivity to the environment

Radioactive materials build up inside the fuel elements by the fission process, as well as in the coolant and structures by activation. The different components like the fuel elements, the primary circuit and the containment act as barriers to retain the radioactivity inside the nuclear power plant. Figure 1.68 shows the system of barriers in case of a PWR.

Depending on the conditions of normal operation or accidents these barriers are efficient in different ways. In normal operation nearly all the fission products are retained inside the fuel elements (see Fig. 1.69).

The content of activity in the primary coolant is mainly caused by activation products. Very efficient systems are used to clean the water and to retain radioactivity inside the nuclear power plant. Radioactive substances are released from the plant in gaseous form via the stack, with waste water, and in solid form. Those values depend on the type of plant, the age and status of the plant, and on operational procedures. The main part of gaseous release rates is constituted by noble gases, C-14 and tritium (see Table 1.21). These values are very small compared to the allowed values following the licensing [01Zus, 95Mic]. As for the waste water the radioactivity contents are also very low today (see Table 1.22).

All release values are far below the allowed limits. This is the result of progress in purification and retention technologies and of the high quality of the fuel elements. Today the calculated radiological exposition of the population caused by emissions from nuclear power plants is very small compared to other expositions (see Fig. 1.70). Nuclear power plants contribute less than 1 % to the total radiological burden of the population. Further reduction is possible, but higher effort in retention and purification technologies would be required [95Mic].

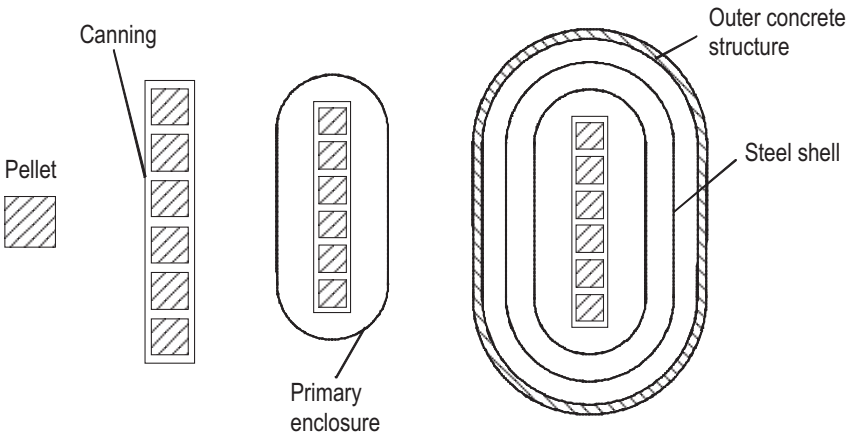


Fig. 1.68. Barrier system for fission product retention in a PWR.

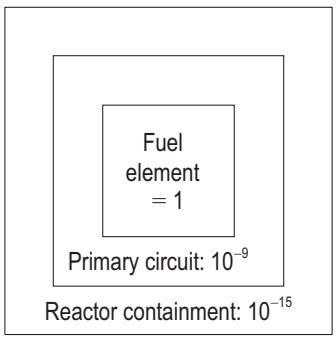


Fig. 1.69. Distribution of activity in normal operation in a PWR (related to the activity content of the fuel elements).

Table 1.21. Release rates in gaseous form of a large modern PWR (German conditions).

Substances	Release [Bq/year]	Allowed values [Bq/year]
Noble gases	9.75×10^{11}	
C-14	5.14×10^{11}	} 1.5×10^{15}
Tritium	3.85×10^{11}	
γ-radiators in aerosols	2.75×10^{5}	4×10^{10}
Iodine-131	2.61×10^{5}	1.5×10^{10}

Table 1.22. Release rates in liquid form of a large modern PWR (German conditions).

Substances	Release [Bq/year]	Allowed values [Bq/year]
Tritium	2.37×10^{13}	7×10^{13}
α-emitters	3.3×10^{5}	
γ-emitters	3.3×10^{7}	} 6×10^{10}
Fe-55	4.3×10^{6}	

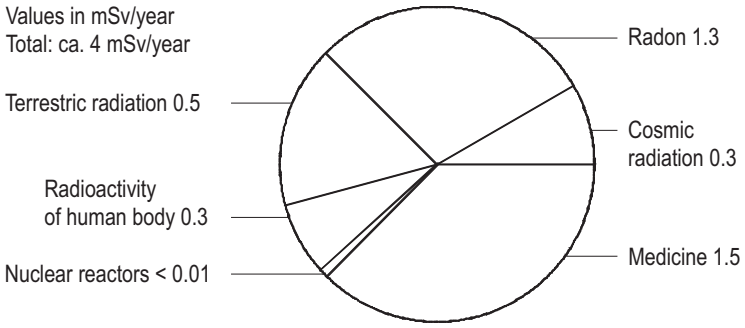


Fig. 1.70. Contribution of nuclear power plants to the average radiation dose received by individuals in Germany.

1.2.4.3 Radiation inside nuclear power plants

In a nuclear reactor the core is the main source for the emission of neutrons and γ-radiation [92Kni]. Activity transported with the coolant plays an important role, too, because the total primary circuit and auxiliary systems are contaminated. In case of BWRs the total steam turbine plant is to be considered with respect to contamination. Spent fuel elements at different stages of the fuel-element life cycle require additional attention with regard to radiation protection. Figure 1.71 shows the various possibilities of producing radiation as a result of fission inside a nuclear reactor. For calculating fields of n- and γ-radiation and suitable shielding systems some important aspects have to be considered. Neutrons and γ-rays are produced with broad spectra inside the core during fission. Neutrons which penetrate to outer parts of the primary system or shielding systems produce additional γ-radiation by (n, γ) reactions. Moreover, β-radiation occurs during the decay of fission products. Highly energetic β-radiation causes γ-radiation by the bremsstrahlung effect.

All these secondary and partly tertiary effects have to be taken into account in radiation shielding analyses. The shielding systems have to be designed to limit the radiation levels at all positions of the plant to allowed values. For some components the material embrittlement by fast neutrons is very important, e.g. for the reactor pressure vessel or for the cannings of fuel elements especially in fast reactors.

The neutron fluxes in a PWR are shown in Fig. 1.72 [02Boe]. It can be seen that the thermal flux drops from the center of the core to the rooms outside the shielding around the reactor vessel by a factor of 10^{10}, nearly the same reduction occurs for the fast neutron flux.

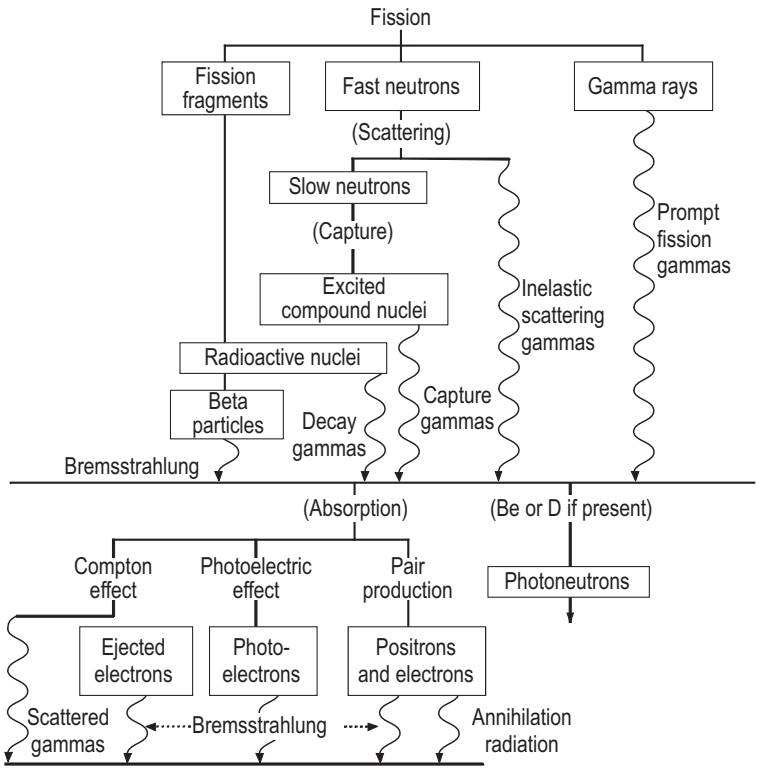


Fig. 1.71. Sorts of radiation produced by fission in a nuclear reactor.

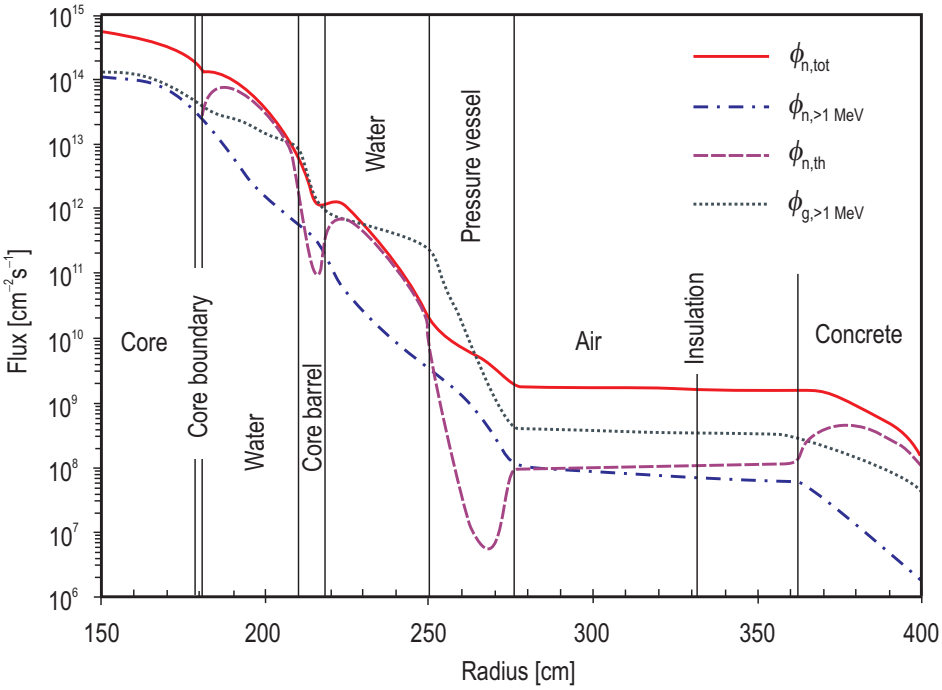


Fig. 1.72. Neutron fluxes, ϕ_n, and gamma flux, ϕ_g, in a PWR system in radial direction.

Using appropriate shielding structures – mainly water and concrete for neutrons, steel for γ-radiation – the radiation fields in modern nuclear power plants are limited to levels that allow inspection, maintenance and repair. Together with improved working strategies this results in a permanent reduction of radiation doses received by the operation personnel. The development of these radiation doses is displayed in Fig. 1.73 for nuclear power plants in different countries.

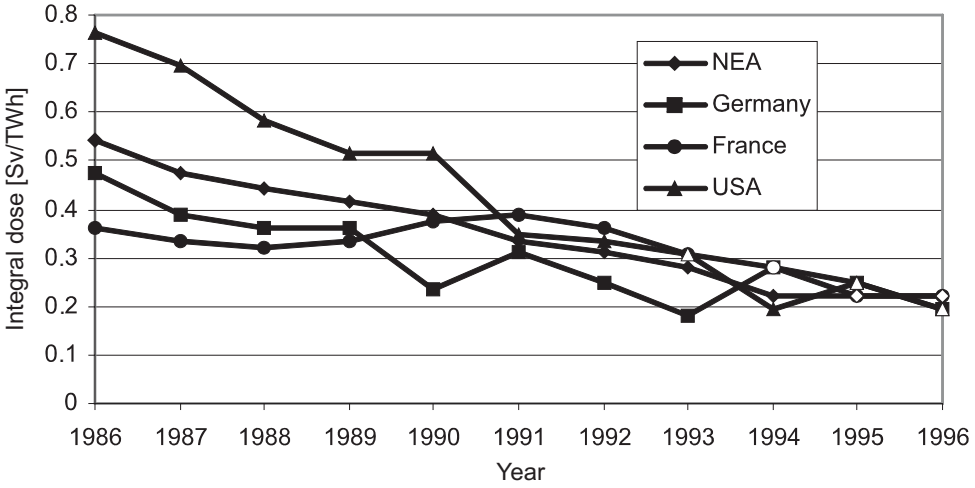


Fig. 1.73. Reduction of the annual radiation dose of operation personnel in nuclear power plants (NEA denotes the Nuclear Energy Agency).

1.2.4.4 Fuel handling

Caused by the burn-up of the fuel, in most nuclear reactors a reloading, in many cases yearly, is necessary. In a PWR the amount of fuel which has to be loaded once a year is estimated by a simple energy balance:

$$\int_0^{1\,\mathrm{year}} P_{\mathrm{el}}(t)\cdot \mathrm{d}t = P_{\mathrm{el}}^0 \cdot T = \overline{\eta}\cdot \int_0^{1\,\mathrm{year}} P_{\mathrm{th}}(t)\cdot \mathrm{d}t = \overline{\eta}\cdot B\cdot \dot{m}_{\mathrm{f}}\,. \tag{1.247}$$

Here, P_{el}^0 is the net electric power, T is the time of full-power operation, $\overline{\eta}$ the average efficiency, B the average burn-up, and $\dot{m}_{\mathrm{f}}$ the amount of fuel which has to be loaded during standstill. Characteristic values are: $P_{\mathrm{el}}^0 = 1300$ MW, $T = 8000$ h/year, $\overline{\eta} = 33$ %, $B = 40\,000$ MWd/t, leading to $\dot{m}_{\mathrm{f}} \approx 33$ t/year.

Nearly the same amount of heavy material has to be deloaded every year. Taking into account that the inventory of heavy material of a large PWR with the power mentioned above is 100 t, this results in changing 1/3 of the fuel elements every year.

The average burn-up of most PWRs worldwide is between 35 000 and 40 000 MWd/t. Further developments try to realize values of almost 60 000 MWd/t. Fuel handling takes place during a three-week standstill of the plant. The operation is carried out under water because of shielding and cooling. In the beginning of the operation the room containing the reactor pressure vessel and the surrounding region of compact storage is flooded (see Fig. 1.74). Then the head of the reactor pressure vessel is removed and positioned in a separate water storage. With the help of a special loading machine the spent fuel elements are removed from the core and transported under water to the compact storage inside the reactor containment. The fresh fuel elements are inserted into their positions using the same loading machine.

Some components as parts of the control system in the upper part of the reactor are removed, too, and stored in a separate water pool. After closing the reactor vessel, tests to guarantee the leak-tightness of the primary system are very important. The purification of the water from radioactive substances and the adjustment of the boron content are further necessary steps before the operation of the reactor can be

started again. As shown in Fig. 1.75 the spent fuel elements contain large quantities of fission products, plutonium, fissionable material which has not been consumed, and actinides which have been built up (see Fig. 1.24).

Safe intermediate storage in the compact storage inside the reactor containment is necessary as a first phase of storage. The essential requirements are subcriticality, safe decay heat removal, and reliable barrier function of the fuel elements to achieve safe enclosure of radioactivity inside the cannings. Naturally the compact storage needs sufficient protection with respect to impact from the outside as well.

The average storage time in the compact storage is three years, during this time decay heat and radioactivity are reduced (see Fig. 1.76). The spent fuel elements are then loaded into transport cases and transported to reprocessing plants or to an intermediate storage building.

In Germany in the next years an intermediate storage building is placed on the site of each nuclear power plant to avoid long-distance transports [01Sta, 01Zwi, 99Dro].

For the intermediate storage of spent LWR fuel elements water bassins and dry cask storage systems are in operation worldwide. Dry case storages have some advantages as far as safety, economy and operation procedures are concerned.

Figure 1.77 shows the concept of storage vessel and building. The vessels are manufactured from cast steel or cast iron and have ribs on the outside and a special closure system. The decay heat removal inside and from the surface of the vessel proceeds via conduction, radiation, and free convection of air.

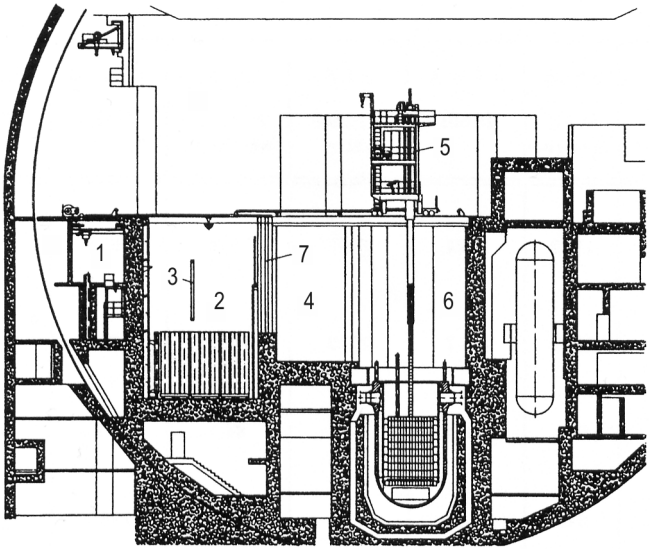


Fig. 1.74. Fuel handling in a PWR, 1: storage for fresh fuel elements, 2: compact storage for spent fuel elements, 3: transportation of spent fuel elements, 4: water storage for removed core internals, 5: fuel handling machine, 6: reactor cavity, 7: dam door. (Courtesy of Framatome ANP GmbH).

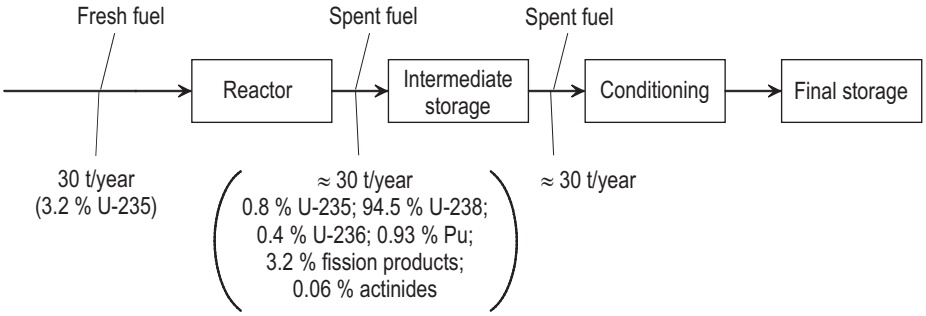


Fig. 1.75. Content of radioactive isotopes in fresh and spent fuel elements (PWR, $B \approx 34\,000$ MW d/t).

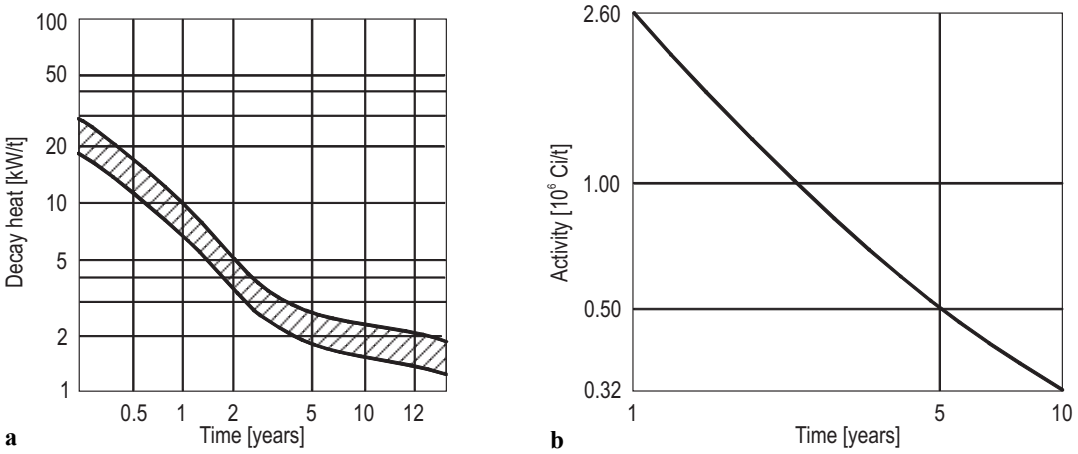
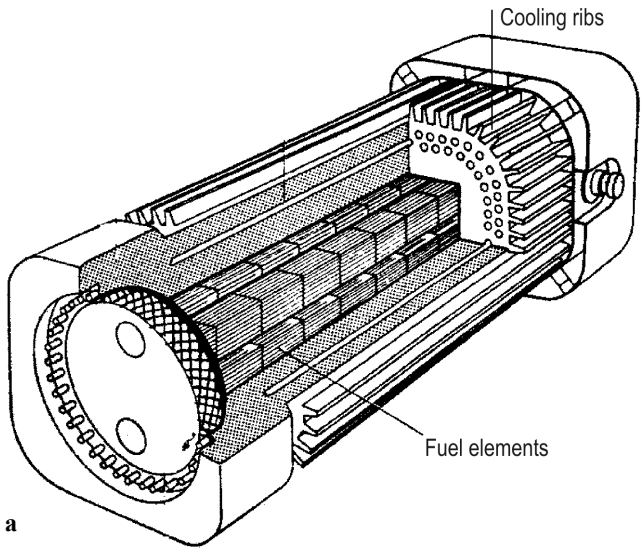
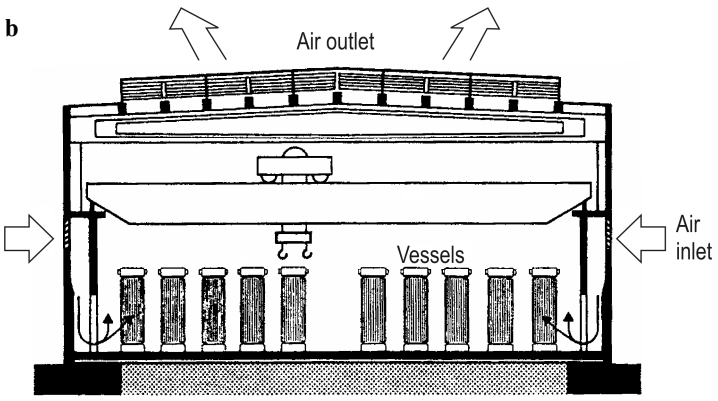


Fig. 1.76. Conditions for intermediate storage of spent fuel elements (PWR, $B \approx 33\,000$ MW d/t): **(a)** decay heat production, **(b)** specific radioactivity.





c

Number of fuel elements	19
Total mass of heavy metal	10.3 t
Max. enrichment	4 %
Average burn-up	5500 MWd/t
Total activity	5.5×10^{17} Bq
Max. power	40 kW
Total mass	125 t
Total height	≈ 6 m
Total diameter	≈ 2.5 m
Wall thickness	0.42 m
Max. fuel temp.	< 370 °C
Max. wall temp.	< 90 °C
Dose rate on surface	< 0.5 mSv/h

Fig. 1.77. Intermediate storage using dry casks: **(a)** storage vessel, **(b)** storage building, (c) technical parameters of a large dry cask for storage of spent PWR fuel elements.

Because of these self-acting heat transport mechanisms the fuel temperature stays at maximum values of 370 °C in the beginning of operation, the surface temperature of the vessels stays below 80 °C. The transport of heat from the vessels to the storage building and from there to the environment takes place just by free convection without any machine. This concept of decay heat removal never can fail. Even in case of terroristic attack on the building, e.g. using a large airplane with large amounts of gasoline, the function of the vessels to retain radioactivity is practically conserved.

With respect to fuel handling there are differences for other reactor types. These differences cause specific conditions as far as operation – or safety – behavior is concerned.

AGR, CANDU and RBMK systems use a quasi-continuous fuel handling, which means that fuel elements are changed during reactor operation. Therefore the excess reactivity required by rising burn-up (reducing the content of fissile material and raising the content of neutron-absorbing fission products) can be avoided in these types of reactors. This is an important safety aspect.

In case of the HTR with pebble-bed fuel there is a continuous loading of fresh fuel elements and deloading of spent fuel elements. Therefore no excess reactivity for burn-up compensation is necessary at all. In this type of reactor the small fuel elements move through the core under the influence of gravity and can be recycled before they reach the average specified burn-up. Caused by this continuous loading, deloading and recycling the fuel cycle of this reactor type is very flexible, the parameters of the fuel elements can be changed completely even during operation, as was demonstrated by the AVR reactor [90AVR].

1.2.4.5 Control

The power of the plant is controlled via the control of the electrical generator. The main parameters of the primary and secondary circuit change as depicted in Fig. 1.78 [86Boh].

Above a value of 50 % of full power the average temperature of the primary coolant is held almost constant, this procedure is applied for fast changes of power corresponding to load following operation. The temperature changes in the reactor cooling system are relatively small. The temperature and pressure of live steam change as well corresponding to the required power. Below 50 % power the parameters of temperature and pressure of live steam stay nearly constant and the mass flow must be changed. The average temperature of the reactor coolant drops at these values of lower power, too.

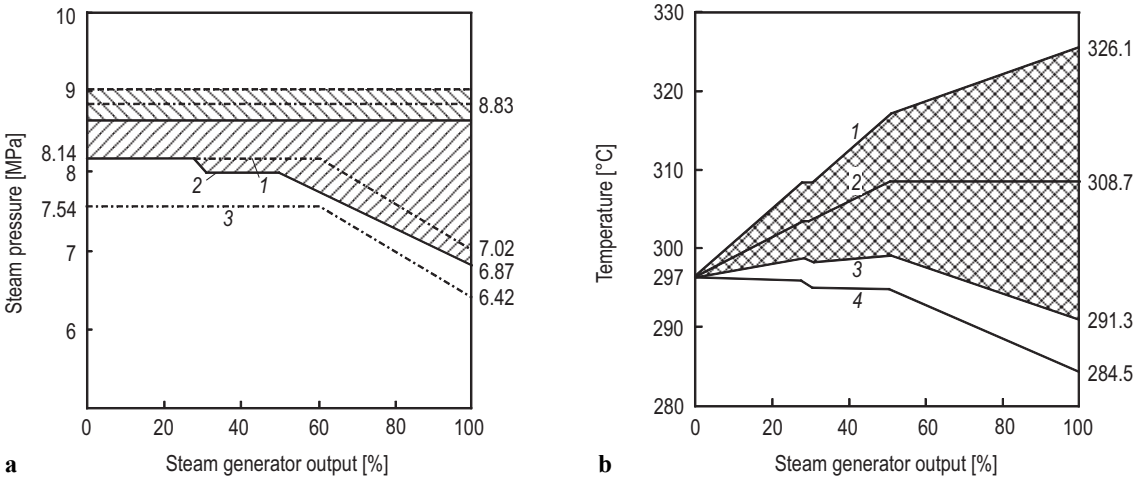


Fig. 1.78. Partial-load diagram of a large PWR plant: **(a)** steam pressure versus steam generator output (1: max. fresh steam pressure, 2: min. fresh steam pressure, 3: fresh steam pressure (steam generator)); **(b)** coolant and steam temperatures (1: outlet coolant temperature, 2: average coolant temperature, 3: inlet coolant temperature, 4: fresh steam temperature (steam generator)).

The control rods of the reactor are the most important parts of the chain to control the coolant temperature of the reactor. The borating and feed water systems are applied in the control procedure as well. At full power the control rods are positioned almost completely outside the core, therefore the highest shutdown capability is available.

Today power changes of 2 % per minute in the region near 80 % of full power and around 5 % per minute near 50 % power are possible. Long-term changes of reactivity caused by the burn-up of fuel and by xenon poisoning are compensated in PWRs with the help of boron in the coolant. In case of accidents such as loss of load, fast shutdown of the turbine, failure of the main coolant pumps, or failure of the feed- water supply groups, shutdown elements are injected into the core. In these cases the plant can stay in operation with reduced power.

Start-up and shutdown of the plant follow the schemes shown in Fig. 1.79: If the plant is started from the cold state the coolant of the primary circuit is heated up by operation of the main coolant pumps. Heating up occurs with the help of the pressurizer up to a pressure of 25 bar. Then the control rods are partly removed from the core and the boron content of the coolant is reduced. By these measures the reactor is made critical. By nuclear heating the temperature of the coolant rises at a rate of 50 °C/h. After around 10 hours the plant is ready for operation and can be started to reach full power [74Old, 93Eme].

During a short standstill the reactor and the steam cycle can stay in warm condition, which allows a fast restart of the plant. In case of a shutdown procedure the power of the plant is reduced down to 20 %. After that the average coolant temperature is reduced until shutdown. The cooling of the core in a first phase is done with the help of the steam generators, in a second phase the decay-heat removal system goes into operation.

1.2.5 Accidents in the design area

1.2.5.1 Overview

From the very first, safety considerations played a very important role in the development of nuclear technology. Accidents were assumed and possible consequences and countermeasures were considered. In the beginning reactivity accidents played a major role, after that loss-of-coolant and cooling accidents became more important. In the recent past, especially severe impacts from the outside and their influence on reactor safety have come into focus [79Smi, 89Per, 77Lew, 80Deu, 89GRS, 77Far, 87Hau, 85Wil].

The large inventory of radioactive material in the core and the requirement that this must always stay inside the nuclear plant made reactor safety a very sensitive field. The neighborhood of a nuclear plant must be protected against the high radiation levels. Naturally the operator has a large interest to protect the plant and thereby his investment. High quality of components, reliable operation, and protection against accidents are requirements from this side, too.

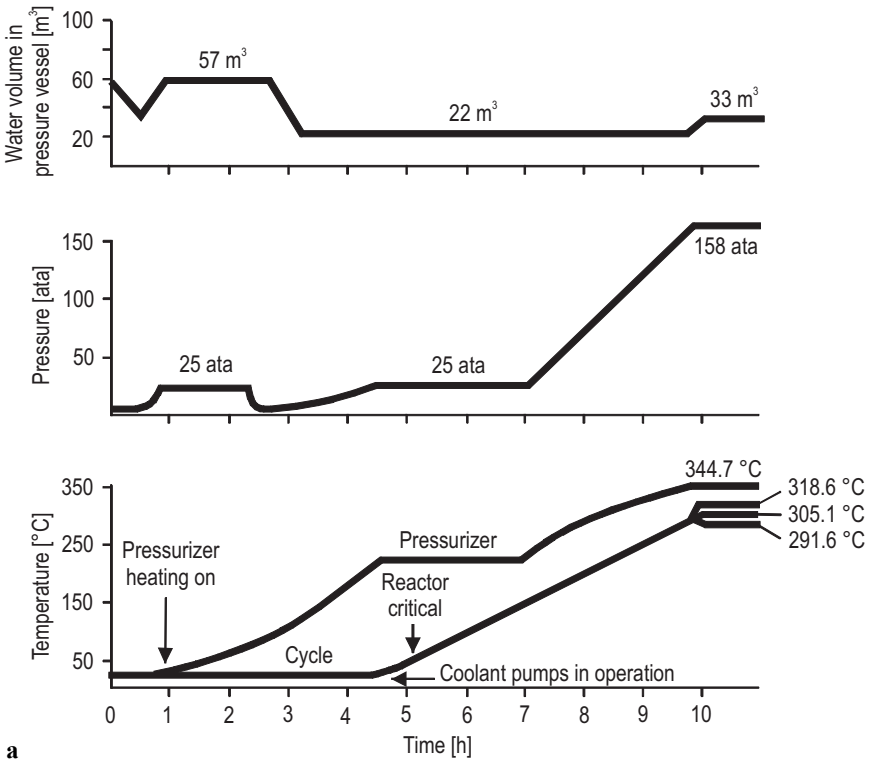
The main safety-related aspects are:

– safe shutdown of the nuclear chain reaction;
– safe removal of decay heat, limitation of all temperatures below allowed values;
– sufficient protection of the plant against impact from outside.

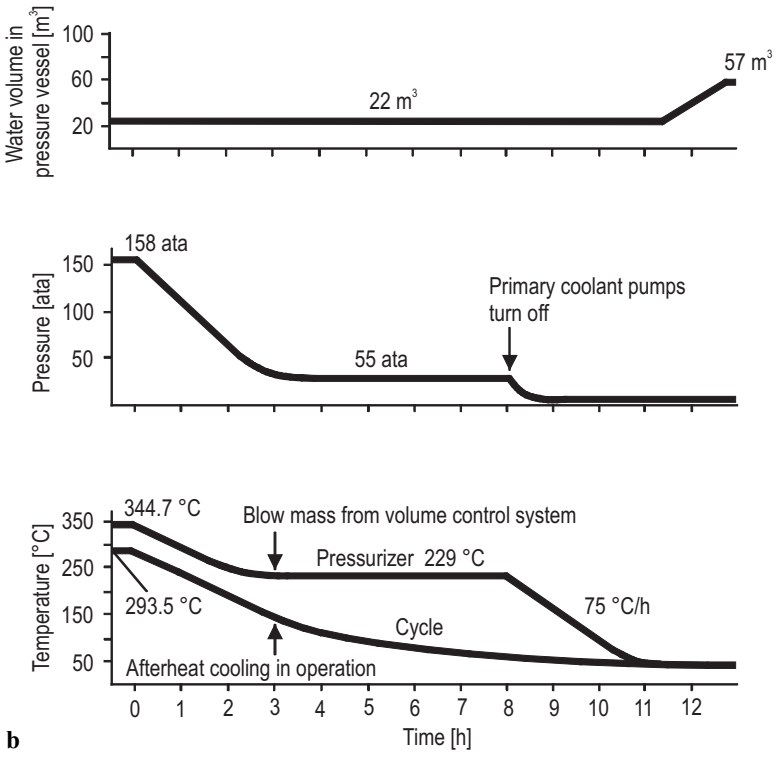
Fulfilling these requirements the radioactive materials can be retained inside the plant even in case of severe accidents. In order to realize this goal, in the last decades comprehensive safety rules have been developed and have been realized in plants worldwide. The main aspects are:

– avoidance of accidents by suited design of plants;
– installations to govern accidents and to avoid damaging consequences.

Measures to avoid accidents are for example: safe design of the plant; suited design of components and systems; choice of suited materials; broad analysis of stresses and fatigue; supervision of embrittlement; high requirements on quality management, inspection, maintenance and repair. Considerable efforts are made to discover disturbances early enough during normal operation and to limit the consequences.



a



b

Fig. 1.79. (a) Start-up and **(b)** shutdown of a large PWR plant. (Courtesy of Framatome ANP GmbH).

In order to govern accidents different safety installations are available, which include passive and active properties. Typical passive properties of current reactors are: barriers against the release of radioactivity (fuel elements, enclosure of primary circuit, reactor containment), passive decay-heat removal systems (large water storages using gravity, evaporation, condensation), passively acting shutdown systems (like strongly negative temperature- or void-coefficients).

In case of the PWR the following active safety installations are realized: reactor protection system (which receives various signals from transients and which initiates reactor shutdown and start-up of decay-heat removal systems); fast shutdown system, decay-heat removal and emergency cooling systems; emergency feed water system; emergency diesel generators; closure of reactor containment; accident management systems (to reduce the pressure of the primary system, to supply water to steam generators).

The very large inventory of fission products inside a nuclear reactor core – a typical value in equilibrium being $\sigma \approx 3\times10^{10}$ Bq/MW (th) – contains isotopes which are important with respect to inhalation and ingestion in case of severe accidents. Table 1.23 contains some isotopes with characteristic data. Iodine-131 as an example is important for short-term consideration after an accident. Cesium-137 and strontium-90 are relevant for long-term ground contamination and possible resettlements because of their very long half-lives.

Table 1.23. Fission products bearing radiological consequences after severe accidents.

Isotope	Half-life	Fission yield [%]	Reactor inventory [a]) [Ci/kW (th)]	Important for
I-131	8.1 d	2.9	26.3	thyroid
I-135	6.7 h	5.9	53.6	thyroid
Sr-90	28 a	5.9	53.6	bones
Y-91	58 d	5.9	53.6	bones
Cs-137	33 a	5.9	53.6	muscles
Ru-103	40 d	2.9	26.3	kidney
Te-129m	34 d	1.0	9.1	kidney

[a]) In equilibrium.

Disturbances and incidents cannot be avoided in technical plants. The incidents can be characterized corresponding to their consequences and their probability of occurrence:

- Disturbance of operation: these occur relatively often, the consequences are negligible with respect to reactor safety.
- Incidents in the design area: these are governed by the available safety systems. There will be no remarkable release of radioactivity from the plant. The occurrence probability will be relatively small.
- Severe accidents beyond design area: (1) beyond the accidents in the design area, accidents could occur with large release rates of radioactivity from the plant. The occurrence will be very improbable, but those accidents can be thought to happen for reasons which are within the concept of the plant.
- Severe accidents beyond design area: (2) there could be accidental situations which are even beyond the category above (1); these can be explained by very strong impact from the outside (very strong earthquakes, terroristic attack). The occurrence probability of these accidents is assumed to be very small.

The list of possible accidents of a PWR is long, the following explanation is just an overview over some very important situations:

- loss of coolant,
- loss of active decay heat removal,
- break of steam generator pipes,
- breaks in the feed water or steam pipe system,
- loss of electrical power,

– damage to the turbine-generator system,
– reactivity accidents,
– accidents at fuel handling,
– fire inside the plant,
– impact on the plant from outside.

For all accidents in the design area, measures are provided to govern the respective situation. In the licensing procedure, in deterministic and probabilistic analyses the effectiveness of these measures is shown and proven. In the analysis of accidents, combinations of different accidents are assumed, sequences and consequences for the plant and the environment are analyzed. Accident management procedures and their results belong to such safety analyses today as well. Figure 1.80 contains the main steps of this type of probabilistic risk analysis.

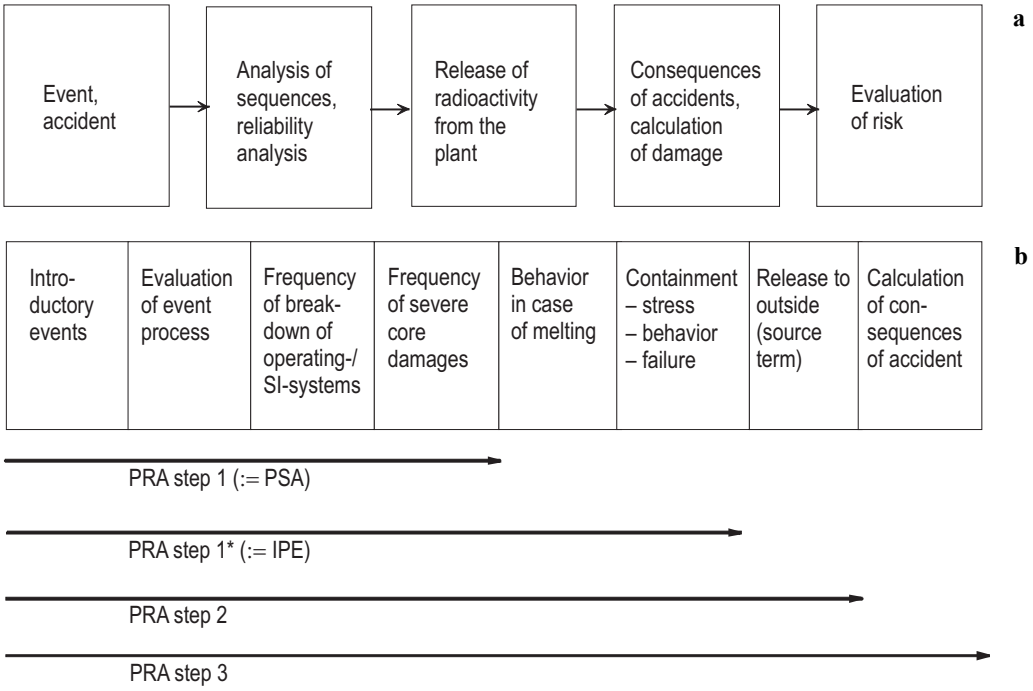


Fig. 1.80. Steps in risk analysis: **(a)** main aspects of probabilistic risk analysis, **(b)** different stages of risk analysis (PRA: Probabilistic Risk Analysis, PSA: Probabilistic Security Analysis, IPE: Individual Plant Examination).

1.2.5.2 Loss of coolant

In connection with loss-of-coolant accidents the decay heat production of nuclear reactors is of great importance. The time dependence of the decay heat production is described quite well by the relation

$$P_D(t)/P_{th} = 0.0622 \cdot \left(t^{-0.2} - (t_0 + t)^{-0.2}\right) \qquad (t, t_0 \text{ in s}). \tag{1.248}$$

Here, t_0 is the time of operation the nuclear chain reaction, and t is the time after shutdown.

The safe removal of decay heat from the reactor core is one of the central requirements of reactor safety. If the decay heat removal failed, the result would be a core meltdown in nearly all reactors known today, especially in LWRs.

For the PWR several different assumptions are made for leaks and breaks in the primary coolant circuit. A very important case is the double end break of a main primary coolant pipe, in earlier times this

accident was called GAU (in German: Größter Anzunehmender Unfall), today this situation is an important one in a broad spectrum of loss-of-coolant accidents (LOCA).

In case of a sudden break of a large primary coolant pipe the total water content of the primary system flows out in a very short time into the reactor containment. During the depressurization the water will be almost completely evaporated at the openings of the break.

The steam velocity will arrive at the critical velocity. The pressure in the primary system will be reduced to the saturation value in a very short time. A two-phase mixture of steam and water flows into the containment. In the meantime the reactor has been shut down caused by different signals, e.g. the primary coolant pressure being too low.

In the containment a maximal pressure of around 5 bar occurs, which is below the design pressure of this component. The temperature in the containment atmosphere stays below 150 °C and this temperature drops again after some hours by heat transfer to the containment structures.

Inside the core the temperature of the fuel rods will rise up, because the conditions for heat transfer have changed completely. Some cannings will fail, fission products will be released from these defect rods to the primary system and from here to the containment.

At the end of this first phase of coolant loss the canning temperatures are reduced already by the cooling by steam. In a second phase the reactor core is filled again with water. After reaching a primary coolant pressure of nearly 26 bar the emergency cooling system will operate. Borated water is injected from the storage vessels into the primary system. It is possible to inject into the hot or cold leg in the primary coolant piping system.

After 150 s the whole core is covered with water again. Figure 1.81 shows the time behavior of water volume and pressure in the core. The curve labeled NW 800 corresponds to an accident of the type "double ended rupture of primary coolant loop".

If the pressure of the primary circuit arrives at 10 bar, the flooding of the core starts. For a time span of around 20 min borated water is injected into the primary circuit. The water leaves the primary circuit through the leak and collects in the sump of the reactor containment. After that follows the phase of long-term emergency cooling of the core. The pumps of the respective system will be switched over automatically to pump the sump water through the core. A special decay heat cooler which is followed by a chain of cooling installations takes out the decay heat from the primary system.

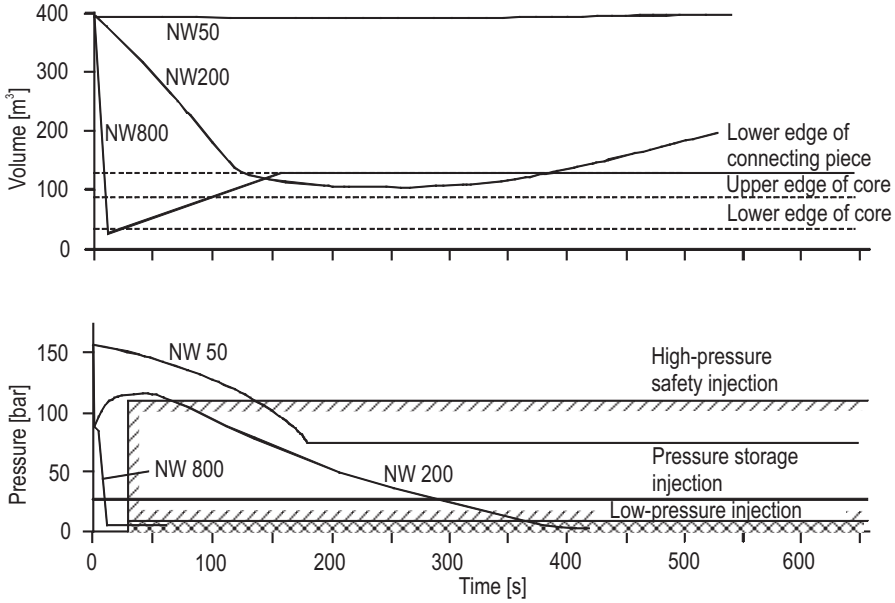


Fig. 1.81. Water volume and pressure in the primary system as a function of time in case of loss-of-coolant accidents (the parameter being the dimension of opening in the primary system). (Courtesy of Framatome ANP GmbH).

Figure 1.82 contains the time dependence of the canning temperature. In the hottest channels a canning temperature of nearly 1100 °C is typical, whereas in most rods the maximum temperature stays below 800 °C. After around 100 s the canning temperatures will be reduced to values below 200 °C.

In the analysis of radiological consequences of this accident the following assumptions are made (German conditions):

- Related to the core inventory the following amount of fission products is released into the containment: 10 % of the noble gases and halides, 5 % of the volatile solid fission products (Cs, Te, Ru), 0.1 % of other solid materials.
- 75 % of all radioactive substances which are released into the containment are deposited, washed out in this building, only the noble gases are not affected.
- As for the halogens, which are still airborne inside the containment, 85 % are in elementary form, 10 % as organic compounds, and 5 % as aerosols.
- Filters behind the containment, which do work after such an accident, have the following efficiencies: 0 % for noble gases, 99 % for organic compounds of halogens, 99.9 % for all other substances.

Therefore, if the emergency cooling and the closure of the reactor containment work, the release rates of radioactivity to the environment are small. It the emergency cooling works and if the containment closure fails, the release rates are medium. If the emergency cooling fails, the core will melt causing very high release rates into the environment.

Small and medium leaks have less consequences in any case. Very large leaks in the primary circuit, which could be caused by catastrophic damage to the reactor pressure vessel, belong to the so-called residual risk. If a pressure vessel bursted, core cooling would be impossible and the core would melt after a short time. Perhaps pieces of a bursting vessel could damage the containment and an early release of large amounts of radioactivity to the environment could be the consequence. Today large efforts are made to achieve very small bursting probabilities. Values of less than 10^{-7}/year are thought to be feasible.

To avoid bursting of vessels is one of the safety features of high temperature reactors, which are discussed in Sect. 1.4.

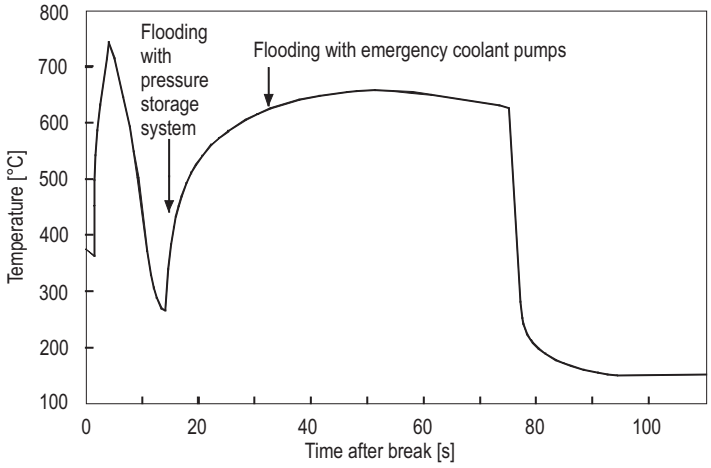


Fig. 1.82. Canning temperatures versus time, for a 1300 MW (el) PWR in case of breakage of the main coolant pipe.

1.2.5.3 Break of steam generator pipe

In case of break of a steam generator pipe in a PWR the primary coolant, which has a pressure of 160 bar, flows into the secondary circuit with a pressure of 60 bar. The flow rate can be estimated to follow

$$\dot{m} \sim A \cdot \sqrt{\Delta p} \,, \tag{1.249}$$

with the leak area, A, and the pressure difference, Δp, between primary and secondary side. As a consequence the pressure in the primary system drops and in the secondary system the pressure increases. Ra-

dioactivity ingresses to the secondary circuit and water has to be released from it. In the primary circuit water has to be injected to cool the core. Via the reactor protection system a fast shutdown of the reactor occurs, after that a fast reduction of pressure in the primary system and thereby a stabilization of the filling height of water in the pressurizer take place. If the pressure of the primary circuit is reduced to around 80 bar, then the defect steam generator can be isolated without opening of safety relief valves in the secondary circuit. Therefore the release of primary coolant via the secondary circuit is avoided.

If only one steam generator pipe breaks, there will be a stable status for the state of filling in the pressurizer, and the emergency water supply will not go into operation. If several steam generator tubes break at the same time, the emergency systems inject water into the primary system after the filling level of the pressurizer has dropped below a specified level.

Figure 1.83 shows results of an analysis in which it was assumed that one steam generator tube breaks and that additionally the spraying system inside the pressurizer fails. After the break there is one hour time until the steam generator is totally filled with water, then the safety injection pumps have to stop their operation.

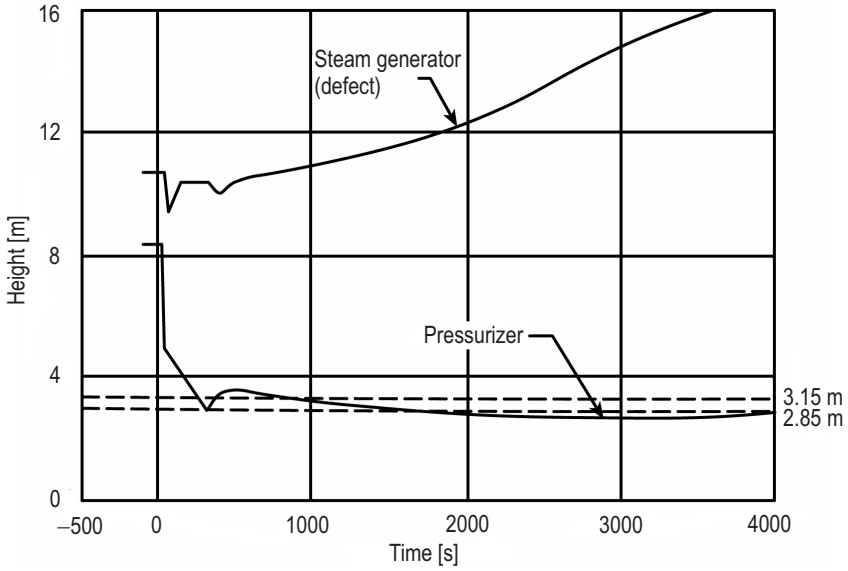


Fig. 1.83. Filling level in the defect steam generator and the pressurizer after break of steam generator pipe and failure in the spraying system of the pressurizer.

1.2.5.4 Loss of power and failure of turbine

The electrical power required by a nuclear power plant is normally supplied by the electrical grid, into which the plant itself feeds its energy. For reasons of redundancy a nuclear power plant is normally connected to two grids. Today the non-availability of these two grids in Europe is thought to be less than 10^{-2}/year.

The main users of electricity in a PWR, like main coolant pumps, feed water pumps and further pumps in auxiliary systems, are fed by the grid. The typical demand of a 1300 MW (el) PWR is around 7 MW. If the grid fails, the cplant is disconnected from the grid and the power ouput of the plant is reduced to its own consumption level. The plant then works in insular operation. If the insular operation fails, too, the reactor will be shut off. The supply of electrical energy then takes place with the help of the emergency power system.

The functions which are necessary for the safety of the reactor, like decay heat removal, pressure reduction, closing the lock of the containment, as well as control and measurement installations, are supplied by this system consisting of diesel generators. There are additional battery systems for some specific

purposes. New reactor concepts will be designed in such a way that the safety functions do not depend on electrical energy, for instance the self-acting decay heat removal, as intended for the HTR, does not need any machine. In this case the heat is removed just by natural convection of air, conduction and radiation. Failure of components in the secondary part of the power plant can cause transients. The reasons can be damage to the turbine, the generator or the pressure vessels in the turbine hall.

After loss of turbine power of the supply of steam to the turbine must be stopped immediately to avoid overspeeding of the turbine (see Fig. 1.84). A steam valve in front of the turbine is closed and – after opening a valve in front of the condenser – the steam is injected into this component and condensed there completely. If this procedure fails, the steam can be blown off into the environment with the help of safety valves.

In order to avoid that parts of an exploding turbine could ever damage the containment in modern nuclear power plants, the reactor containment and the turbine hall are arranged in a way as shown in Fig. 1.85. Pieces of the turbine flying in tangential direction relative to the turbine shaft cannot hit safety-relevant parts of the reactor.

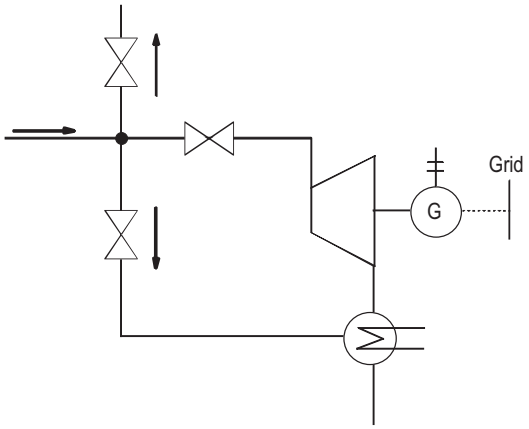


Fig. 1.84. Installation for fast shutdown of the turbine (injection of steam into the condenser or blow-off of steam into the environment).

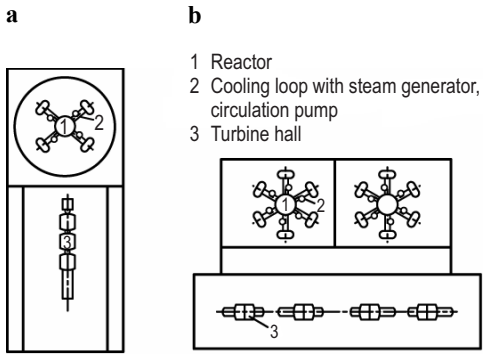


Fig. 1.85. Arrangement of reactor containment and turbine hall for protection against turbine explosion: **(a)** modern concept (WWER 1000); **(b)** old concept (WWER 440).

1.2.5.5 Break of pipes in the secondary system

Breaks in the secondary system of a PWR have consequences in the primary circuit and thereby in the core by the coupling through the steam generator. Therefore not only the thermohydraulic but also the neutron-physical conditions inside the core can be changed. Important accidents are breaks of the feed water or steam pipes. Figure 1.86 shows the schematic arrangement of these components in a PWR plant.

If a steam pipe breaks behind a shut-off valve a reduction of pressure in all steam generators occurs (see Fig. 1.87), because they are connected with the leak via the sampler for fresh steam. A fast shutdown of the reactor is initiated and a fast shut-off of the turbine. The signal for the reactor safety protection system is $dp/dt <$ max. value in a fresh steam pipe.

After that the valves for the fresh steam are closed and the main feed water pumps are stopped. The steam generator which is mainly influenced by the break will be separated totally from the feed water supply. Thereby too strong a reduction of the cooling temperature and a reactivity feedback are avoided.

The feed water supply of the remaining steam generators is accomplished with pumps for starting and diversified and redundant injection systems. If by an additional failure the shut-off valves for the steam pipes stay open, total evaporation of the water in the steam generator can occur. In the other steam generators, after closing of the valves for the fresh steam, the pressure will rise up to 75 bar, then safety valves will blow off.

If a feed water pipe breaks between the steam generator and the last reverse-flap the water level and pressure in the steam generator drop. Fast shutdown of the reactor is initiated by signals like dp/dt. After fast closure of the valve in front of the turbine and opening of the valve in front of the condenser, the main feed water pumps are stopped and the blow-down stations are closed. No further water is supplied to the defect steam generator loop.

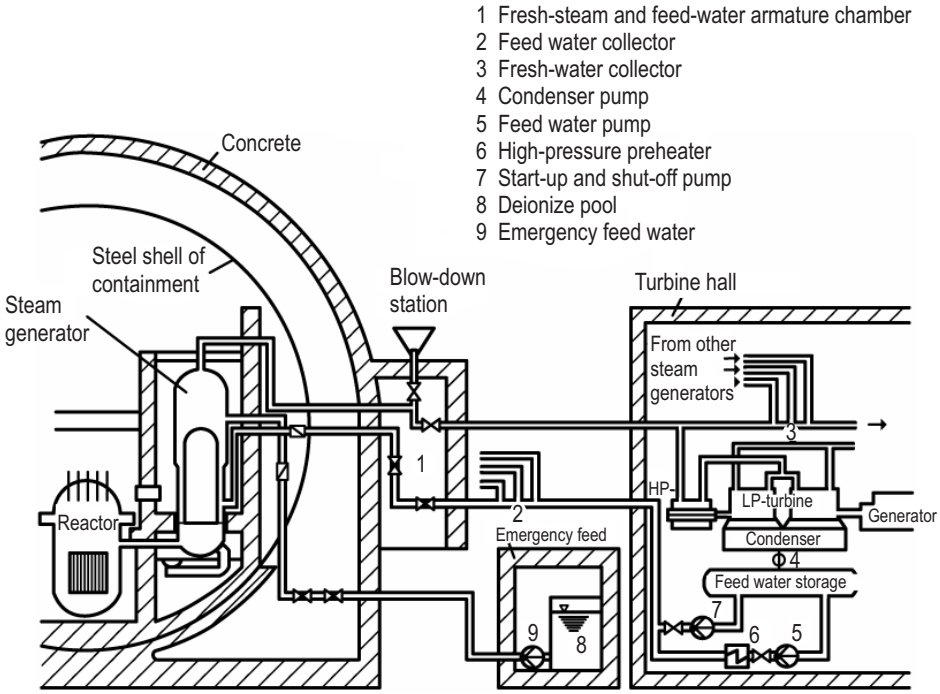


Fig. 1.86. Schematic arrangement of feed water and steam pipes in a PWR plant. (Courtesy of Framatome ANP GmbH).

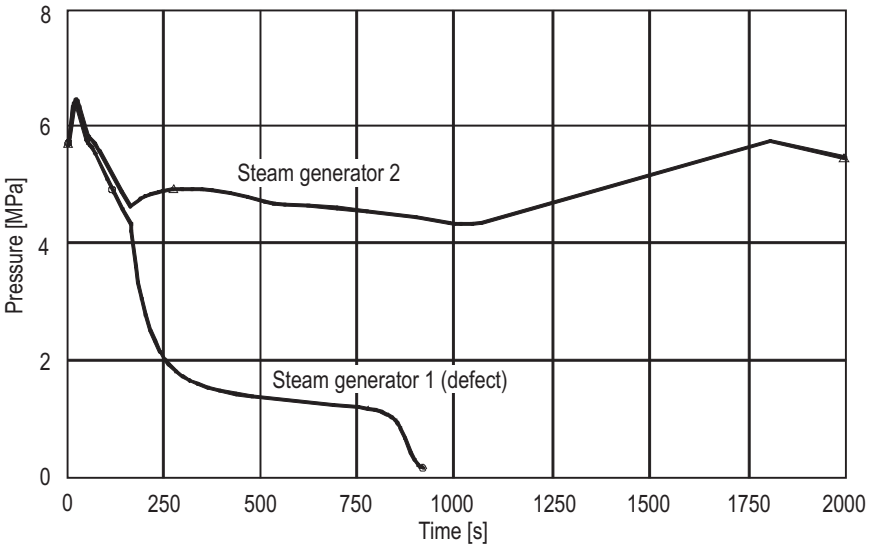


Fig. 1.87. Pressure in steam generators in case of a pipe break.

1.2.5.6 Loss of a control rod

Disturbances of the reactivity balance can be caused by malfunction of rods, or by changes of the moderator characteristics. As for the PWR, changes of the moderator temperature or density, as well as changes of the boron concentration have to be considered. As a result of a failure in the control system it could happen that the control rods move out of the reactor core with maximal velocity. In this case there would be a gain in reactivity. This would cause feedback effects on the reactor power, the mass flow of steam, the coolant pressure and temperature, as well as on the safety factors against boiling in the core. In Fig. 1.88 two different cases are considered, a fresh core and one with higher burn-up, which differ mainly by the extent of the temperature coefficient.

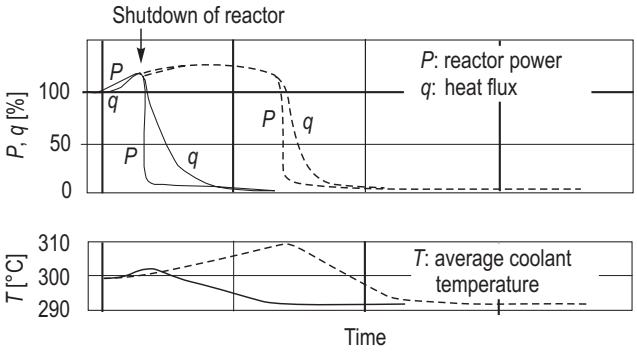


Fig. 1.88. Time dependence of some reactor parameters (PWR) in case of failure of the control system. The change of reactivity is $d\rho/dt \approx 10^{-4}$/s. The temperature coefficient of the coolant is $\Gamma_c = 0$ for a fresh core (solid line) and $\Gamma_c = -3 \times 10^{-4}$ °C^{-1} for a burned-out core (dashed line).

The shutdown of the chain reaction takes place by the fast shutdown procedure, mainly caused by too small a value of the safety factor against boiling. The fast shut-off of the turbine and the injection of steam into the condenser are important measures to limit the temperatures in the primary coolant.

In principle there are possible accidents concerning changes of the boron content in the coolant. The boron content is necessary to compensate for the progressing burn-up of fuel and the changes in xenon concentrations. If non-borated water is injected into the primary circuit, reactivity changes are initiated. If the velocity of changes by this effect is below that of changes by the control rods, the accident can be governed by intrusion of the control rods.

The ingress of cold water into the reactor core also causes a reactivity gain because of the negative temperature coefficient. This case is important only for reactors in which cooling loops can be separately shut off.

As a major accident in case of PWRs the sudden loss of a very effective control rod is considered. It is assumed that a reactivity gain of 0.3 % occurs (see Fig. 1.89). The time dependence of the power and the relevant temperatures shows that after fast shutdown of the reactor no impermissibly high temperatures would occur. However, the fast shutdown must be actuated.

1.2.5.7 External events

External events due to natural or anthropogenic reasons have to be considered in safety analysis. Figure 1.90 contains some events which have to be taken into account in the licensing procedure (German conditions).

The functions of shutting down the reactor, removing the decay heat and retaining the fission products inside the plant have to be guaranteed under all circumstances in case of these accidents. Main events are airplane crash, gas cloud explosion and earthquakes. For the airplane crash the following assumption is made: a Phantom military aircraft with a mass of 20 t and a velocity of 215 m/s strikes the reactor building; an area of about 7 m^2 is affected, the time-dependent force for this accident is shown in Fig. 1.91a. The necessary wall thickness of the reactor building dependent on the strength of the concrete is displayed in Fig. 1.91b. It is required that no penetration of the concrete occurs.

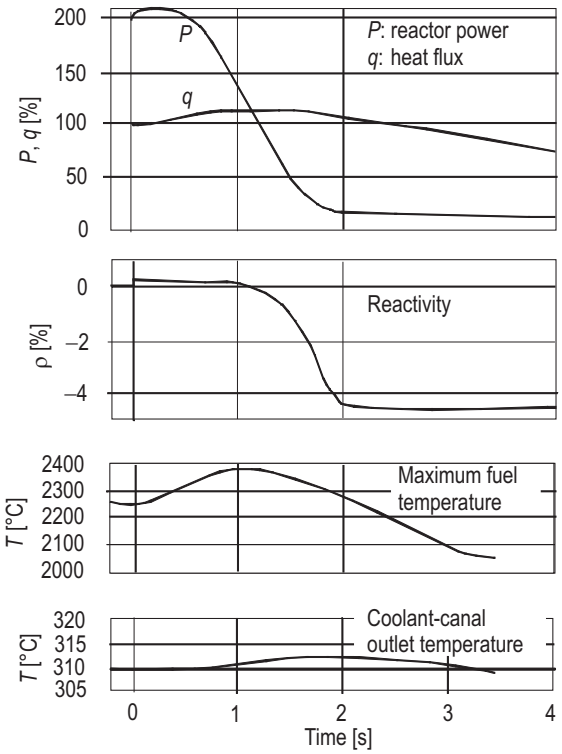


Fig. 1.89. Time dependence of reactor parameters (PWR) after sudden loss of one very efficient control rod ($\Delta\rho = 0.3$ %).

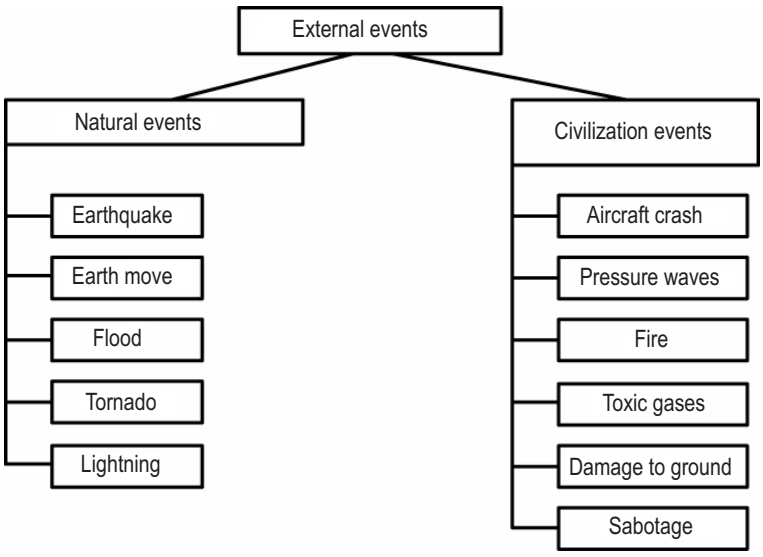


Fig. 1.90. External events to be considered in the licensing procedure (German conditions).

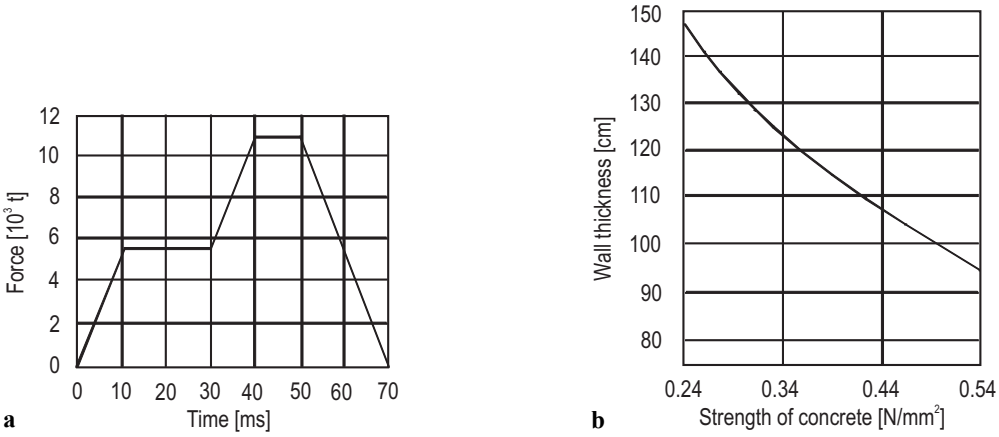


Fig. 1.91. Aircraft crash on a reactor building: **(a)** force acting on the containment wall versus time, **(b)** wall thickness of the containment required to avoid penetration dependening on the quality of the concrete.

In today's modern nuclear power plants in Germany the reactor containment, the building for the diesel generators and for water supply, as well as the storage buildings for radioactive waste are protected against this intrusion (see Fig. 1.92). For older plants an airplane crash was considered to belong to the so-called residual risk, but today it is included in the licensing process. All new nuclear power plants in Germany would have to fulfill this specific requirement.

Terroristic attacks with large civil airplanes, which have large amounts of gasoline on board, require specific considerations. Underground siting of future plants would offer protection against these very extreme accidents. In particular, if the reactor core cannot melt in case of severe accidents there is a good chance to retain the radioactivity inside the plant. If core-catcher installations (see Sect. 1.4) can be realized in the future, they may also offer a chance to limit fission product release to the environment.

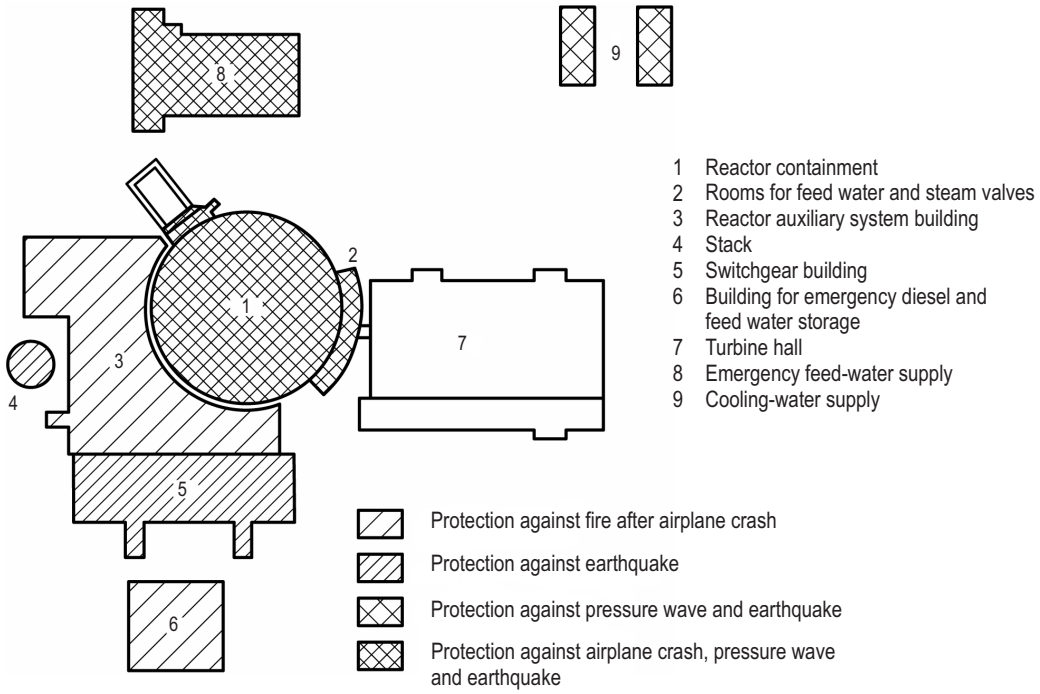


Fig. 1.92. Protection of buildings in a modern PWR (German conditions). (Courtesy of Framatome ANP GmbH).

Gas cloud explosions near the reactor containment cause an overpressure the time dependence of which is displayed in Fig. 1.93a. The wall thickness of modern containments of 2 m is sufficient to take the loads from gas cloud explosions. It is known, however, that specific hydrocarbons can cause detonations with much higher overpressure (more than 10 bar) if the geometrical conditions promote the formation of very high flame velocities. Therefore, some distance between the storage of these hydrocarbons and the reactor is required, depending on the amount of hydrocarbons (see Fig. 1.93b). Especially if nuclear power plants are located on rivers, the transport of liquefied gases by ships on the river could violate this rule.

Earthquakes with a maximum magnitude of 0.3 g for the horizontal acceleration and 0.15 g for the vertical component are included in the licensing procedure for nuclear power plants in Germany today. This corresponds to an intensity of 7 to 8 on the Mercalli scale used to characterize the strength of earthquakes (see Table 1.24). The relevant buildings, the primary system and all systems which are important for shutdown of the reactor, removal of decay heat and retention of fission products inside the plant have to be designed to withstand these loads. The technologies required are available.

Table 1.24. Classification of earthquakes following the modified Mercalli intensity scale.

Intensity	Characterization	Ground acceleration [cm/s^2]	Relation to gravitational acceleration
I	Detected only by sensitive instruments.	1	
II	Felt by few persons at rest, especially on upper floors; delicate suspended objects may swing.	2	
III	Felt noticeably indoors, but not always recognized as a quake; standing autos rock slightly, vibration like passing truck.	4 6	0.005 g
IV	Felt indoors by many, outdoors by a few; at night some awaken; dishes, windows, doors disturbed; motor cars rock noticeably.	8 10	0.01 g
V	Felt by most people; some breakage of dishes, windows and plaster; disturbance of tall objects.	20	
VI	Felt by all; many frightened and run outdoors; falling plaster and chimneys; damage small.	40 60	0.05 g
VII	Everybody runs outdoors; damage to buildings varies, depending on quality of construction; noticed by drivers of autos.	80 100	0.1 g
VIII	Panel walls thrown out of frames; fall of walls, monuments, chimneys; sand and mud ejected; drivers of autos disturbed.	200	
IX	Buildings shifted off foundations, cracked, thrown out of plumb; ground cracked; underground pipes broken.	400 600	0.5 g
X	Most masonry and frame structures destroyed; ground cracked; rails bent; landslides.	800 1000	1 g
XI	New structures remain standing; bridges destroyed; fissures in ground; pipes broken; landslides; rails bent.	2000	
XII	Damage total; waves seen on ground surface; lines of sight and level distorted; objects thrown up into air.	4000 6000	5 g

In the course of the safety analysis the functionality of the enclosure of the primary circuit, of the shutdown system, the decay-heat removal system and the containment are proven. In Germany two types of earthquakes are presently discussed:

- the earthquake relevant for the design: even after several times of occurrence further operation should be possible; an acceleration value of ≈ 0.1 g is usually assumed;
- the earthquake of safety relevance: even after several times of occurrence the function of the safety-relevant systems is maintained; an acceleration value of around 0.2 g is assumed.

The calculations regarding the structural stability include special assumptions about spectra, frequencies and damping. Especially supports and coolant pipes require attention. The probability of damage is correlated with the acceleration (see Fig. 1.94).

For each site a probability exists that the strength of an earthquake is larger than the value assumed in the safety analysis. Figure 1.94 shows these estimations for a specific site in Germany. In case of these larger accelerations damage can occur.

In many countries of the world earthquakes can cause much larger acceleration values. This underlines the necessity to design future nuclear reactors which do not depend on active safety systems.

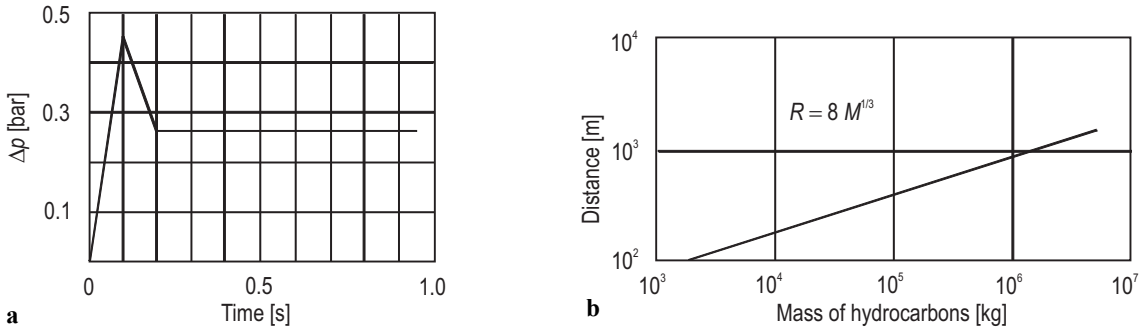


Fig. 1.93. Assumptions for gas cloud explosions near the reactor containment in Germany: **(a)** overpressure Δp dependent on time for the case of deflagration, **(b)** distance to the reactor containment versus allowed amount of liquid hydrocarbons.

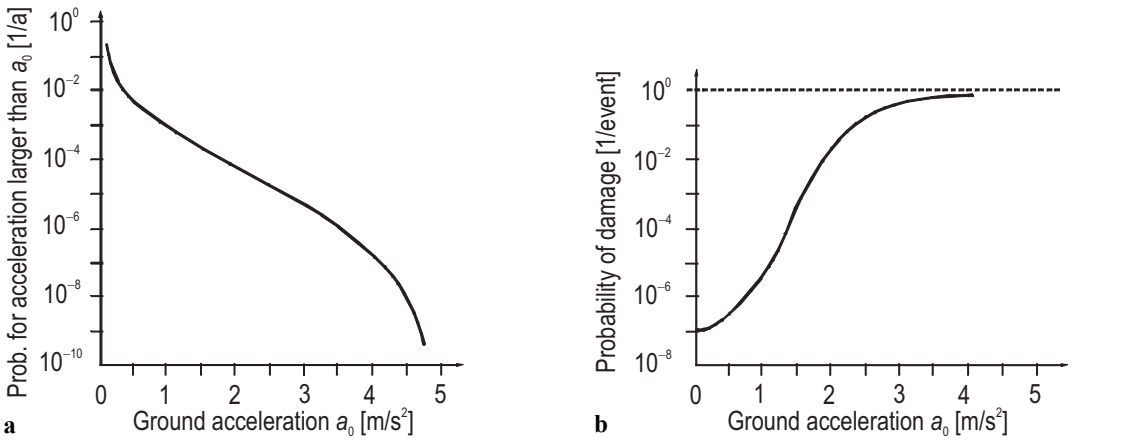


Fig. 1.94. Aspects of extreme earthquakes (Biblis site in Germany): **(a)** probability for the occurrence of an acceleration value larger than a_0, **(b)** probability of damage to structures depending on ground acceleration.

1.2.6 Other types of nuclear reactors

1.2.6.1 Boiling water reactors

In the boiling water reactor (BWR) [00GKN, 86Boh, 69Sau, 92Led, 88Ton] steam generation takes place inside the core with typical values of 285 °C and 70 bar. Figure 1.95 shows a schematic flow sheet of this type of nuclear power plant, and Table 1.25 contains some important data of a large BWR.

In contrast to the PWR, the BWR does not feature a primary and secondary circuit separated by a steam generator. Owing to this direct cycle, the steam turbine and all components of the steam cycle are part of the nuclear system. The efficiency of these plants is similar to that of PWRs. One important difference is that the average power density in the core is half of that in a PWR.

The reactor pressure vessel and the relevant internal structures of a modern BWR are depicted in Fig. 1.96. The fuel elements are arranged in the lower section of the vessel. A steam separator is arranged above the core. The control rods are positioned at the bottom of the reactor pressure vessel and are moved upwards into the core. Inlet nozzles for feed water and outlet nozzles for saturated steam are situated in the upper part of the reactor pressure vessel.

The coolant flows upwards form the bottom of core and is evaporated in the upper part of the core. A sparger ring distributes the feed water, which cools the shroud surrounding the core. After passing the steam separator system saturated steam with less than 1 % humidity leaves the reactor pressure vessel and is guided directly to the high-pressure steam turbine. A part of the live steam is used for reheating the steam leaving the high-pressure turbine.

The fuel elements of a BWR contain fuel rods similar to those in a PWR. However, the dimension of the oxidic fuel pellets is larger and the fuel rods are arranged inside a metallic box (Fig. 1.97).

For the control of the reactor rods with cruciform cross section are used, which are injected from the bottom of the reactor into the core. Pipes for feed water and live steam connect the reactor directly to the turbine and the steam/water cycle. Because the steam is not totally free from contamination and because of the direct coupling of reactor and turbine the turbine hall is part of the nuclear island. For the removal of decay heat there are similar installations as explained for the PWR.

Specific for BWR plants is the condensation chamber inside the reactor containment, which acts for the condensation of the steam flowing out from the coolant circuit in case of loss-of-coolant accidents. This reactor type requires yearly loading and deloading of fresh and spent fuel elements, too. Figure 1.98 contains some technical details of a specific BWR (Germany). Concepts in other countries differ from this type, especially with respect to the form of the containment.

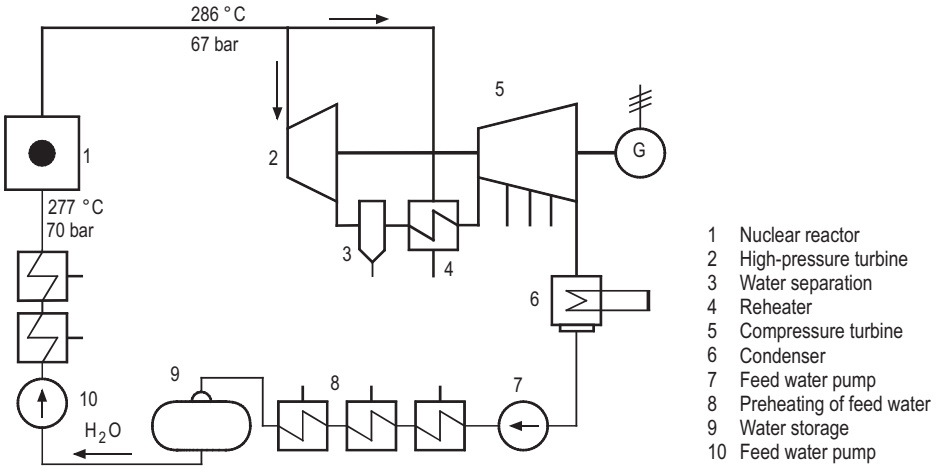


Fig. 1.95. Schematic flow sheet of a BWR.

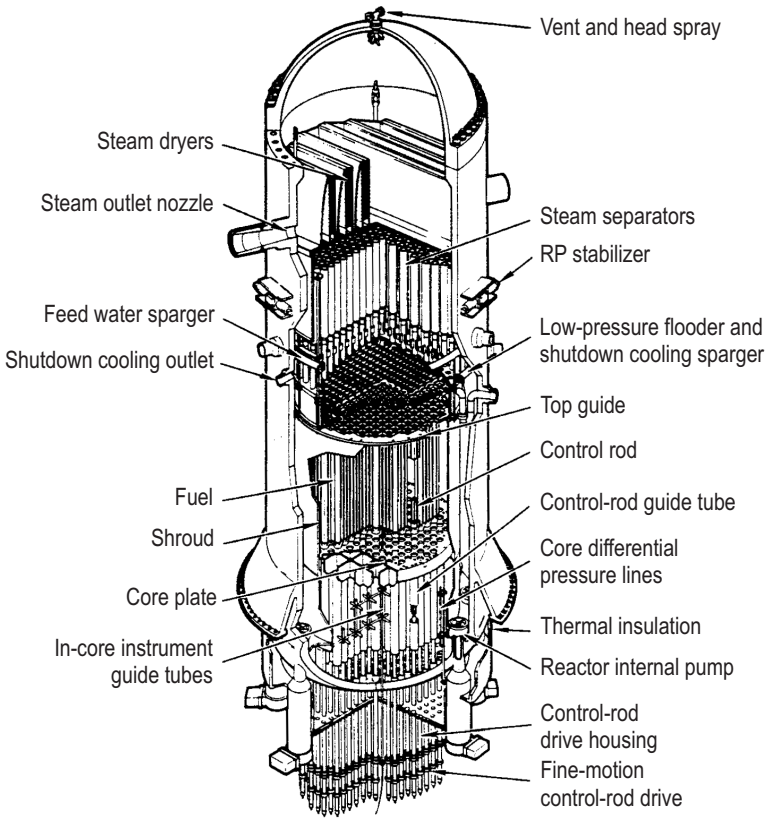


Fig. 1.96. Primary system of a modern boiling water reactor.

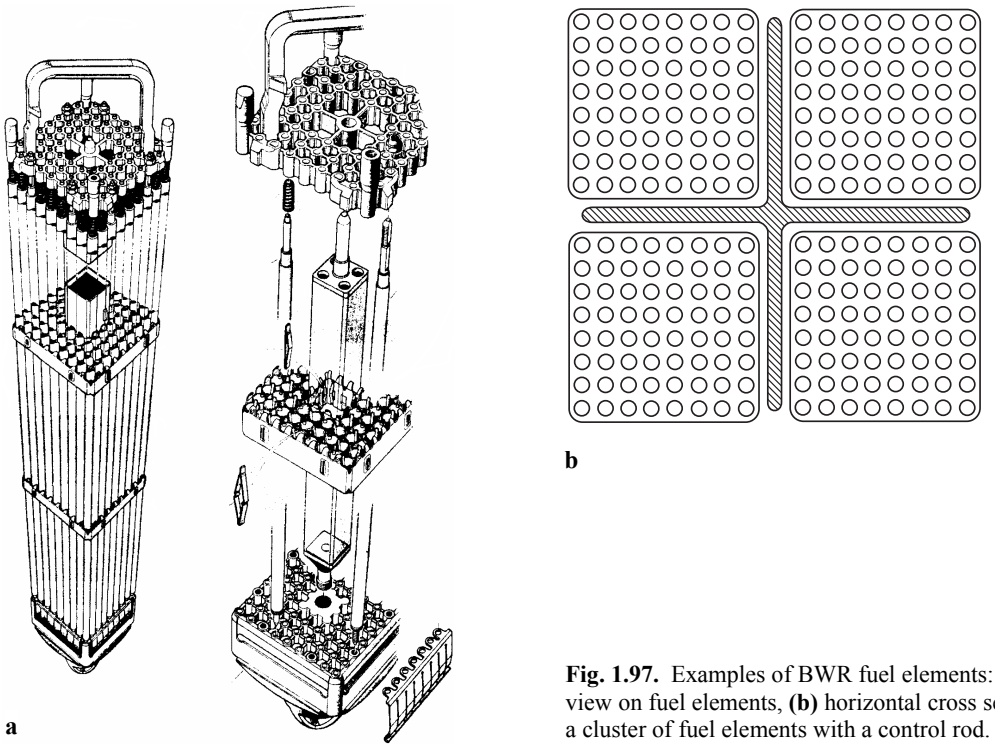


Fig. 1.97. Examples of BWR fuel elements: **(a)** isometric view on fuel elements, **(b)** horizontal cross section through a cluster of fuel elements with a control rod.

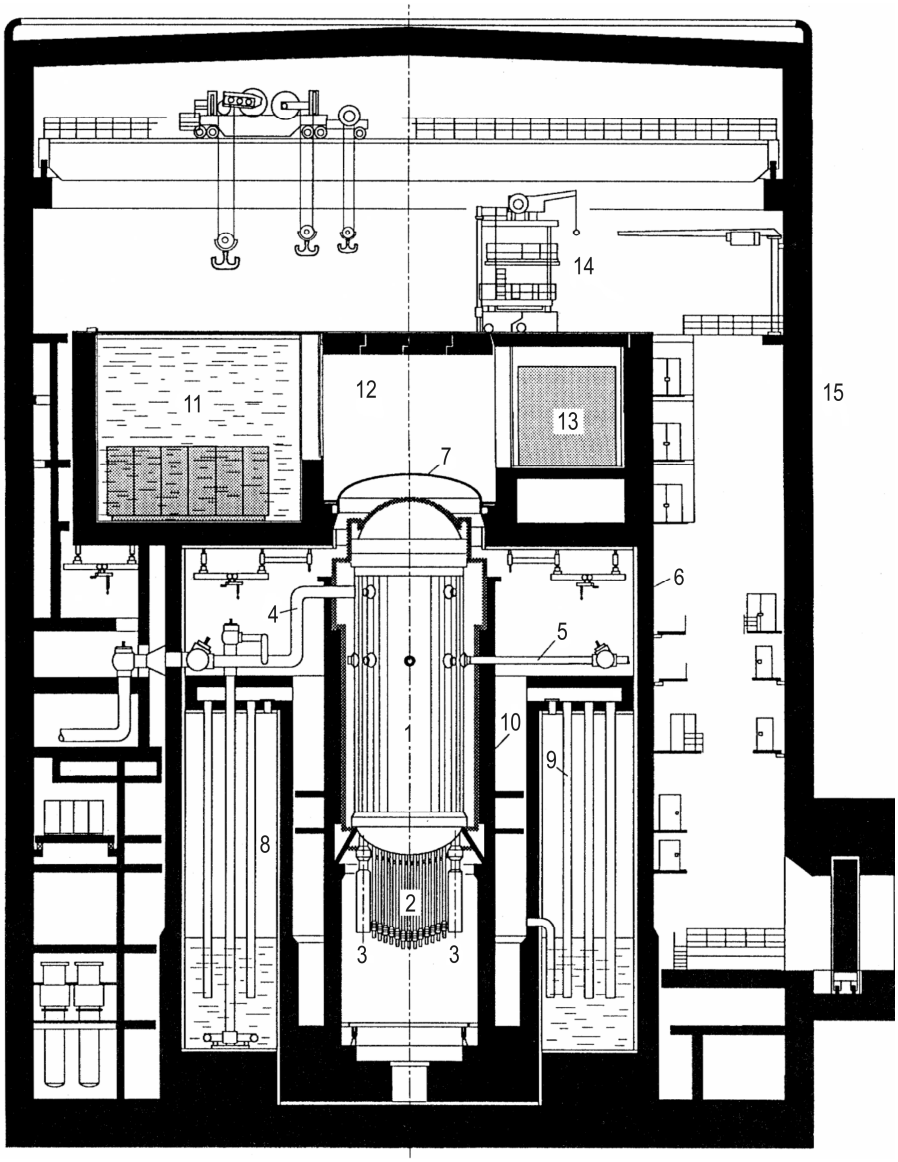


Fig. 1.98. Cross-section drawing of a German BWR, 1: reactor pressure vessel, 2: drives of control rods, 3: main coolant pumps, 4: inner pipes for live steam, 5: pipes for feed water, 6: reactor containment, 7: loading closure, 8: condensation chamber, 9: condensation pipes, 10: biological shield, 11: fuel storage, 12: flooding room, 13: steam separator room, 14: loading machine, 15: reactor building. (Courtesy of Framatome ANP GmbH).

Table 1.25. Important parameters of a modern BWR plant.

Thermal power of the core	3840 MW	Rod diameter	10.75 mm
Number of fuel elements	784	Weight of a fuel element (UO_2)	173 kg
Geometry of fuel elements (positions of absorber fingers)	9 × 9 (1)	Average linear power of rod	159 W/cm
		Average core power density	56.8 MW/m^3
Active length of fuel rod	3710 mm	Fuel enrichment	3.14 %
Core diameter	≈ 6620 mm	Average burn-up	34000 MW d/t

1.2.6.2 CANDU reactors

Using heavy water both as moderator and as cooling medium is the main characteristic of CANDU reactors [75Mor, 87Can, 77EPR]. The fuel is contained in sealed tubes cooled by the cooling medium. The tubes are horizontally installed in a tank full of moderator known as "calandria" (see Fig. 1.99a).

Each of the 12 fuel bundles (Fig. 1.99b) is arranged in a channel and can be replaced without shutdown. The use of heavy water as moderator causes practically no parasitic absorption of neutrons, therefore natural uranium can be used as fuel, no enrichment is necessary. Nevertheless modern designs are on the basis of utilizing very low enriched uranium.

The heavy water heated in the sealed tubes is fed to a boiler, just like in the case of a PWR, using U-shaped tubes. After and being cooled down it is pumped back to the sealed tubes (Fig. 1.99a). In the boiler saturated steam (255 °C, 43 bar) is generated which results in a total plant efficiency of 32 %. The process flow diagram of the secondary cycle and the way it is operated is similar to that of a PWR.

In order to control the whole system, shutdown rods are used which can be inserted, if necessary, through the calandria container (operating at normal pressure) among the sealed tubes (from top downwards). A quick shutdown is also possible by fast drainage of the moderator. Figure 1.100 includes the arrangement of the reactor in more detail.

CANDU reactors have been designed, installed and commissioned in a large number especially in Canada, where they have been operating successfully for several decades.

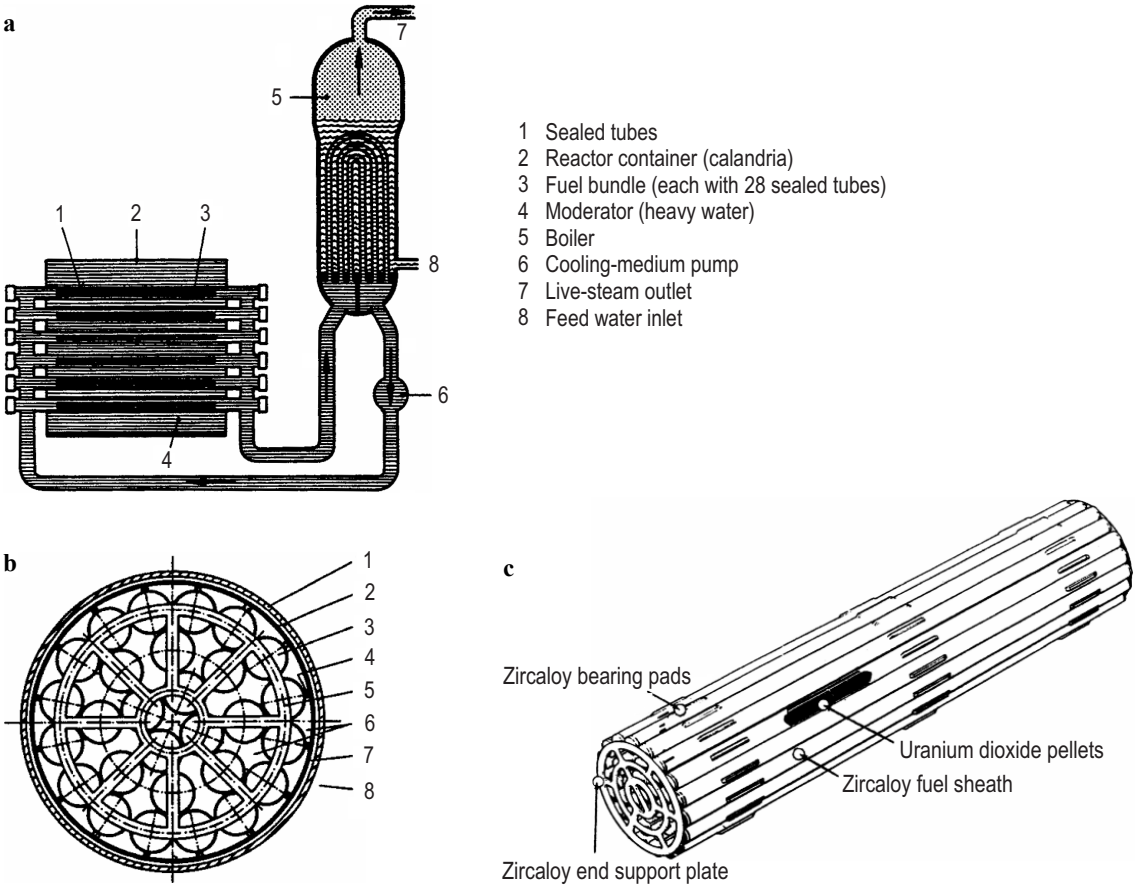


Fig. 1.99. CANDU reactor: **(a)** primary cycle, **(b)** cross section of a sealed tube (1: calandria tube of the reactor container, 2: sealed tube, 3: fuel rod with Zircaloy-4 cover tube and UO_2 pellets, 4: spacer, 5: Zircaloy supporting structure, 6: cooling-medium channel, 7: gas-filled space, 8: heavy water in calandria tank), **(c)** sealed tube.

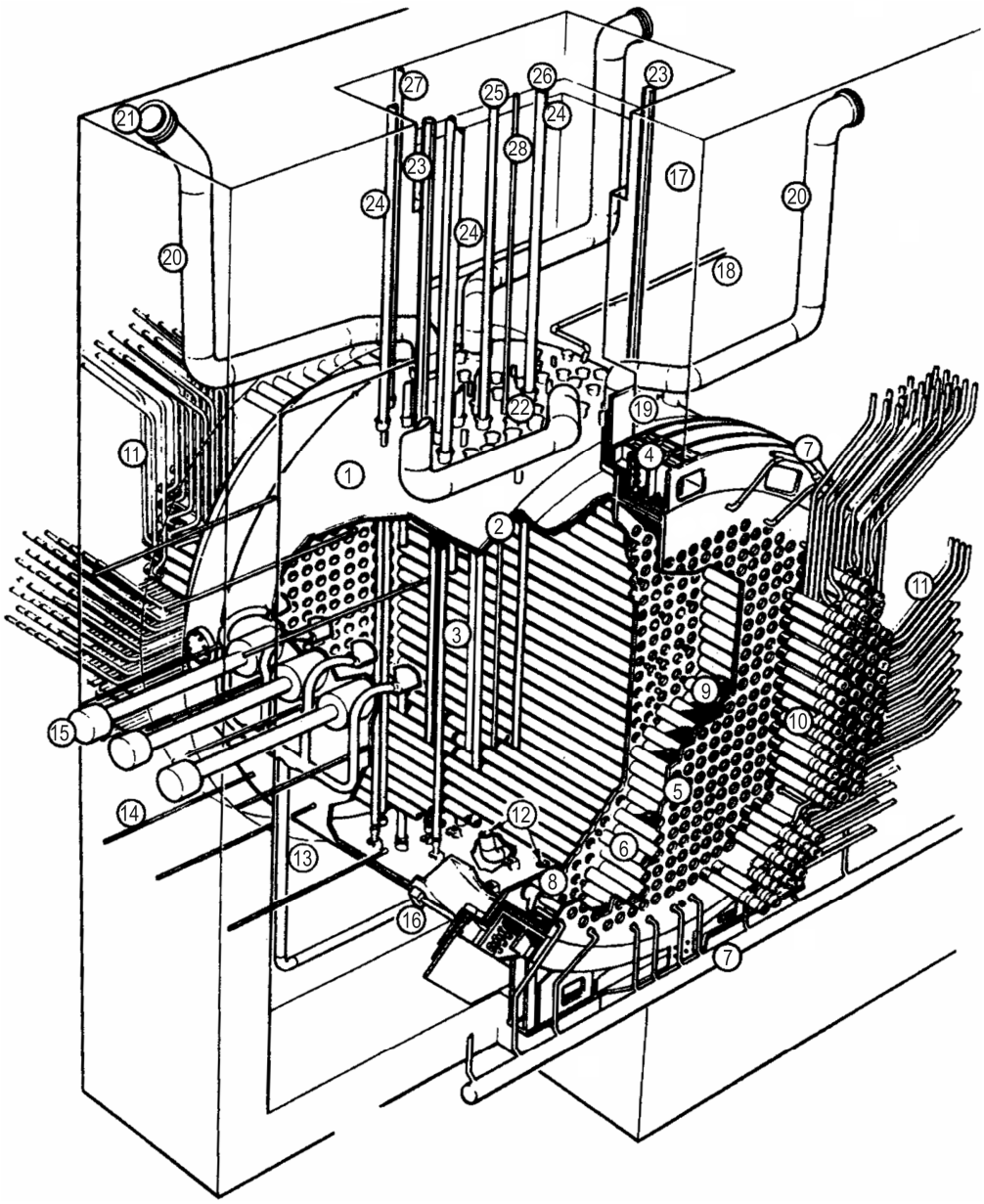


Fig. 1.100. CANDU reactor: 1: calandria, 2: calandria – side tubesheet, 3: calandria tubes, 4: embedment ring, 5: fueling machine – side tubesheet, 6: end-shield lattice tubes, 7: end-shield cooling pipes, 8: inlet-outlet strainer, 9: steel ball shielding, 10: end fittings, 11: feeder pipes, 12: moderator outlet, 13: moderator inlet, 14: horizontal flux detector unit, 15: ion chamber, 16: earthquake restraint, 17: calandria vault wall, 18: moderator expansion to head tank, 19: curtain shielding slabs, 20: pressure relief pipes, 21: rupture disc, 22: reactivity control unit nozzles, 23: viewing port, 24: shut-off unit, 25: djuster unit, 26: control absorber, 27: zone control unit, 28: vertical flux detector unit. (Courtesy of AECL, Canada).

1.2.6.3 RBMK reactors

RBMK reactors were designed and installed in the former USSR. They are classified as boiling-water pressure-tubes reactors using graphite as moderator [91Ull, 86IAE]. The core of such a reactor is made of a large cylindrical block of graphite bricks, with vertical holes drilled for the pressure tubes, as well as for control and shutdown rods. An RBMK reactor with an electrical power of 1000 MW contains about 1700 pressure tubes. Every single tube is separately cooled by water passing through it. There are two separate fuel elements inside a tube (see Fig. 1.101c), each of which is comprised of 18 fuel rods (13.5 mm diameter) containing UO_2 pellets.

The cooling water evaporates to some extent when moving through the tubes and absorbing the heat released from the fuel. The moderation process takes place to a large extent outside of the tubes within the graphite structure. Owing to the presence of (in comparison to graphite) strongly absorbing water, this reactor concept is considered to have a positive temperature coefficient in case of complete loss of the cooling medium or increasing steam quantity.

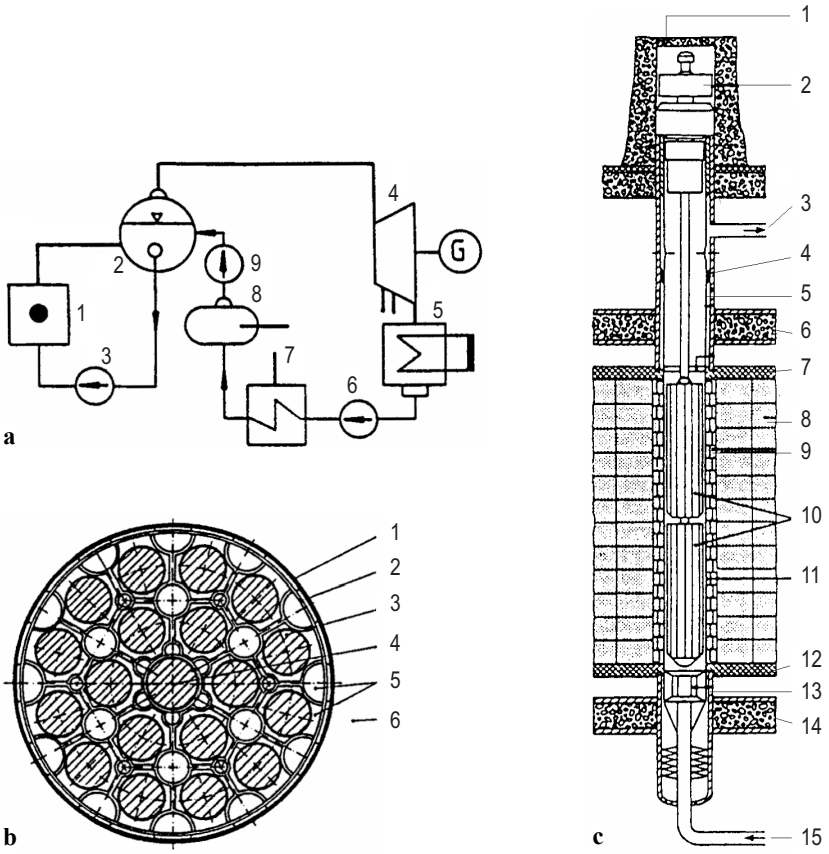
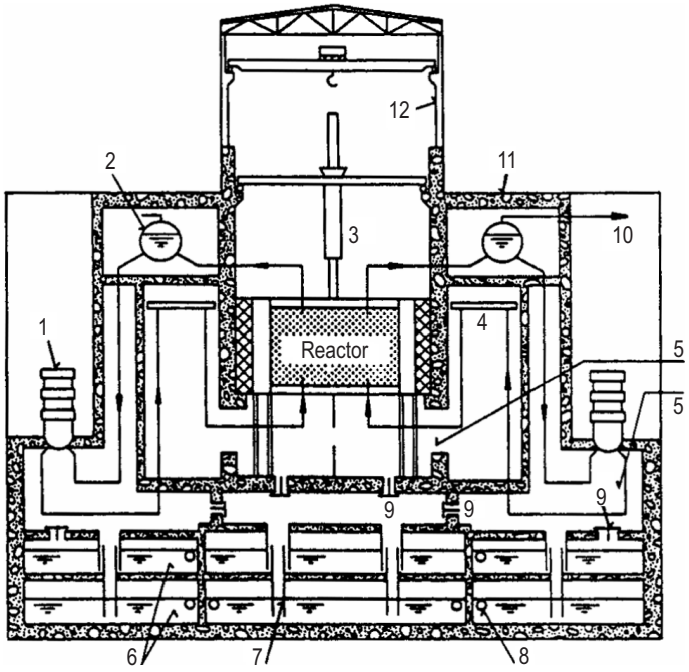


Fig. 1.101. RBMK reactor: **(a)** block diagram (1: reactor, 2: steam-water separator, 3: main cooling-agent pump, 4: saturated-steam turbine, 5: condenser, 6: condensate pump, 7: feed-water preheating line, 8: feed water container, 9: feed water pump), **(b)** cross section through a pressure tube (1: tube, 2: supporting set, 3: fuel rod containing UO_2 pellets and zircon-made cover tube, 4: structure rod (steel), 5: cooling-water channels, 6: graphite (moderator)), **(c)** lateral cross section of a pressure tube (1: sealing block, 2: tube plug, 3: steam-water outlet, 4: guiding tube, 5: sealed tube, 6: upper shielding, 7: thermal shield, 8: graphite block, 9: graphite sleeve, 10: fuel elements, 11: pressure tube (ZrNb), 12: thermal shield, 13: zirconium connecting bush, 14: lower shielding, 15: cooling-water inlet).

The steam-water mixture (285 °C, 70 bar, $\chi = 0.25$) formed in each sealed tube moves upwards, exits the core and is directed to the catchers and from there to the steam-water separators. The steam gathered is fed to a saturated-steam power plant (the technical aspects of which are well known) with an overall efficiency of about 32 %.

Contrary to a boiling water reactor the cooling cycle and power cycle work independently (i.e. as far as the mass flows are concerned). For example the mass flows through the main cooling-agent (water) pumps are 7 times greater than those of saturated steam produced (considering full-load conditions). In order to control the whole system a given number of sealed tubes filled with a certain kind of absorber material can be inserted into the core from the top. Single fuel rods can be replaced without shutdown during operation.

The thermodynamical data and efficiencies are comparable with those of light water reactors. Figure 1.102 shows a typical RBMK reactor building. In case of a leakage the steam flowing out is directed to condensers installed under the reactor where it condenses. Altogether 13 large reactors of this type are in operation in Russia (11) and Lithuania (2). One RBMK unit is under construction in Russia. Following the Chernobyl accident in 1986, any plans for the construction of further RBMK reactors have been frozen. Figure 1.103 shows the present status of the destroyed reactor in Chernobyl. The accident led to radioactive contamination of large areas in Russia, White Russia and in the Ukraine, as depicted in Fig. 1.104.



1 Main cooling-agent pump
2 Steam-water separator
3 Charging machine
4 Feed water manifold
5 Pressurized chamber with steel liner
6 Condensation chamber
7 Condensation pipe
8 Drainage
9 Excess-current opening
10 Live-steam tubing
11 Reactor building
12 Reactor hall with crane

Fig. 1.102. RBMK-1000 reactor building.

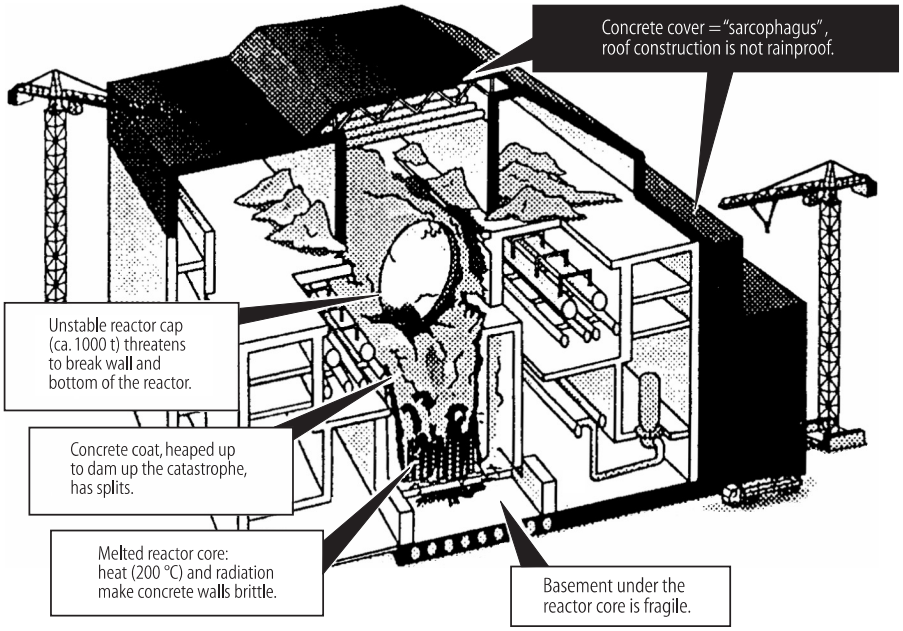


Fig. 1.103. Status of the destroyed RBMK plant in Chernobyl.

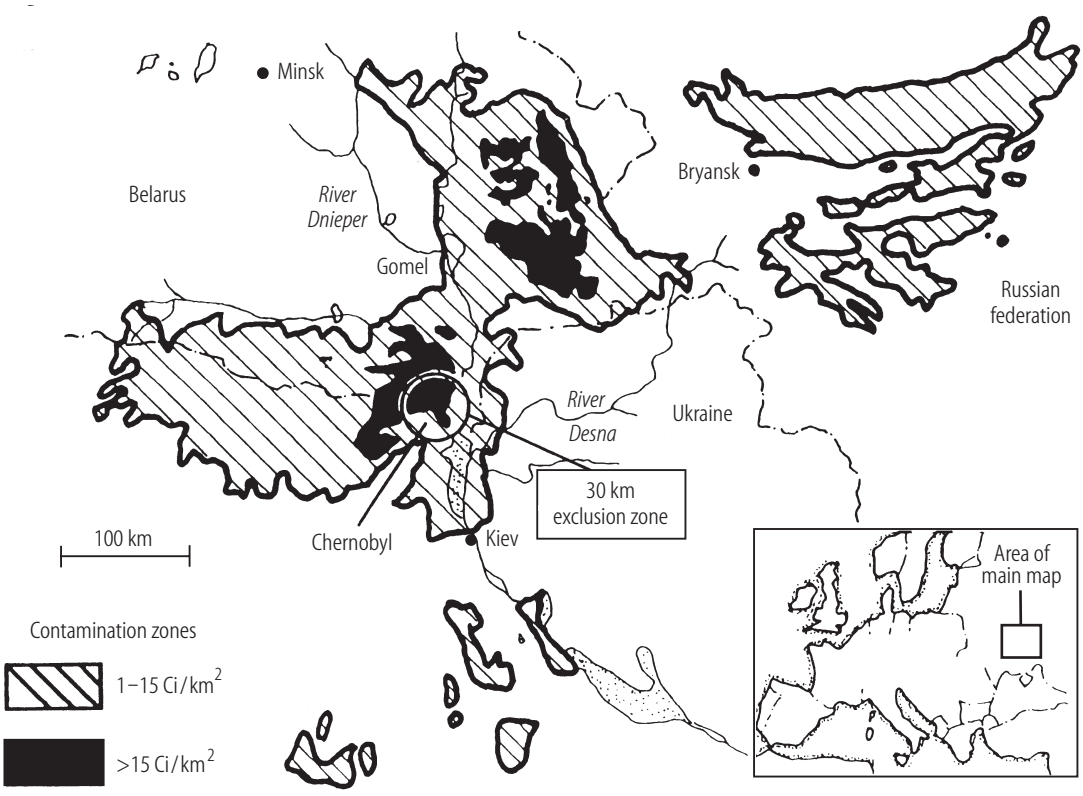


Fig. 1.104. Areas contaminated by the reactor accident in Chernobyl (^{137}Cs, half-life: 30 a).

1.2.6.4 Advanced gas-cooled reactors (AGR)

This type of reactor uses CO_2 as cooling medium and graphite as moderator and construction material [92Mod, 80Den]. Steel is used as canning material just to enclose the fuel elements. Such a combination of materials allows the cooling medium to warm up to about 650 °C and as a result to have conventional live-steam states. Electrical efficiencies of about 40 % can be achieved in this way. This type of reactor has been designed and installed in the UK and has been in operation successfully until now. Figure 1.105 shows the process flow diagram of an AGR power plant. The secondary cycle operates on the basis of a reheat cycle having a live-steam temperature of about 530 °C.

Figure 1.106 shows a fuel element containing low enriched UO_2 pellets (14 mm diameter). The fuel rods are steel tubes. A fuel element is comprised of many fuel rods, each surrounded by graphite pipes. The fuel elements are installed vertically in the reactor core and are cooled by CO_2 gas flowing upwards. The whole primary cycle is enclosed in a pre-stressed concrete container (Fig. 1.107) and the boilers are installed in caverns within the container wall or in the reactor cavern.

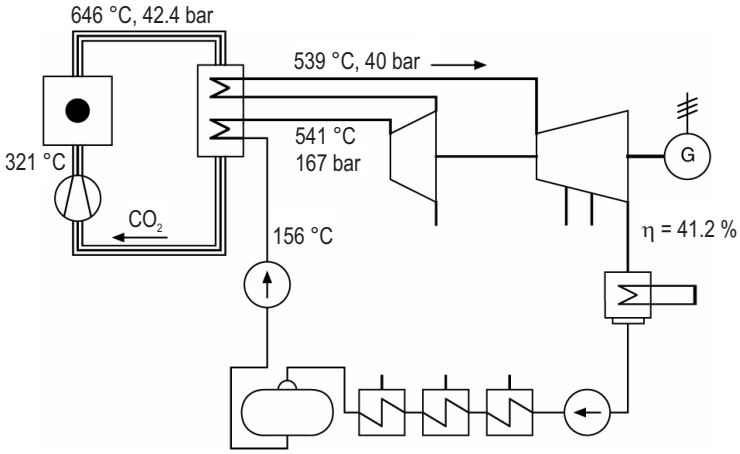


Fig. 1.105. Process flow diagram of an AGR power plant.

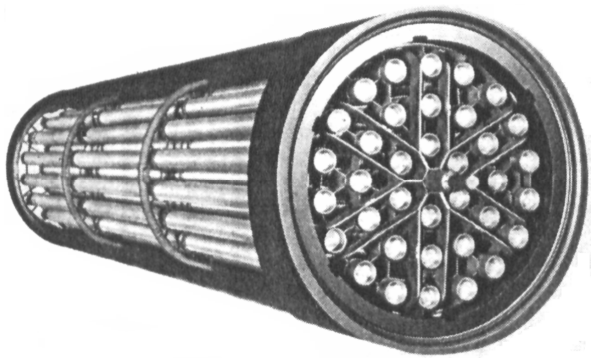


Fig. 1.106. AGR fuel element with fuel rods.

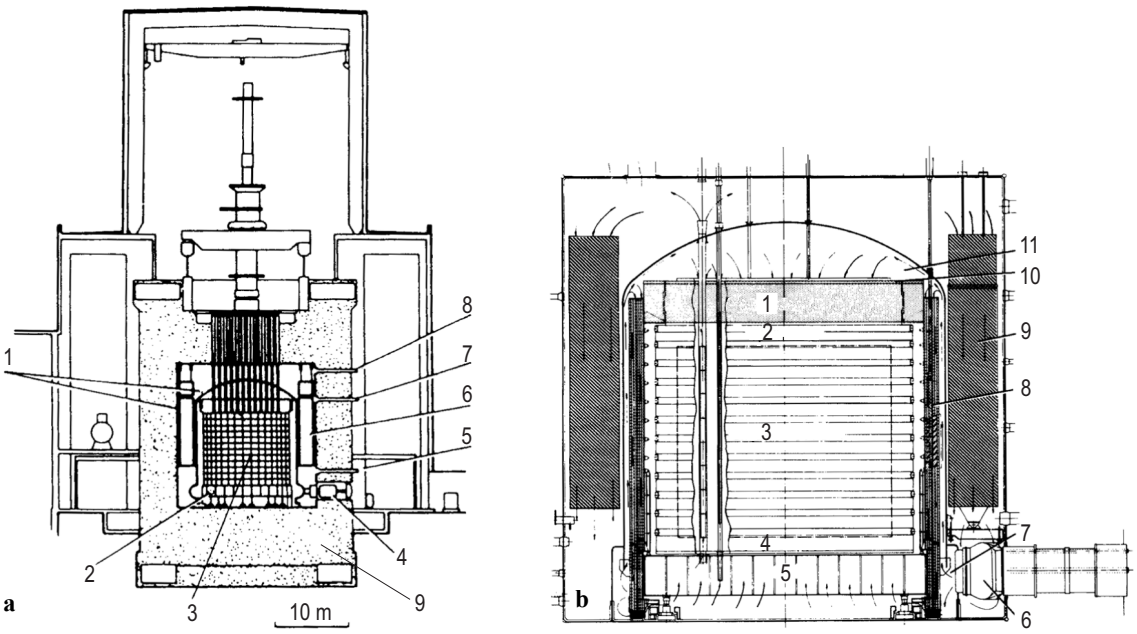


Fig. 1.107. AGR: **(a)** primary system (1: gas-ducting component, 2: supporting structure, 3: core, 4: CO_2 blower, 5: feed water inlet, 6: steam generator, 7: live-stream outlet, 8: reheat-stream outlet, 9: prestressed-concrete reactor pressure vessel), **(b)** reactor core (1: neutron shield (graphite), 2: top reflector, 3: graphite moderator, 4: bottom reflector, 5: diagrid, 6: circulator, 7: outlet gas duct, 8: boiler shield wall, 9: boiler, 10: neutron shield (steel), 11: re-entrant gas flow).

The steam generator pipes have a helical shape, the CO_2 flows outside the tubes. The CO_2 circulators are integrated into the walls of the reactor pressure vessel which consists of concrete. Tightness is provided by an inner steel liner. This component is cooled by water and insulated with the help of metallic foils. The vessel is prestressed in the axial and radial directions by metal cables. Owing to the prestressing, there are only compressive stresses in the walls of the vessel. This type of vessel can never burst because the prestressing system has a very high redundancy. At the time the AGR was designed, it was thought that a containment is not necessary owing to this special type of vessel. Therefore the reactor is placed inside a hall with special measures to guide and filter the cooling gas released in case of depressurization accidents. The deloading of spent fuel elements and the loading of fresh fuel elements is done with the help of a loading machine during normal operation. The availability and the safety behavior of this type of reactor was good during the last decades.

1.2.6.5 High temperature reactors

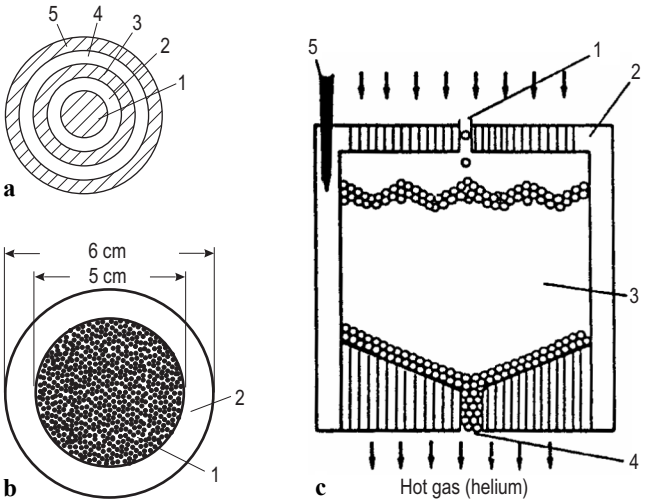
High temperature reactors (HTR) are designed to produce considerably higher temperatures than light water reactors. Thus, they are based on a technical concept which is quite distinct from other reactor types, bearing important consequences with regard to energy gain and safety matters [90AVR, 89Kug, 84Mel, 84KWU, 72Bed]. The nuclear fuel is comprised of a large number of tiny particles embedded in a graphite matrix. Helium is employed as coolant. As for the shape of the fuel elements, two major approaches have been pursued, featuring spherical and block-shaped fuel elements, respectively.

Figure 1.108 shows the working principle of a pebble-bed HTR. The heat source are UO_2 particles (100 µm diameter) which are coated with several layers of pyrolytic graphite and silicon carbide. Using this kind of coated system, excellent control of the fission products can be achieved. The coated particles are embedded in a graphite matrix which in turn is enclosed by a 5 mm thick graphite cover free of fuel. One fuel element contains about 10000 to 30000 finely dispersed coated particles. The fuel elements are spherical with a diameter of about 6 cm. They are arranged in the form of a cylindrical pebble bed con-

fined in a graphite reflector. During operation, pressurized helium flowing through the reactor core either upwards or downwards is used for cooling.

Continuous loading and discharging of the fuel elements is achieved by a fuel circulating system, as indicated in Fig. 1.108b. Inside the core the spheres move downward by gravity (at an approximate velocity of 2 cm/h). Outside the core they are continuously handled and recycled by the fuel handling system (see Fig. 1.109b).

In contrast to other types of reactors, HTRs feature relatively low fuel temperatures but high coolant temperatures (about 700…950 °C or even up to 1050 °C in case of the AVR and THTR). This is due to the very low power density of about 2…6 MW/m^3 and, on the other hand, high thermal conductivity of the coating layers.



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Fuel type	UO_2
Enrichment	≈ 8 %
Diameter of coated particle kernel	0.5 mm
Coating layers	C/SiC/C
Diameter of coated particle	1 mm
Diameter of fuel element	60 mm
Number of coated particles in fuel element	≈ 20000
Av. core power density	3 MW/m^3
Peak factor of core power density	≈ 1.6
Burn-up of fuel	80000…100000 MWd/t

Fig. 1.108. Working principle of an HTR: **(a)** coated particle (1: seed (fuel), 2: buffer layer, 3: inner carbon layer, 4: silicon carbide layer, 5: outer carbon layer), **(b)** fuel element (1: fuel zone, 2: fuel-free zone), **(c)** core structure (1: fuel element input, 2: graphite reflector, 3: pebble bed, 4: fuel element output, 5: absorber rod in reflector), **(d)** some technical facts of the fuel elements and the core.

The reactor core as well as the graphite reflector are enclosed with the cooling cycle (helium cycle) including the blowers, ventilators and heat exchangers. The latter have been constructed as boilers in the existing HTRs. Distributing the cooling gas through the cooling cycle is done to some extent by special constructions in the body of the reactor and partially using special gas pipes.

The so-called MODUL reactor shown in Fig. 1.109 is equipped with a coaxial hot-gas pipeline. Modern HTRs are controlled and, if necessary, shut down by varying the positions of absorbers in reflector places. This reactor type has a strongly negative reactivity coefficient which guarantees inherent shutdown safety in case of a temperature increase. Figure 1.109a shows the layout of a MODUL as well as of a boiler both confined in a pressure vessel. The thermal power of the reactor is 200 MW. Higher power can be achieved by connecting several reactors in parallel. The live steam has a temperature of 530 °C and a pressure of 190 bar. So electricity can be generated using conventional steam power cycles having an overall net efficiency of about 40 % (see Fig. 1.110).

MODUL reactors are not only useful for electricity generation but they can also be employed for heat and electricity combined cycles. With a helium temperature of 950 °C, this reactor could even run a chemical process involving endothermic reactions or be directly connected to a gas power cycle to run a gas turbine. The latter option would raise the overall efficiency of the system to about 48 % (see Figs. 1.111 and 1.112).

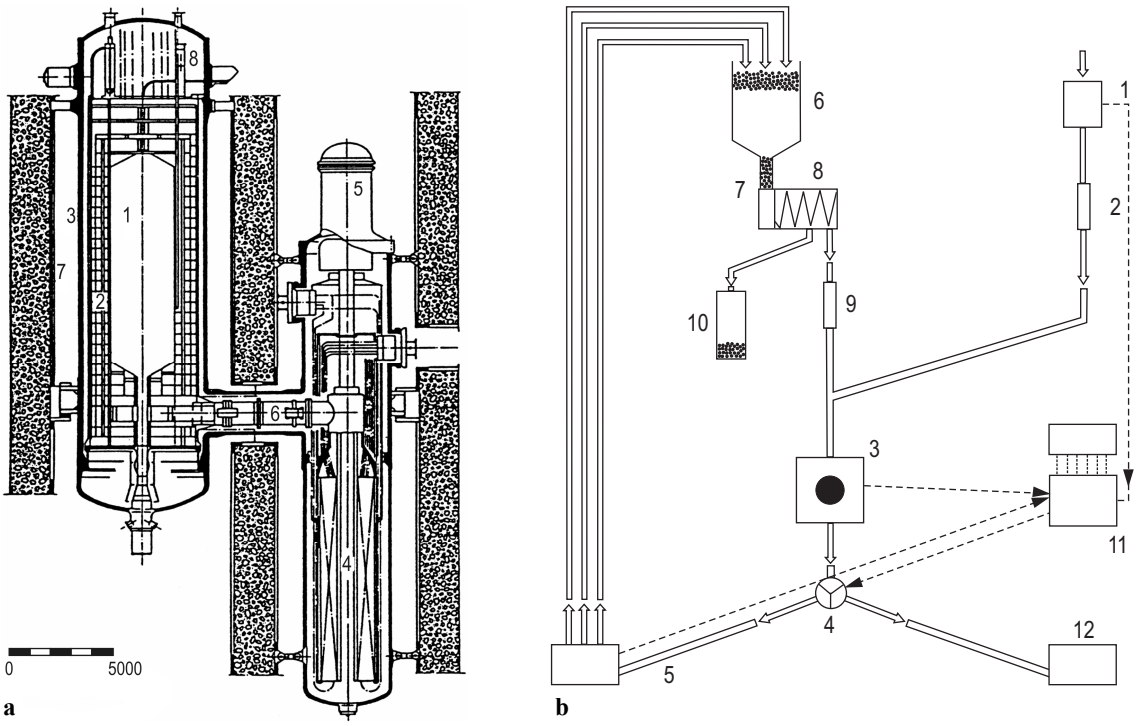


Fig. 1.109. HTR MODUL plant: **(a)** reactor pressure vessel (1: reactor core (bed), 2: side reflector, 3: reactor pressure vessel, 4: boiler, 5: blower, 6: hot-gas pipeline, 7: unit cooling system, 8: absorber rod drive), **(b)** fuel handling system (1: input of fuel elements, 2: buffer, 3: burn-up measurement, 4: switch, 5: elevator system, 6: core, 7: discharge system, 8: scrap separator, 9: storage system, 10: canister for damaged fuel elements, 11: computer, 12: storage for spent fuel elements).

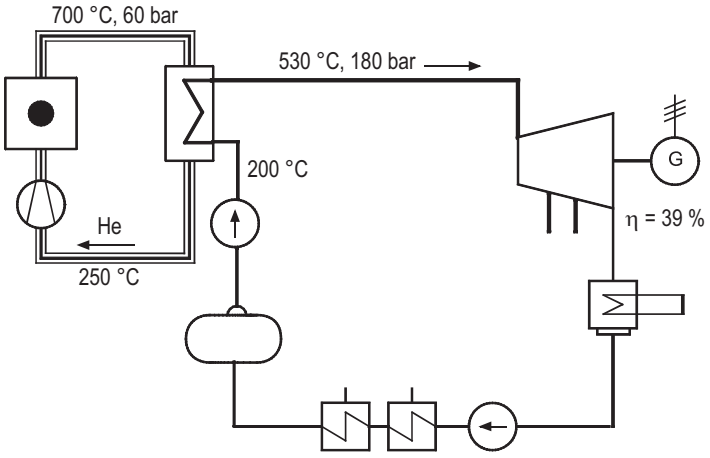


Fig. 1.110. Schematic flow sheet of a steam turbine process with a modular HTR as heat source.

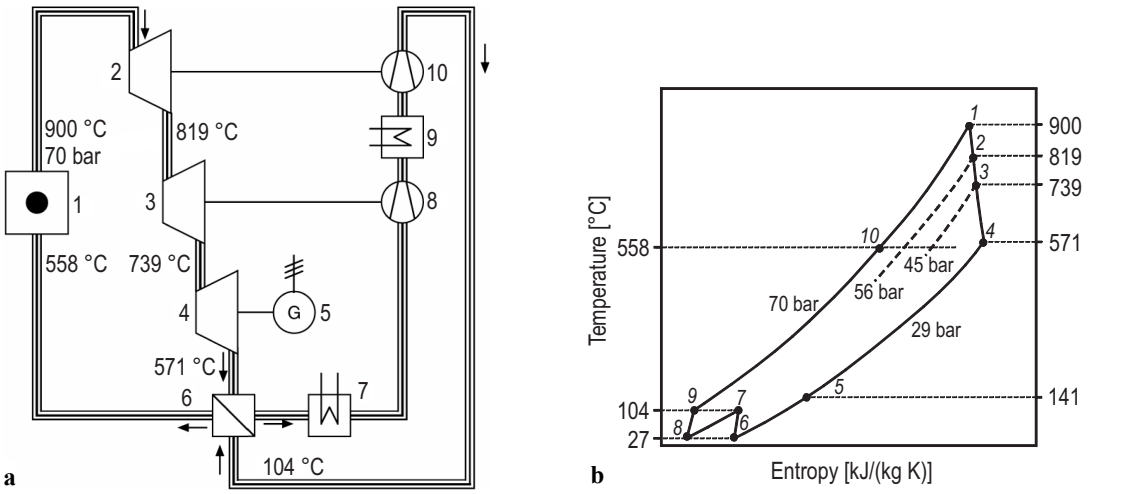


Fig. 1.111. Gas turbine process with a modular HTR as heat source: **(a)** principal flow sheet (1: reactor, 2: high-pressure turbine, 3: medium-pressure turbine, 4: working turbine, 5: generator, 6: recuperator, 7: precooler, 8, 10: compressor, 9: intercooler), **(b)** temperature versus entropy for the respective process.

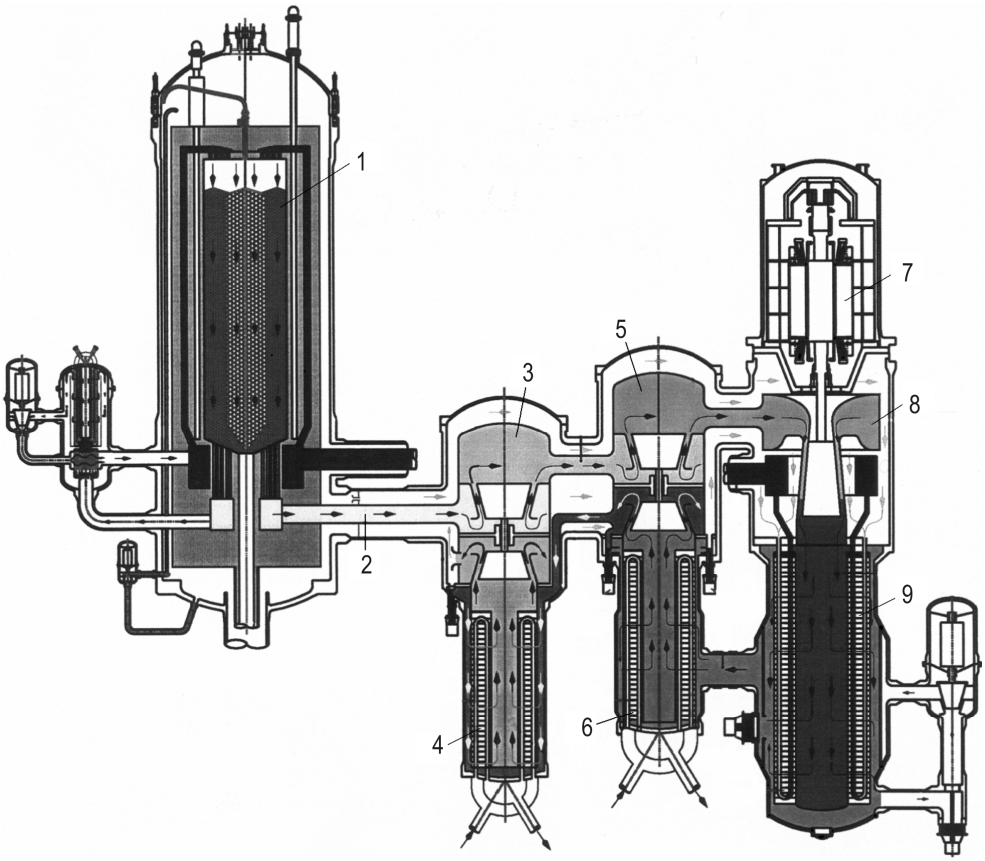


Fig. 1.112. Components of a gas turbine cycle combined with a modular HTR as heat source, 1: pebble bed core, 2: hot gas duct, 3: high pressure turbine + compressor, 4: intercooler, 5: medium-pressure turbine + compressor, 6: precooler, 7: electric generator, 8: working turbine, 9: recuperator. (Courtesy of PBMR, South Africa).

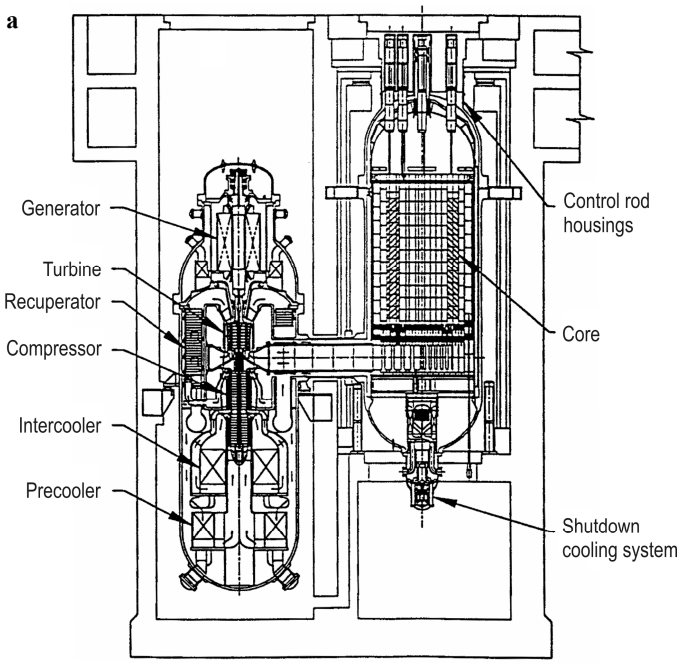
The comparatively low core power density as well as the high amount of graphite inside and around the reactor core lead to an extremely smooth and slow behavior of the reactor in case of any thermal disturbance or serius damage, e.g. due to a failure in the decay-heat removal system.

With a cleverly designed system, it is possible to have a passive, automatically switchable cooling system for decay heat removal through both mechanical and physical heat transport mechanisms like heat conduction or heat radiation. In this way the heating up or even melting down of the reactor can be avoided in case of any breakdown. Details of this matter are dealt with in Sect. 1.4.

In a joint US and Russian project another gas-turbine modular helium reactor (GT-MHR 600 MW (th), 288 MW (th)) is under development for electricity production and for simultaneous generation of electricity and process heat. Additionally it is planned to burn weapon-grade plutonium very effectively. The energy conversion system is based on an intercooled and recuperated Brayton cycle with a one-shaft turbo machine.

The GT-MHR plant consists of one or more identical power units, each module is housed in a reinforced concrete structure, which serves as an independent vented low-pressure confinement. The primary system (Fig. 1.113) is contained within three steel pressure vessels, one for the reactor, one for the power conversion unit and one for the connecting coaxial gas duct system. The reactor vessel contains an annular core consisting of block-type fuel elements, the control system, the refueling access penetrations and a shutdown cooling system at the bottom of the vessel.

a



b

Thermal power per module	600 MW
Electrical power per module	288 MW
Cycle	Brayton
Turbo machine	gas turbine with two compressors on one shaft
Loading of core	discontinuously
Decay heat removal	turbo machinery; auxiliary cooler; conduction, radiation, and natural convection

Fig. 1.113. GT-MHR: **(a)** power module arrangement, and **(b)** technical facts. (Courtesy of General Atomics, USA).

Inside the power conversion vessel a one-shaft turbo machinery and three heat exchangers (precooler, intercooler, recuperator) are arranged. Magnetic bearings are foreseen for the operation of the gas turbine and two compressors on one shaft. The pressurized helium flows from an upper plenum above the core at a temperature of 490 °C downward through the annular core and is heated up to a gas outlet temperature of 850 °C. A coaxial hot-gas duct serves to transport the hot helium to the turbine. The helium at a temperature of 490 °C leaving the secondary side of the recuperator is used to hold the total primary enclosure at this operation temperature. In case of total loss of cooling, self-acting decay heat removal takes place as explained above.

The active region of the core (see Fig. 1.114) is built from block-type fuel elements arranged in an annular shape. The center and the outer positions of the core consist of unfueled, solid removable reflector

blocks. The graphite structure of the core is surrounded by a steel core barrel, this barrel is housed inside an uninsulated reactor vessel. The fuel particles are bound together to form fuel rods, which are inserted inside vertical holes in the graphite fuel blocks. The helium flows through adjacent vertical coolant holes in the fuel blocks (see Fig. 1.115).

It is foreseen to operate the plants with a once-through fuel cycle with three years residence time of the fuel elements in the core, one half of the core shall be refueled every 18 months. For this purpose a fuel handling equipment is used which is able to replace the reflector elements, too.

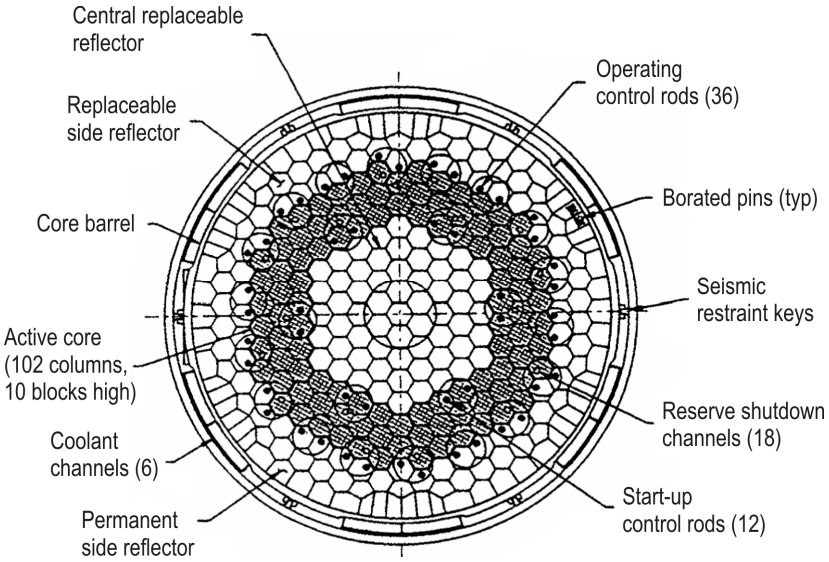


Fig. 1.114. GT-MHR core arrangement. (Courtesy of General Atomics, USA).

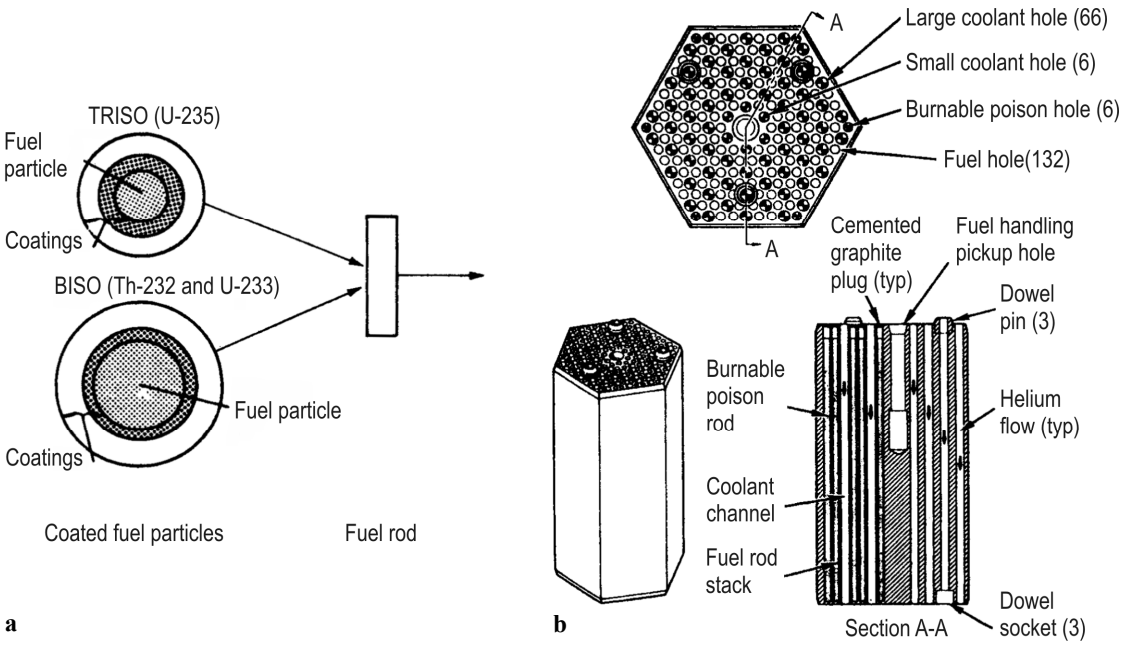


Fig. 1.115. GT-MHR fuel element: **(a)** coated particles (fuel with TRISO, breeding particle with BISO), **(b)** block-type fuel element. (Courtesy of General Atomics, USA).

The reactivity of the core is controlled via control rods which are moved inside the reflectors and the annular core. In addition, onally there is a diversified reactivity control system consisting of boron carbide pellets, which can drop from hoppers above the core into special channels in the core by gravity. A strongly negative temperature coefficient of the core is fundamental to the neutronic safety behavior.

The nuclear fuel is contained in two types of coated particles (see Fig. 1.115). The fissile particles have uranium oxicarbide kernels with nearly 20 % enrichment, the fertile particles contain natural uranium oxicarbide kernels. It is planned to burn plutonium from nuclear weapons as fuel, too. In this case the plutonium would be placed inside the fissile particles. In any case TRISO particles will be used because of their excellent fission product retention in normal operation and in severe accidents.

1.2.6.6 Liquid metal cooled fast breeder reactors (LMFBR)

In all types of thermal reactor systems described above, to a great extent only the energy released from the fission of U-235 is gained and used in further processes. The breeding process takes place just to a limited extent in thermal reactors, according to the following reactions:

$$\begin{aligned} &{}^{238}_{92}\mathrm{U} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{239}_{92}\mathrm{U}^{*} \xrightarrow{\beta^-} {}^{239}_{93}\mathrm{Np} \xrightarrow{\beta^-} {}^{239}_{94}\mathrm{Pu} \ldots, \\ &{}^{232}_{90}\mathrm{Th} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{233}_{90}\mathrm{Th}^{*} \xrightarrow{\beta^-} {}^{233}_{91}\mathrm{Pa} \xrightarrow{\beta^-} {}^{233}_{92}\mathrm{U} \ldots \end{aligned} \qquad (1.250)$$

Pu-239 and Pu-241 as well as U-233 are fissile, too. In thermal reactors such new-born fissile substances are used in situ for further fission reactions, resulting in an overall utilization efficiency for uranium of only about 1 %.

With fast reactors [72Bed, 81Wal, 76Bra, 71Gra, 95Mic] and by using multiple reprocessing stages, a complete conversion of the fertile material is aimed at. In this way, e.g. by continuous usage of the fissile material bred in an optimal fast breeder, uranium can be used about 40 times better than in current light water reactor. This aspect is a great motivation worldwide for employing fast breeders.

The core of a fast breeder consists of two distinct zones (Fig. 1.116). The inner zone contains fuel rods with a mixture of about 80 % UO_2 (natural uranium) and 20 % PuO_2. This zone is the energy-producing zone in which fission takes place. In the outer zone known as the breeding zone the fuel elements contain only UO_2 (U-238). Here new fissile material is produced and is partially used to generate power. In a fast breeder comparatively high concentrations of fissile material must be used, for the fission cross sections in the high-energy zone (fast zone) are much lower than in the thermal zone. Due to this fact the core is very compact and has a very high power density.

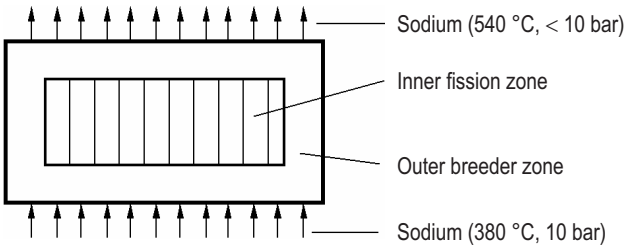


Fig. 1.116. Schematic diagram of a fast breeder core.

Sodium is used as cooling medium in order to keep the moderation of neutrons as low as possible resulting in a neutron spectrum as fast as possible. This is necessary because the η values in the fast-energy zones are very high. Considerations regarding efficient heat removal from the system working under not too high pressures leave no doubt about using sodium as coolant.

As can be seen in Fig. 1.117 in detail, the cooling agent enters from the bottom of the core with about 380 °C and leaves with about 12 bar and 540 °C. The fuel rods having a diameter of about 5 mm and a length of about 0.75 m are situated in a way to form hexagonal fuel bundles each containing 166 fuel rods.

An intermediate heat exchanger is connected to the reactor heating a secondary Na stream to about 520 °C (14 bar). From this point the sodium is pumped back to the reactor through the primary cooling-agent pump. Using the secondary hot Na stream, steam is generated in a boiler with about 500 °C and 167 bar.

Given the highly exothermic reaction between sodium and water, a secondary cooling cycle is necessary to avoid severe damage in case of contact between the two substances. The reaction between sodium and air (oxygen) must be avoided, too. Therefore all pipes containing sodium are installed in chambers filled with inert gas.

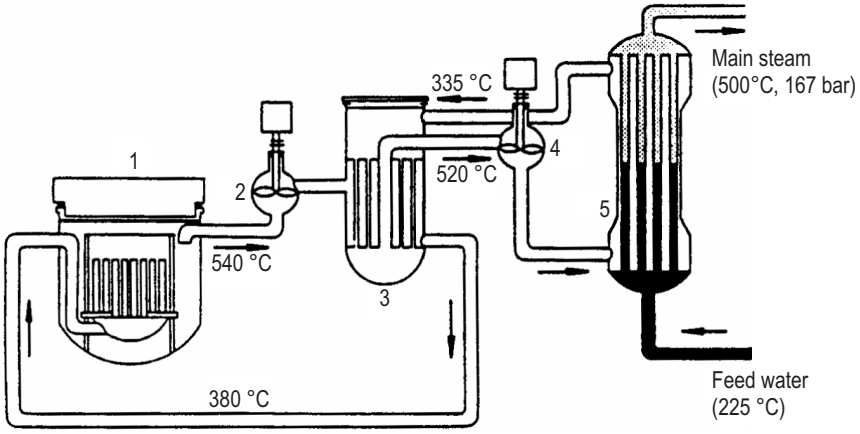


Fig. 1.117. Schematic flow diagram of a sodium-cooled fast reactor, 1: reactor core, 2: cooling-agent pump, 3: Na/Na intermediate cycle, 4: secondary cooling-agent pump, 5: boiler.

During operation the sodium in the primary cycle is exposed to strong radioactive radiation, so that special measures must be taken to protect it from being converted to a radioactive material. Moreover, provisions regarding a preheating system for sodium are required because of its melting point of 98 °C. The structure of a nuclear heat generation system is depicted in Fig. 1.118.

For reactor control, rods can be inserted in the core from the top. If necessary an open-link chain-system can be drawn into the reactor core.

The reactor can be charged only during a shutdown. This can be done at the end of each year within a shutdown period. As long as reactor safety is concerned the same points as in case of light water reactors are valid and must be taken into account. However, other than in light water reactors, in case of vaporization of the cooling agent there is a positive temperature coefficient because of the absorbing nature of the cooling medium (Na). In such a case the occurence of large bubbles could lead to a nuclear blow-up. This phenomenon is known as Bethe-Tait accident. Thus, a meltdown of the reactor core must be prevented through clever and reliable engineering measures. If a meltdown occurs in spite of all provisions and safety measures considered, there should exist a core-catcher system right under the reactor vessel to catch the melted core and cool it down in a safe manner (Fig. 1.118).

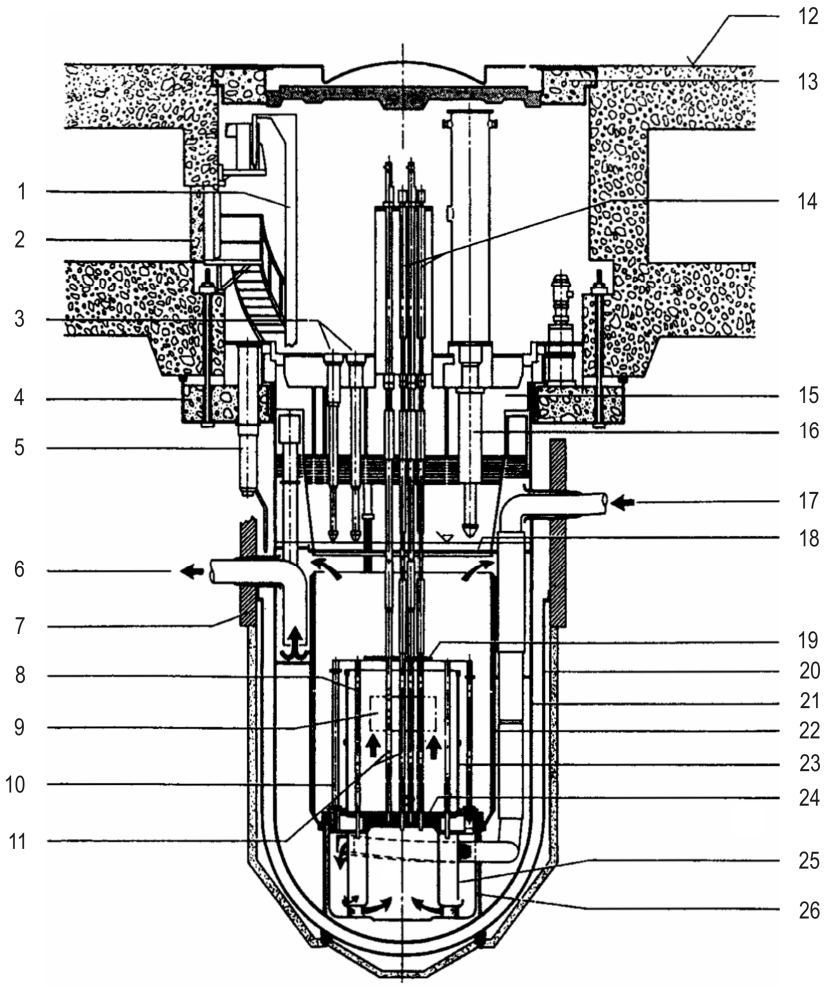


Fig. 1.118. Primary system of the sodium-cooled fast breeder SNR-300, 1: cable transport facilities, 2: access to reactor cover, 3: fuel-element exchange channel (internal), 4: reactor support, 5: inspection man way, 6: sodium outlet, 7: primary shield, 8: fertile material, 9: reactor core, 10: fuel-element conversion position, 11: control rods, 12: platform, 13: hole cover, 14: control rods driver, 15: rotatable reactor cover, 16: fuel-element exchange channel (external), 17: sodium inlet, 18: moving plate, 19: instrument plate, 20: two-layer vessel, 21: reactor vessel, 22: shielding vessel, 23: core mantle, 24: core support, 25: laid construction for fluid flow, 26: accumulator. (Courtesy of Framatome ANP GmbH).

1.2.6.7 New concepts in nuclear technology

The accidents and risks caused by the final storage of radioactive waste have been analyzed worldwide for several years. These studies deal with the disposal of spent light-water reactor fuel elements, with glass blocks loaded with high-level radioactive waste from the reprocessing plants, as well as with spent ceramic fuel elements coming from HTRs.

As an example, Fig. 1.119 shows the Swiss calculations (PSI, 1993) for glass blocks which would be stored in granite. The result is that for the assumption of normal water flow the maximal annual dose received by individuals in case of an accident after some 10^5 years would be lower by a factor of 10^4 compared to the natural radiation dose in Switzerland. Even if it is assumed that the water flow is higher by a factor of 100, the maximal dose remains lower by a factor of 10^2 compared to natural radiation. For direct final storage of LWR fuel elements similar values are expected. These results all lead to the conclusion that final storage is not a serious safety problem compared to the consequences of severe accidents of today's LWRs.

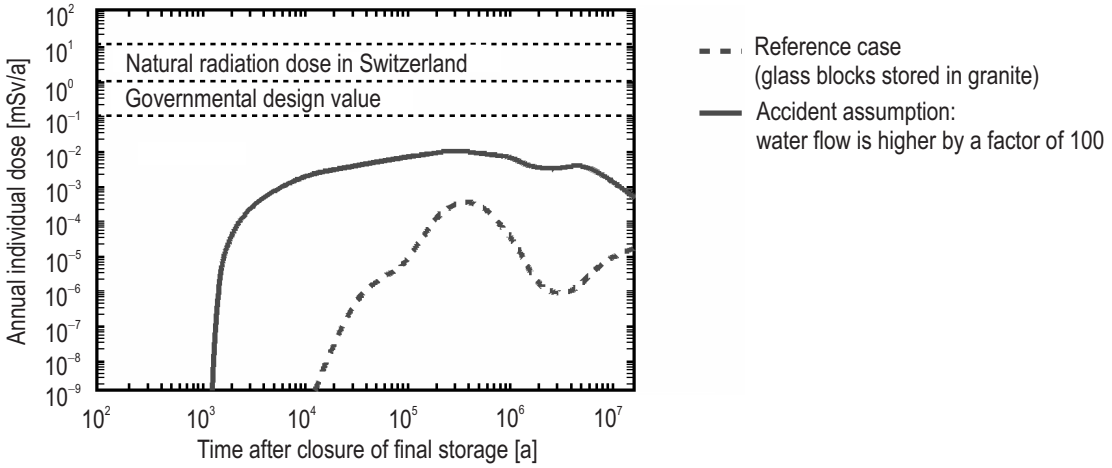


Fig. 1.119. Annual individual dose from glass blocks stored in granite [94PSI].

For ceramic HTR fuel, especially with TRISO-coated particles, it is expected that corrosion will not occur and the retention of fission products is even better compared to glass blocks or LWR fuel. Additional measures like the filling of gaps between the spheres are possible and indicate an even better behavior of spent fuel elements in the final storage.

Even if these results should not be acceptable in the future, there is the possibility of partitioning and transmutation of the long-lived isotopes by accelerator-driven processes [93Rub, 94Bow, 97Liz, 94Rub]. Certainly in this case reprocessing would be necessary. As a result of such transmutation processes the amount of actinides in the final storage would be reduced by a factor of 100 compared to normal reprocessing, as displayed by Fig. 1.120.

Certainly both options shown in Fig. 1.121 have to be compared as far as total risks, accidents, state of technology, doses to operation staff, non-proliferation issues, costs and additional environmental impact are considered. After a very careful comparison of these aspects, transmutation may be a good choice and can help to make the final storage a duty to be done not for millions of years but for about 1000 years.

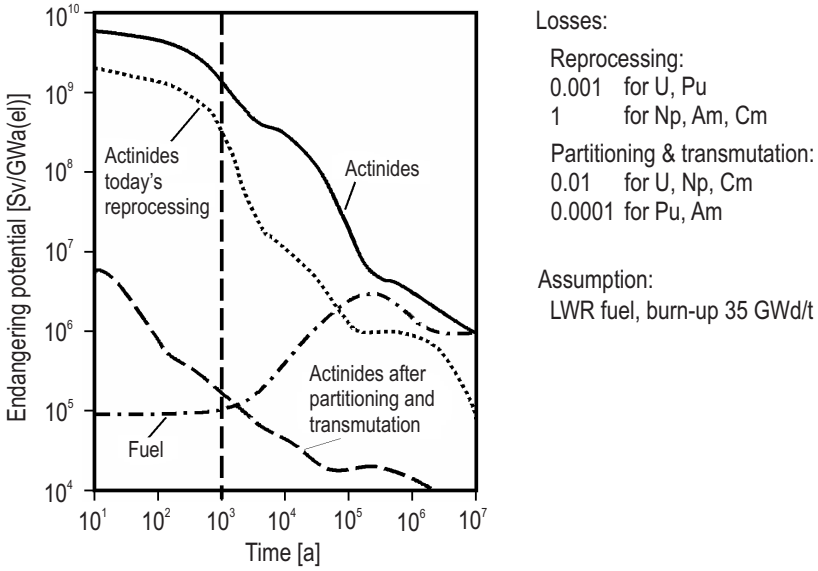


Fig. 1.120. Possible reduction of the toxicity potential by partitioning and transmutation of long-lived actinides.

Concerning technical concepts for transmutation, several proposals have been made and are under development worldwide. One proposal featuring an accelerator to produce high-energy protons and undercritical multiplication of the neutrons produced in spallation processes is presented in Fig. 1.122. Protons are accelerated to an energy of around 1.5 GeV. One beam proton produces nearly 30 neutrons in a spallation target. By undercritical neutron multiplication ($n/n_0 = 1/(1 - k_{\mathrm{eff}})$, $k_{\mathrm{eff}} < 1$) the number of neutrons can be raised by a factor of 5 to 10. Therefore high neutron fluxes can be produced inside the blanket of the spallation system. Isotopes with very long half-life, which are to be removed by this technology disappear according to the relation

$$\frac{\mathrm{d}N}{\mathrm{d}t} = -\lambda \cdot N - \sigma_{\mathrm{a}} \cdot N \cdot \phi,$$
$$N(t) = N_0 \cdot \exp\left(-(\lambda + \sigma_{\mathrm{a}} \cdot \phi) \cdot t\right). \tag{1.251}$$

If the term $\sigma_{\mathrm{a}} \cdot \phi$ is large compared to λ, the "effective" half-life of the isotope,

$$T_{1/2} = \frac{\ln 2}{\lambda + \sigma_{\mathrm{a}} \cdot \phi} \tag{1.252}$$

can be much shorter with the help of this transmutation process.

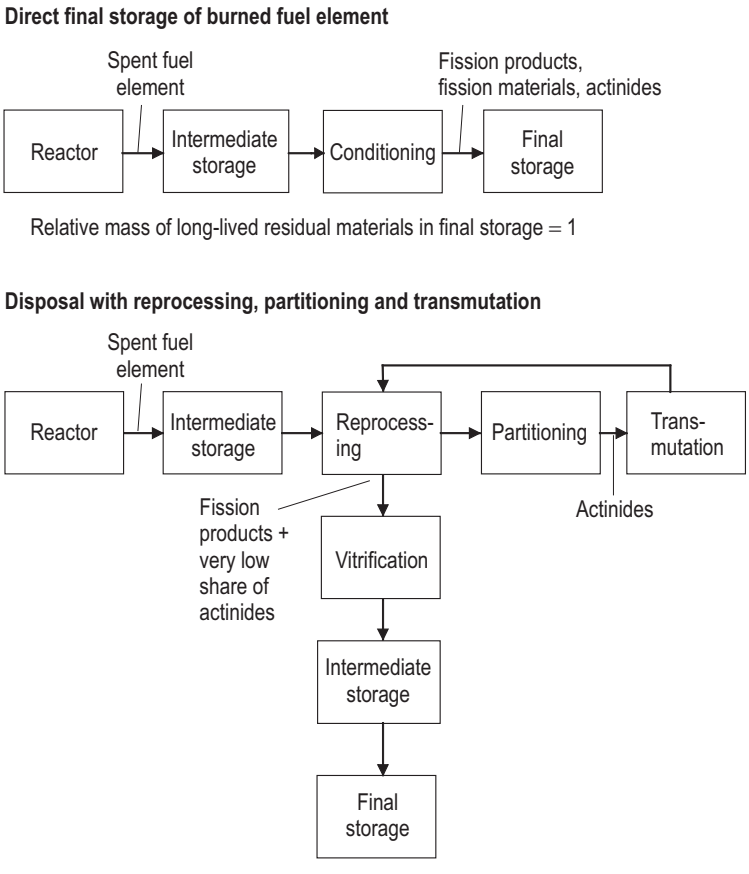


Fig. 1.121. Two options for the final disposal.

The fuel for the process shown in Fig. 1.122 are actinides, and thorium is used as fertile material. Using this process long-lived isotopes can be converted, new fissile material can be produced by breeding and net electrical energy can be produced. Naturally technical questions have to be solved before such a process is feasible: accelerator technology needs progress, the proton window in the spallation target has to tolerate high loads, and fuel elements for the blankets have to be developed.

In principle, however, the process of partitioning and transmutation opens the option to achieve a solution for nuclear technology with a relative short lifetime of the final storage (1000 years). Technical solutions for this process step are available, and proving them in the licensing process is relatively easy.

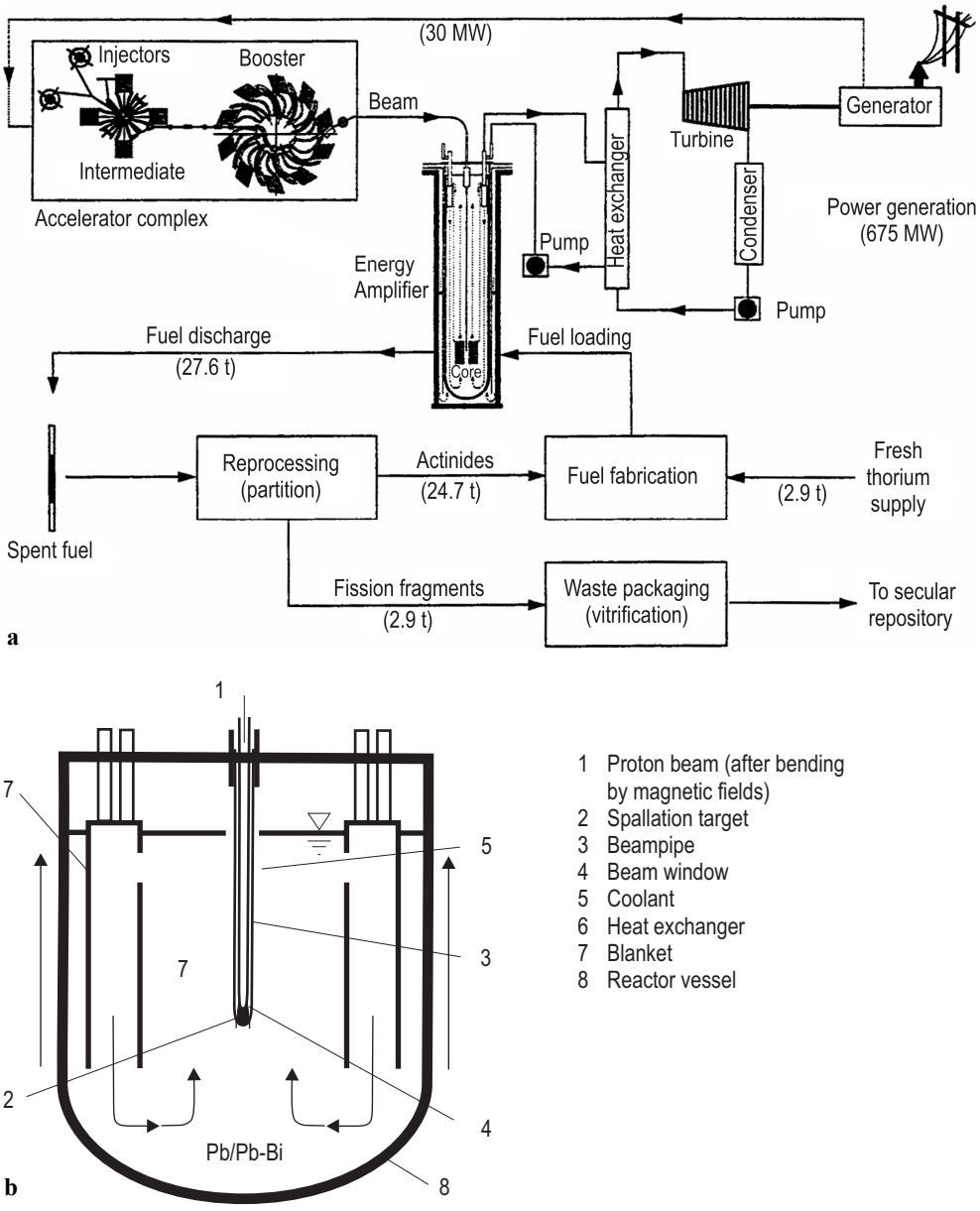


Fig. 1.122. **(a)** Concept for transmutation, **(b)** principle of reactor vessel for a transmutation reactor.

1.3 Economic aspects

1.3.1 Calculation formula for the generating costs of electricity

Different methods are used to analyze the economic conditions of power plant operation [85Wil, 72Mus, 83Han, 93Kug]. A simple one, which is suited to compare different types of power plants shall be applied here. The costs occurring during the working period of a power plant can be first of all divided into power-dependent and work-dependent costs (Fig. 1.123).

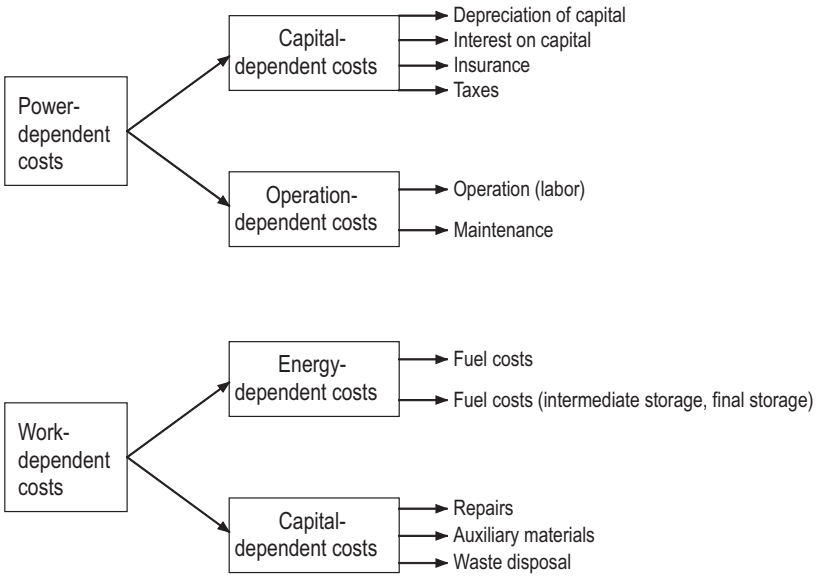


Fig. 1.123. Distribution of costs into power-dependent and work-dependent shares.

A successful and economic operation of the plant can be expected in case that the total amount of returns, R_i, is larger than or at least equal to the total amount of expenses, E_j. This requirement can be applied to an operation time of one year:

$$\sum_i R_i \geq \sum_j E_j \tag{1.253}$$

Starting from this requirement, by integration of all returns and expenses over the period of one year, the following simplified formula results for the electricity generation costs:

$$x = \frac{K_{\text{inv}}}{P_{\text{el}}^0 \cdot T} \cdot \bar{a} + \frac{C \cdot k_C}{P_{\text{el}}^0 \cdot T} + \frac{\dot{m}_{\text{F}}^0 \cdot k_{\text{F}}}{P_{\text{el}}^0 \cdot T} + \sum_i \frac{\dot{m}_{\text{ai}} \cdot k_{\text{ai}}}{P_{\text{el}}^0 \cdot T} + \frac{\dot{m}_{\text{D}}^0 \cdot k_{\text{D}}}{P_{\text{el}}^0 \cdot T} \; . \tag{1.254}$$

In a slightly modified manner (1.254) can be written as:

$$x = x_{\text{C}} + x_{\text{oper}} + x_{\text{F}} + x_{\text{ai}} + x_{\text{D}} \; . \tag{1.255}$$

The parameters in (1.254) and their dimensions are as follows:

x power generating costs [ct/kWh (el)],
P_{el}^0 net electric power of the plant [kW],
K_{inv} overall plant investment costs [$],
T full-load hours per year [h/year],
$\bar{a}$ capital factor (includes depreciation, interest, insurance, tax on capital, repairs) [%/year],

$\dot{m}_F^0$ annual amount of fuel [tU/year],
k_F specific fuel costs [$/tU],
C staff members for service [persons],
k_C annual rate of expenses for staff [$/person·year],
$\dot{m}_{ai}^0$ annual amount of auxiliary materials (chemicals etc.) [t/year, m^3/year…],
k_{ai} specific expenses for auxiliary materials [$/t, $/$m^3$…],
$\dot{m}_D^0$ annual amount of waste [t/year],
k_D specific expenses for waste disposal [$/t].

Furthermore the single terms in (1.255) can be interpreted as follows:

x_C capital-dependent costs,
x_{oper} operating service costs,
x_F fuel costs,
x_{ai} costs of auxiliary materials,
x_D costs of waste disposal (including intermediate and final storage).

The capital-dependent costs, x_C, are also often used in the form $x_C = k_{spec} \cdot \bar{a} / T$, with k_{spec} as the specific investment costs [$/kW (el)]. The share x_F can be written as

$$x_F = \frac{k_F}{B \cdot \bar{\eta}}, \tag{1.256}$$

with $\bar{\eta}$ being the net efficiency of the plant [%] and B the burn-up of the fuel. The dimension of B is MW d/t U or kWh/kg U if uranium is inserted as fuel. In case of use of plutonium in MOX fuel elements (mixed oxide) the total amount of heavy materials is inserted. The parameter k_F contains all expenses for uranium ore, enrichment, conversion and fuel manufacturing. Basically the capital-dependent costs as well as the staff expenses are virtually independent from the amount of electrical energy produced. All further costs are proportional to the amount of the electrical work produced. This dependence can be written in the form of total costs per year:

$$K_{total} = C_1 \cdot P_{el}^0 + C_2 \int_0^{1a} P_{el}(t) \cdot dt = (C_1 + C_2 \cdot T) \cdot P_{el}^0 . \tag{1.257}$$

A very important parameter is T, the number of hours of full-power operation per year (Fig. 1.124). For nuclear power plants, which have relatively high specific investment costs, it is important to realize as high as possible values of T. These plants are operated in the base-load range.

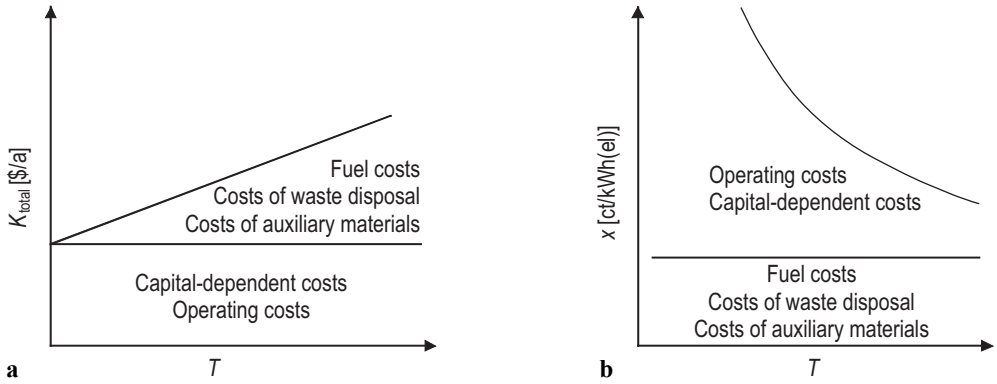


Fig. 1.124. Qualitative dependence of costs on the annual number of full-load hours, T: **(a)** total costs per year, K_{total}, (**b)** electricity generating costs, x.

In general power plants are operated for more than 30 to 40 years. Including the time spans for planning (T_P), construction (T_C), operation (T_O), safe enclosure (T_{SE}) and decommissioning (T_{DE}), such a project has a lifetime of more than 70 years. During this time, cost parameters like fuel costs, operation and labor costs, as well as interest rates and legal aspects can heavily change. Figure 1.125 presents a qualitative picture of expenditures during the whole project lifetime. Therefore the application of dynamic methods for comparing the cost structures of different power plants is necessary. One very important method is the life-cycle cost assessment, in which all the expenses and benefits of later years are backdated to the first year of operation.

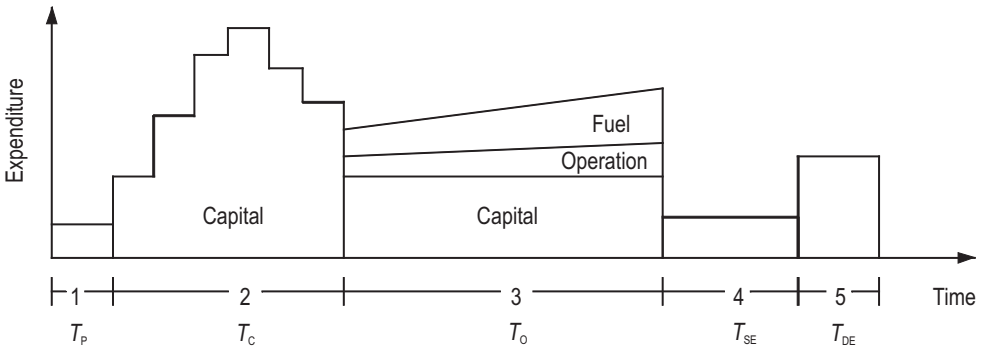


Fig. 1.125. Expenditures during the different phases of a nuclear power plant project. The time spans shown are for planning (T_P), construction (T_C), operation (T_O), safe enclosure (T_{SE}), and decommissioning (T_{DE}).

1.3.2 Investment costs

The overall investment costs of a power plant, k_{invest}, first of all include the direct investment costs, k_{direct}, for the plant as a whole. Additional costs for interest, taxes and insurance of the capital during the construction period as well as additional shares due to inflationary effects have to be taken into account. Furthermore, material and financial contributions by the utility itself, such as for ground, infrastructure, authorizations, checks and starting operation as well as similar expenses have to be considered calculating the direct investment costs. For the plant's closure and decommissioning after its service time some financial assets have to be reserved already at the moment of the plant's initiation. These also have to be added to the investment costs. Summing up the direct investment costs and adding the extra shares, the total investment costs result:

$$K_{\text{invest}} = K_{\text{direct}} \cdot \left(1 + \sum_i \alpha_i\right). \tag{1.258}$$

The extra shares α_i, being given in percent, are explained below. The numbers in brackets were roughly valid for the last nuclear power plants built in Germany in the 1980s.

- α_1 for payment of interest during construction time (15 %),
- α_2 for insurance during construction time (3 %),
- α_3 for taxes during construction time (3 %),
- α_4 for inflation (4 %),
- α_5 for the starting phase (2 %),
- α_6 for builder's own material and financial contributions (3 %),
- α_7 for plant shutdown and decommissioning (10 %).

According to these numbers in case of a nuclear power plant an extra charge of around 30 % had to be expected. For a coal-fired plant an extra charge of nearly 20 % and for a plant based on natural gas (com-

bined cycle: gas and steam turbine plant) with a very short construction time only less than 15 % are necessary at present.

The investment costs of German power plants have substantially risen in the years between 1970 and 1990 not only due to inflation rates but especially because of ever higher requirements as far as the legislation and permissions as well as standards for plant operation are concerned. Substantial additional costs, for example, were caused by higher requirements concerning protection against outer impact on the plant (such as earthquakes or airplane crashes) as well as technical improvements, for instance in the range of decay-heat removal systems. Figure 1.126 shows the temporal development of the specific investment costs of nuclear power plants in Germany.

A development ocorresponding to the "normal" inflation rate would have caused an increase by only a factor of two between 1970 and 1986. For calculations concerning the economic conditions of nuclear power plants it is therefore essential to take account of the year in which the operation of the power plant started.

Today the overnight construction costs in OECD countries are estimated to range between 1500 and 2000 $/kW (el). In some non-OECD countries cost estimates between 1000 and 1800 $/kW (el) have been published [00Dur, 98OEC].

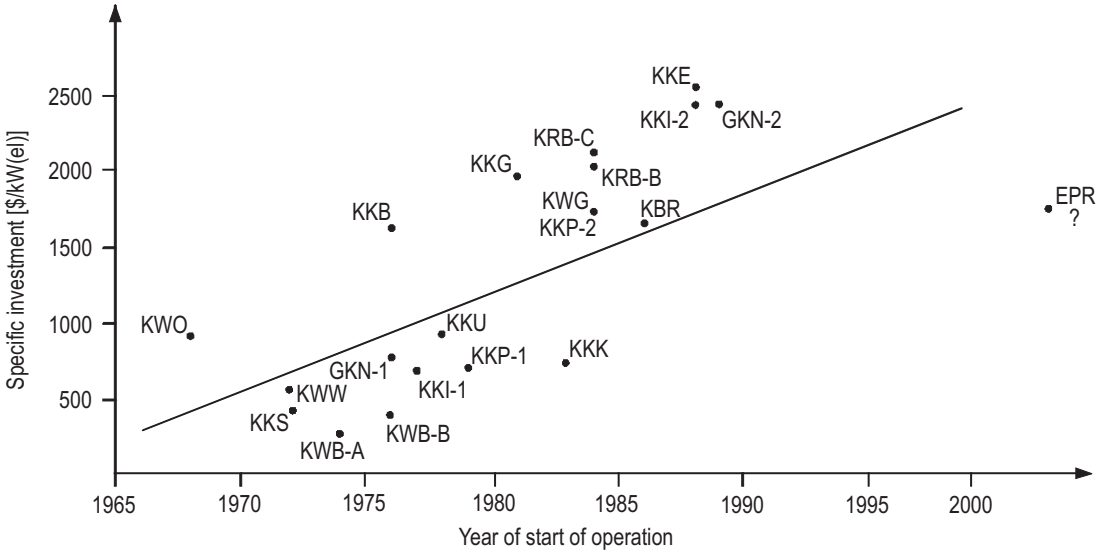


Fig. 1.126. Specific investment costs for German nuclear power plants dependent on the point of time when operation started.

1.3.3 Capital factors

The capital factor, $\bar{a}$, in (1.254) is comprised of a share a (annuity) for the capital depreciation and payment of interest on capital, a share b_1 for taxes, as well as a share b_2 for the insurance. A further share b_3 for repairs required is often added as well. The figure b_3 can of course vary significantly depending on the type of plant, and during the time of operation. Therefore suitable average figures have to be defined. Consequently, the following expression can be defined for $\bar{a}$:

$$\bar{a} = a + b_1 + b_2 + b_3. \tag{1.259}$$

The figure a can be derived as a so-called annuity factor by means of compound interest calculation and makes clear, that annually a constant rate of money, ΔK, related to the plant capital, K_{inv}, has to be raised to finance the plant. The factor of interest payment, q, is determined by the interest rate, p. With the

payment of interest, $q = 1 + p$, one can compare the interest on the total capital, K, about N years with yearly payment of an amount of money, g, with interest payment over the relevant years:

$$K \cdot q^N = g\left(1 + q + q^2 + ... + q^{N-1}\right) = g \cdot \frac{q^N - 1}{q - 1}. \tag{1.260}$$

From this relation and with $g = K \cdot a$ one obtains the annuity:

$$a = \frac{q^N \cdot (q-1)}{q^N - 1}. \tag{1.261}$$

The annuity, a, is shown in Fig. 1.127 dependent on the interest rate, p, and the depreciation time, N. For rough estimates to assess types of plants, in Germany currently a figure of about $\bar{a}$ = 15 %/year can be employed. After the depreciation time has passed (17 years are often chosen in this context), the shares b_1, b_2 and b_3 have to be considered separately. However, these percentages then certainly relate to the conditions of the present cost structures.

Naturally the situation in other countries might be different, especially higher inflation rates can require much more detailed analyses even for rough comparisons. In these cases a life-cycle analysis including assumptions on escalation rates (e) on labor, fuels and auxiliary materials will be necessary.

When comparing different concepts of electricity production sometimes an estimate including the initial investment as well as the fuel and operation costs over the total lifetime helps to identify the economic conditions:

$$K_{\text{total}} = K_{\text{inv}} + \Phi(e, p, N), \tag{1.262}$$

$$\Phi(e, p, N) = \left(\frac{1+e}{1+p}\right) \cdot \frac{1 - \left(\frac{1+e}{1+p}\right)^N}{1 - \frac{1+e}{1+p}}. \tag{1.263}$$

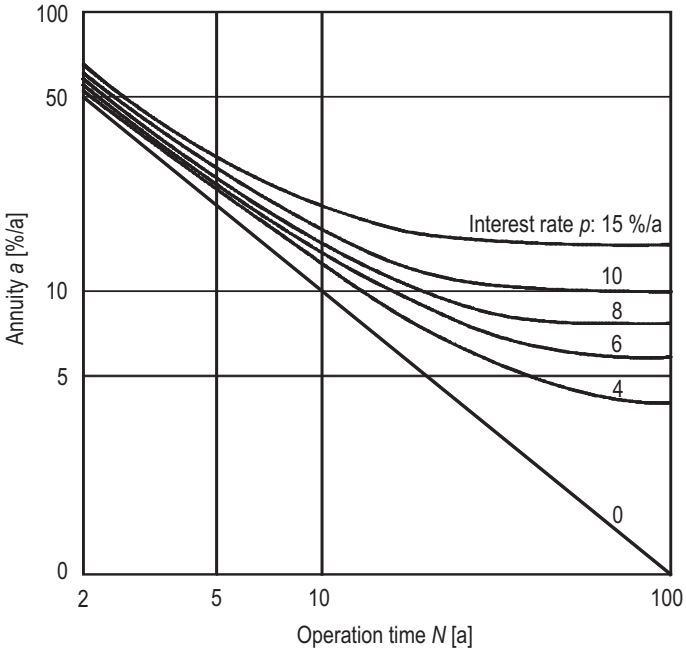


Fig. 1.127. Annuity, a, dependent on the interest rate, p, and the depreciation time, N.

1.3.4 Hours of full-power operation

The number of hours of full-power operation, T, of the plant during one year can be calculated from the relation:

$$\int_0^{1a} P_{el}(t) \cdot dt = P_{el}^0 \cdot T \,, \tag{1.264}$$

on the basis of the operational diagram of the plant. Load factors, l, are often used as well:

$$l = T / 8760 \,. \tag{1.265}$$

As an example, Fig. 1.128 represents the operation history of a German nuclear power plant during a very successful year. The time and work availability of the plant were 96 % and 92 %, respectively. There was just one standstill because of change of fuel elements and yearly maintenance and inspection. During April there was the usual stretchout operation because of reactivity reasons. The number of unplanned standstills was zero. Nowadays a very good availability of nuclear power plants is realized in other countries as well. Figure 1.129 shows the development of this parameter and of the collective doses as a measure for the quality of operation in the USA and Germany [01Ato]. Worldwide an average availability of 80 % is realized by now (Fig. 1.130).

The availability factor of nuclear power plants has been substantially improved in recent years by applying technical improvements, improved nuclear services and general modes of operation. Availability factors of about 90 % are nowadays usual for good plants. Therefore orders of magnitude of 8000 h/a can be applied for calculating the generating costs of electrical energy.

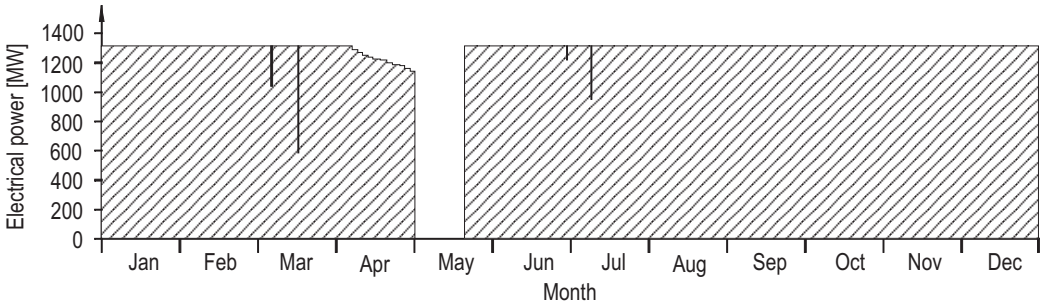


Fig. 1.128. Operation history of a nuclear power plant (PWR, Emsland/Germany, 1300 MW (el), 1998).

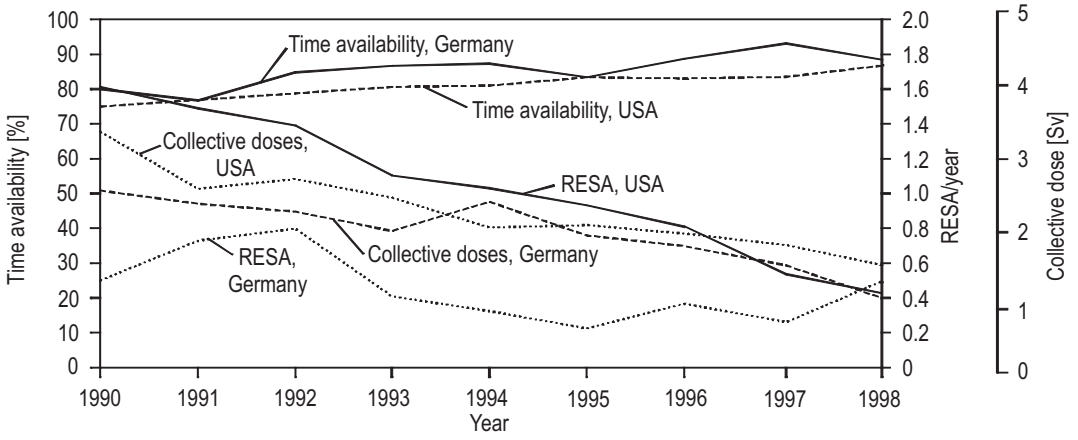


Fig. 1.129. Development of time availability, of RESA (fast shutdown of reactor), and of collective doses in the USA and Germany.

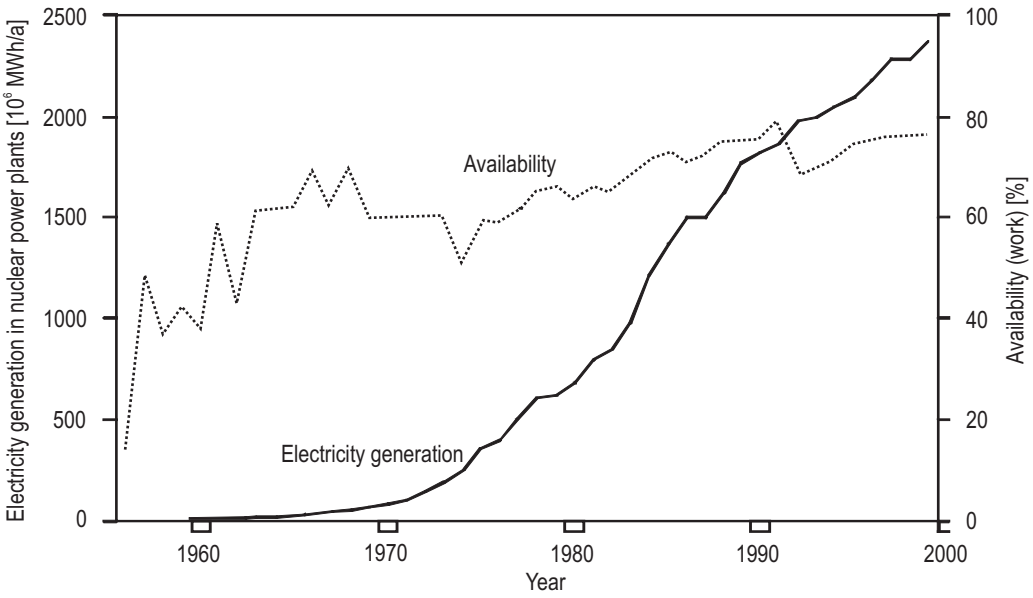


Fig. 1.130. Worldwide development of electricity generation and availability of work in nuclear power plants.

1.3.5 Efficiencies of plants

For the efficiency of plants one finds the relation:

$$\eta_{\text{net}} = \prod_i \eta_i = \eta_{\text{SG}} \cdot \eta_{\text{th}} \cdot \eta_{\text{mech}} \cdot \eta_{\text{gen}} \cdot \eta_{\text{del}} , \tag{1.266}$$

including the single efficiencies of the steam generator (η_{SG}), the thermal cycle (η_{th}), the turbine (η_{mech}), and of the electrical generator (η_{gen}). The electrical energy required to operate pumps, auxiliary systems and all other electrical consumers in the power plant itself is represented by the efficiency of delivery, η_{del}.

Principally the net electrical power and fuel consumption are subject to changes during the year. The cooling conditions change, and therefore the net efficiency is not constant either. Net efficiencies of plants can be defined by the expression:

$$\overline{\eta} = \frac{\int_0^{1\text{a}} P_{\text{el}}(t) \cdot \mathrm{d}t}{\int_0^{1\text{a}} \dot{m}_{\text{F}}(t) \cdot B \, \mathrm{d}t} \quad \Rightarrow \overline{\eta} = \frac{P_{\text{el}}^0 \cdot T}{\dot{m}_{\text{F}}^0 \cdot B} , \tag{1.267}$$

with $\dot{m}_{\text{F}}^0$ being the amount of nuclear fuel [t/year], and B denoting the burn-up of fuel elements [MWd/t]. Table 1.26 contains some efficiencies parameters of different reactor types.

The net efficiencies of LWRs is presently around 33 %, for the future EPR the manufacturer offers more than 34 %. HTR plants with steam cycle have reached 40 % until now, the potential with higher steam temperatures (600 °C) will be 43 %. HTR gas-turbine cycles are designed for values of 45 to 48 % with a potential of nearly 50 %. In the future similar values will be valid for combined cycles in connection with an HTR as heat source.

Table 1.26. Efficiencies of different reactor types.

Reactor type	Max. coolant temp. [°C]	Steam temp. [°C]	Steam pressure [bar]	Efficiency [%]	Remarks
PWR	325	279	63	33.5	Wet cooling tower
BWR	286	283	67	32.5	” ” ”
CANDU	310	260	47	29.4	” ” ”
RBMK	284	284	70	32	” ” ”
Magnox	410	391	38.7	32.9	Seawater cooling
AGR	646	541	167	41.2	” ”
HTR (steam turbine)	750	530	186	40	Dry air cooling tower
HTR (gas turbine)	900	—	—	45	Seawater cooling
LMFBR	546	495	165	38	Wet cooling tower

Naturally the conditions of cooling influence the efficiencies, therefore the values mentioned above can change during the year. Compared to cooling by fresh water from rivers or from the sea, in plants with wet cooling towers there is a loss of efficiency of 1 point, whereas with dry air cooling towers the loss is more than 2 points. Furthermore all types of plants can be used as cogeneration plants, therefore the total use of energy could be as high (around 80 %) in nuclear power plants as in conventional plants.

1.3.6 Burn-up of nuclear fuel

The burn-up is defined by the relation

$$B = \frac{1}{\rho} \int_0^{\tau} \bar{E}_\mathrm{f} \cdot \Sigma_\mathrm{f}(t) \cdot \phi(t) \cdot \mathrm{d}t \,. \tag{1.268}$$

Here, ρ is the UO_2 density, $\bar{E}_\mathrm{f}$ is the average energy of the fission process (200 MeV), Σ_f denotes the macroscopic fission cross section, ϕ is the neutron flux, and τ is the time of fuel insertion. For today's PWRs the value of B is between 40000 and 45000 MW d/t heavy metal, corresponding to an enrichment of uranium of 3.5 to 4 %. High temperature reactors with entirely ceramic fuel elements reach 100000 MW d/t and more. In case of LWRs there is a tendency to change to more than 5 % enrichment and to reach burn-up values of 60000 MW d/t, which could be more economic.

The knowledge of the burn-up and the number of hours of full-power operation allows a very simple estimate of the amount of fresh fuel which has to be inserted into the reactor per year, as well as of the amount of spent fuel which has to be removed from the core:

$$\dot{m}_\mathrm{F}^0 = \frac{P_\mathrm{el}^0 \cdot T}{B \cdot \bar{\eta}} \,. \tag{1.269}$$

A simple example for a 1300 MW PWR yields the following values: $\bar{\eta} = 33$ %, $T = 7500$ h/year, $B = 40000$ MW d/t, resulting in around 30 t uranium/year as the necessary amount of fresh fuel which is loaded during the yearly standstill period of the plant.

The annual demand of natural uranium can be calculated from the following relation:

$$\dot{m}_\mathrm{U}^0 = \dot{m}_\mathrm{F}^0 \cdot \frac{e_\mathrm{F} - e_\mathrm{T}}{e_\mathrm{N} - e_\mathrm{T}} \cdot \zeta \,, \tag{1.270}$$

where $\dot{m}_\mathrm{F}^0$ is the amount of nuclear fuel required [t/year], e_F is the enrichment of the fresh fuel [%], e_T denotes the tail enrichment [%], e_N denotes the enrichment of natural uranium [%], and ζ is a technical factor (> 1).

In detail the amount of natural uranium and the separative work required to produce 1 kg of enriched uranium depend on the degree of enrichment and on the extent of the tail assay. Some numbers are given in Table 1.27.

The value function is given by

$$V(e) = (1-2e)\cdot \ln\left((1-e)/e\right), \tag{1.271}$$

where e is the enrichment in weight fraction. The separative work unit (SWU) associated with a specific amount of enriched uranium is defined by the expression

$$SWU = M_P \cdot V(e_P) + M_T \cdot V(e_T) - M_F \cdot V(e_F), \tag{1.272}$$

including the amount of tail (M_T), feed (M_F) and product (M_P).

Furthermore the following balance holds:

$$M_T = M_F - M_P. \tag{1.273}$$

Finally one obtains for the separative work unit:

$$SWU = M_P \cdot [V(e_P) - V(e_T)] - M_F \cdot [V(e_F) - V(e_T)], \tag{1.274}$$

which has the unit of mass (kg). The separative work needed to produce a given amount of product increases with enrichment as shown in Table 1.27.

In order to produce 30 t of enriched uranium (≈ 3.5 % enrichment), one needs almost 200 t of natural uranium. This relatively low amount of material is sufficient for the supply of a 1300 MW (el) LWR plant with 8000 hours of full-power operation per year.

Table 1.27. Requirements of natural uranium and separative work to produce 1 kg of enriched uranium (tail assay: 0.2 weight % U235).

Enrichment [weight %]	kg natural U per kg product	kg separative work per kg product
0.711	1.000	0.000
1.00	1.566	0.380
2.0	3.523	2.194
2.5	4.501	3.229
3.0	5.479	4.306
3.5	6.458	5.414
4.0	7.436	6.544
10.0	19.178	20.863
20.0	38.748	45.747
90.0	175.734	227.341

1.3.7 Costs of nuclear fuel

The fuel costs for nuclear power plants, x_F, consist of different shares:

$$x_F = x_U + x_E + x_P = k_F / (B \cdot \overline{\eta}), \tag{1.275}$$

x_U cost share of natural uranium [ct/kWh (el)],
x_E cost share of uranium enrichment [ct/kWh (el)],
x_P share of manufacturing cost for the fuel elements [ct/kWh (el)],
k_F costs for the ready-to-use fuel [ct/t heavy metal],

B fuel burn-up [kWh/t heavy metal],
$\bar{\eta}$ net efficiency of the plant [%].

The investment for the initial core, i.e. the depreciation and payment of interest for the fuel used inside the plant during the entire operation time, can be considered with regard to either the fuel-cycle costs or the investment costs. In the following cost estimation this will be dealt with according to the latter method. In any case, after finishing of the power plant, the fuel of the first core can be used and has a value. Determining the cost shares of natural uranium as well as the enrichment, the exact enrichment of the fuel cycle used has to be considered. The costs of the ready-to-use fuel can be calculated according to:

$$k_F = k_{uranium} \cdot m + k_{enrichment} \cdot a_s + k_{production} \cdot z \,. \tag{1.276}$$

The meaning of the single figures is given below:

$k_{uranium}$ costs of the natural uranium [ct/t U_{nat}],
m quantity factor of the enriched uranium [t U_{nat}/t U_{enrich}],
$k_{enrichment}$ specific costs of enrichment [ct/t separation work],
a_s separation-work-factor [t separation work/ t U_{enrich}],
$k_{production}$ specific costs of production [ct/fuel element],
z quantity of fuel elements [number of fuel elements/t U_{enrich}].

The parameters $k_{uranium}$, $k_{enrichment}$ and $k_{production}$ have been subject to market changes in recent times, or have been reduced by technical progress. Especially in the case of enrichment, the change from gas diffusion to centrifugal procedures have resulted in an extreme decrease of energy demand as well as a significant lowering of costs. In the last decades the burn-up of fuel went up, too. Figure 1.131 shows the development of some cost parameters in time, related to the conditions in Western Europe or Germany [97Jah]. As can be seen in Fig. 1.131a, the price of uranium ore has been particularly low during the last decade.

Total costs of around 1.5 mill. $/t heavy metal for LWR fuel with 4 % enrichment is a rough value for the calculation carried out below. This results in a cost figure of around 0.45 to 0.5 ct/kWh (el) for fuel supply without expenses for waste disposal, if a burn-up of 40 000 MW d/t is assumed.

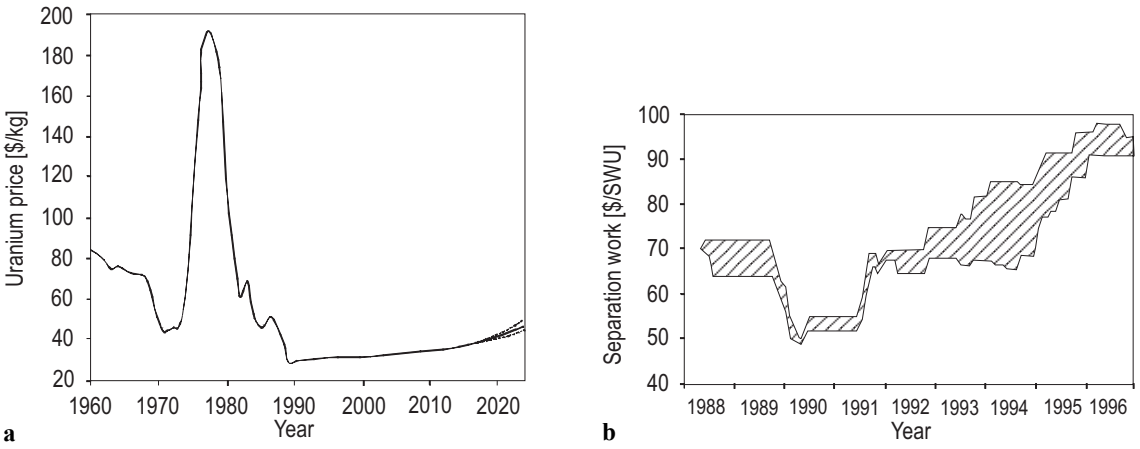


Fig. 1.131. Development of fuel cycle parameters in time: **(a)** uranium price, **(b)** enrichment price (separation work).

1.3.8 Costs of intermediate storage and final storage of spent fuel or high-level radioactive waste

Today there are two main options to handle the spent fuel elements: (1) direct final storage in geological depositories after a sufficiently long intermediate storage time, (2) reprocessing of the spent fuel elements with solidification of the high-level radioactive waste in glass and final storage of the glass products in a deep geological depository. Figure 1.132a shows a schematic flow sheet for these two options.

For the specific case of direct final storage of spent fuel elements, an estimate of the costs can be carried out a follows. The overall costs of waste disposal, x_D, include expenses which are required for interim storage as well as the final storage of spent fuel elements. Furthermore the costs for disposal of low and medium active waste material have to be added:

$$x_D = x_{IS} + x_{FD} + x_{AW}\,, \tag{1.277}$$

$$x_D = \frac{k_{IS}}{B \cdot \bar{\eta}} + \frac{k_{FS}}{B \cdot \bar{\eta}} + \frac{k_{AW} \cdot m}{P_{el}^0 \cdot T}\,, \tag{1.278}$$

with the different parameters explained below:

x_{IS} costs of interim storage [ct/kWh (el)],
x_{FS} costs of final storage [ct/kWh (el)],
x_{AW} costs of waste disposal for low and medium active waste [ct/kWh (el)],
k_{IS} specific costs of interim storage [$/t],
k_{FS} specific costs of final storage [$/t],
k_{AW} specific costs of low and medium active waste [$/m^3/a],
B burn-up [kWh (th)/t],
$\bar{\eta}$ average net efficiency [%],
m amount of low and medium active waste material [m^3/a],
P_{el}^0 electrical power [MW (el)],
T full-load hours [h/a].

These cost factors were also subjected to changes in the course of the recent years [95Hen]. Table 1.28 shows the specific costs of waste disposal, k_{IS} and k_{FS}, which were typical in Germany in the last years. The option reprocessing is included in the table, too.

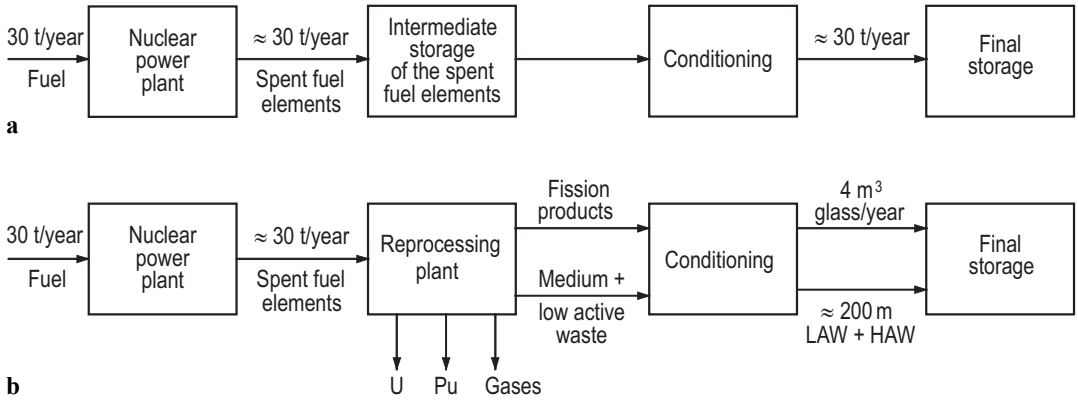


Fig. 1.132. Principles of handling spent fuel elements: **(a)** direct final storage, **(b)** reprocessing and final storage of glass containers (the numbers are related to a 1300 MW (el) LWR plant).

Table 1.28. Costs of waste disposal for LWRs in Germany, representing the two options of direct final storage of spent fuel elements and of reprocessing and final storage of glass blocks.

Position	Reprocessing and final storage of glass blocks	Direct final storage of spent fuel elements
Transport of spent fuel elements, intermediate storage of spent fuel elements		300…400 $/kg U
Conditioning, direct final storage of spent fuel elements in salt mine		1000 $/kg U
Reprocessing, intermediate storage of glass blocks	1300…1800 $/kg	
Final storage of glass blocks in salt mine	500…800 $/kg	
Total specific costs of waste disposal	0.55…0.75 ct/kWh (el)	0.4…0.45 ct/kWh (el)

From a technical point of view in both cases the procedure is based on an interim storage in an air-cooled cast-iron transport vessel either for glass coquilles or for spent fuel elements. Final storage of glass coquilles and direct final storage of spent fuel elements, respectively, is intended to happen in salt mines. With a burn-up of 40000 MW d/t a share of the disposal costs of less than 0.5 ct/kWh (el) results from the interim and final storage of the spent fuel elements. Additionally the expenses for the disposal of low- or medium-activity waste material has to be taken into account, but this is small compared to the above cost figures.

The route of reprocessing, interim storage of glass blocks and again final storage in a salt mine is thought to be a little more expensive than direct final storage of spent fuel elements today. This, however, is a specific condition in Germany at the moment because the by-product plutonium is not considered a worth as future fuel for fission reactors, therefore no credit is assumed for the production of new fuel. In other countries this estimation is different.

Referring to the overall costs for interim and final storage, an important optimization problem needs to be solved. Due to the decrease of decay heat in the course of time and therefore changing conditions in the final storage, there are contrary tendencies as far as the costs of both steps are concerned. An optimal time span of interim storage, t_{is}, has to be fixed, which will be much more likely around a figure of 100 years rather than around 10 years. Naturally the risks caused by a longer intermediate storage above ground and the risk by final storage have to be optimized, too, in the total balance of the fuel cycle.

1.3.9 Overall nuclear fuel cycle costs

The development of the overall nuclear fuel cycle costs during the last decades, including all stages of supply and disposal, is presented in Fig. 1.133. Particularly the reprocessing and disposal costs have strongly increased since the mid-1970s, while the costs of uranium and production of the fuel elements have only moderately grown. The shares of the single process steps or positions sho, that especially the influence of the uranium price on the entire result is virtually of weak importance. Currently the costs of uranium account for merely 3 % of the overall production costs of electricity. Increasing uranium prices therefore can be coped with rather easily. The relation between uranium ore costs and total generating costs of electricity has changed very much during the last three decades. Therefore, as an example, the importance and early need for reprocessing has changed, too. Table 1.29 contains a breakdown of the different shares of an LWR fuel cycle. Naturally this is an estimate for a specific country, in this example for Germany [02Wir]. In other countries the economic conditions are different [94OEC].

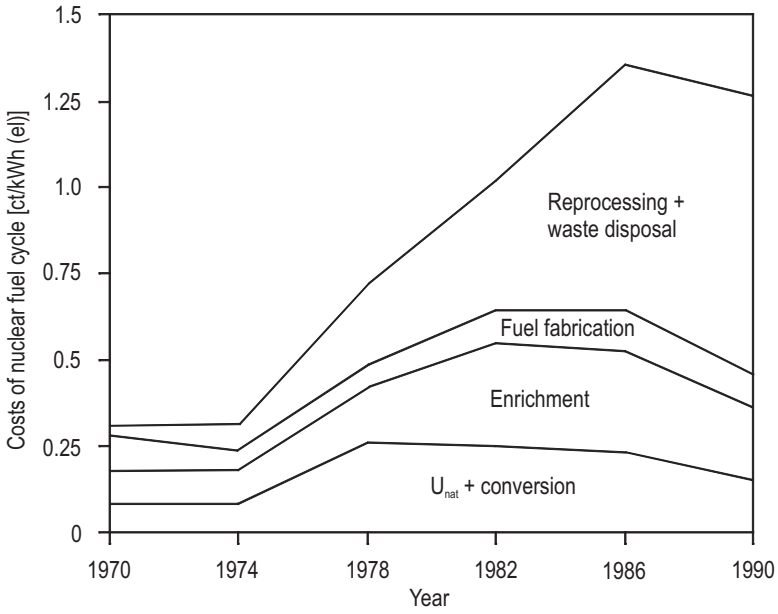


Fig. 1.133. Development of nuclear fuel cycle costs in Germany from 1970 to 1990.

Natural uranium is relatively cheap on the world market today. The amount of cost-effective uranium ore is limited (see Table 1.30) and so is the time period of supply of nuclear power plants [01Bar]. However, large amounts of more expensive uranium resources should be available in the future, especially the use of breeding processes allows to use nuclear technology with large capacities for thousands of years.

Depending on the breeding factor and the losses of plutonium and uranium in the fuel cycle, especially in the reprocessing steps, the utilization of uranium might be a factor of 20 to 40 higher in fast breeder reactors than in present LWRs. Naturally there is the possibility to realize thermal high converter reactors on the basis of a LWR or HTR, too. This allows to have a utilization factor of 10 compared to today's fuel cycle economy. Additionally it should be mentioned that thorium-232, which can be converted to fissionable U-233, is available with similar reserves as uranium. Altogether it can be stated that fuel for nuclear power plants will be available for a very long time of thousands of years.

Table 1.29. Cost shares in the fuel cycle of a German LWR (1998, $\bar{\eta} = 33$ %, $B = 40\,000$ MW d/t).

Position	Specific value	Specific costs	Cost figure [mill. \$/t U_{enr}]	Share in generating costs [ct/kWh (el)]	Share [%]
Natural uranium	6.3 t UO_2 / 1 t U (3.4 %)	65 \$/kg	0.41	0.13	16
Conversion ($UO_2 \rightarrow UF_6$)		10 \$/kg	0.065	0.02	2.4
Enrichment	5.2 kg SWU / 1 kg U (3.4 %)	120 \$/kg	0.65	0.2	24.3
Fuel fabrication		300 \$/kg	0.3	0.1	12.2
Interim storage	1 Castor vessel / 2 t U	1 mill. \$ / 1 Castor	0.5	0.16	19.5
Final storage	1 steel vessel / 3 t U	2 mill. \$ / 1 steel vessel	0.66	0.21	25.6
Total			2.58	0.82	100

Table 1.30. Some data of uranium reserves.

Type of reserves	Quantity [10^6 t]	Uranium content [kg/t]	Estimated production costs [$/kg]
Uranium ore (certain)	1.5	≈ 30	< 40
Uranium ore (estimated)	2.5	≈ 30	< 80
Uranium ore (estimated)	≈ 25	< 10	< 300
Uranium in shale	≈ 200	10^{-1}…10^{-2}	> 500
Uranium in granite	≈ 2000	10^{-2}…10^{-3}	> 500
Uranium in seawater	≈ 4000	3×10^{-6}	> 1500

1.3.10 Personnel costs and costs for auxiliary materials

Regarding the overall production costs, personnel costs and costs for auxiliary materials usually remain rather small, especially for large-sized nuclear power plants. Concerning plants of smaller size, this share can be significantly more important depending on the concept of operation and has to be analyzed in detail. For large plants a figure of 0.2...0.3 persons/MW (el) can be assumed. This corresponds to around 0.15...0.2 ct/kWh (el) for expenses for personnel.

As far as auxiliary materials are concerned, extra payments constitute a fixed percentage of the investment capital. This assumption is based on long-term experiences. Normally a contribution of 0.1 to 0.15 ct/kWh (el) can be taken into account for this position. Sometimes the costs of repairs are added to the operation costs, too. Altogether the operation costs represent more than 15 to 20 % of the total generating costs in the present nuclear power plants.

1.3.11 Total generating costs of electricity in nuclear power plants

An estimate of the total generating costs results in curves as shown in Fig. 1.134. Here, three different nuclear power plants in Germany are considered, which have been put into operation in 1976 and 1986, respectively. The figure shows the great importance of the number of full-power hours per year and of the point of time when operation started. Nuclear power plants are typically base-load power plants with more than 7000 hours of full-power operation per year. After 17 years (German conditions) power plants have normally been depreciated, and the capital-dependent part of the costs is reduced.

A further very important result of a detailed analysis is the small share of uranium ore in the total generating costs. Today this is in the order of 3 to 5 % for new nuclear power plants. Therefore even drastically rising uranium ore costs would not influence the economic result very much. This is a big advantage compared to power plants based on fossil fuels in which this dependence is much higher (50 to 80 %). The uncertainty of the development of gas prices, for instance, obstructs the construction of new gas-fired power plants (combined cycles).

New reactors to be built today would cause lower investment costs. This is a result of strong international competition on the market of power plants, of technical progress, and of the globalization of the companies which build power plants. A reasonable cost figure is 1500 to 2500 $/kW (el). This allows for generating costs of new plants around 3 ct/kWh (el).

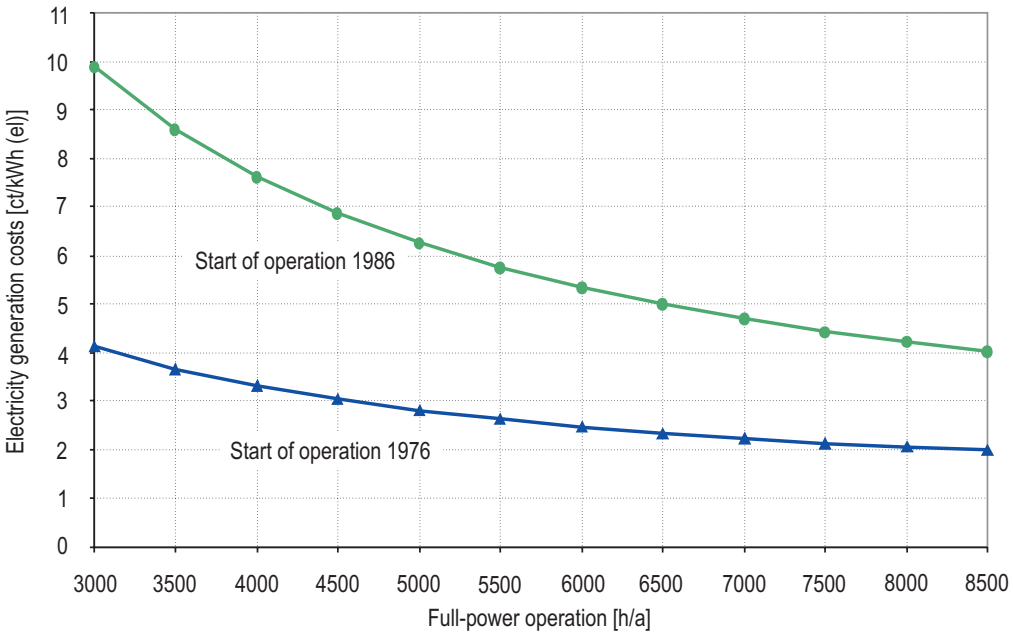


Fig. 1.134. Costs of electricity generation in nuclear power plants as a function of the number of full-load hours per year ($P_{el}^0 = 1300$ MW, $\bar{a} = 15$ %/a, $C = 350$, $k_C = 3.5 \times 10^4$ \$/c·a, $x_D + x_E = 0.82$ ct/kWh (el), start of operation 1976: $k_{inv} = 600$ \$/kW (el); start of operation 1986: $k_{inv} = 1750$ \$/kW (el)).

1.3.12 Comparison of generating costs

Compared to other options of electricity generation, nuclear energy is competitive in Germany and in many countries of the world, as shown in Table 1.31.

For the new EPR concept (European pressurized reactor, 1700 MW (el)) a cost figure of 1700 \$/kW (el) was estimated for the investment [01Bre]. Under these conditions the EPR would be competitive with electricity from hard coal from the world market or from natural gas in combined processes, if life-cycle cost analysis is applied. In this method the escalation of fuel prices during decades of operation is included. This changes the picture because the production costs of the nuclear power plants stay almost constant. The reason is that the share of uranium in the total costs is very small. In Fig. 1.135 the costs for different electricity generating options are displayed (related to conditions in Finland). Figure 1.136 shows calculated shares of capital, fuel and operation costs for the different energy options, underlining the minor role of fuel costs in case of nuclear power plants.

Table 1.31. Comparison of generating costs of electricity (German conditions, year 2000).

Primary energy	Specific investment [\$/kW (el)]	Fuel costs [ct/kWh (el)]	Generating costs of electricity [ct/kWh (el)]
Coal (world market)	1000	1.3	4
Natural gas	400	3	4.5
Wind power	1000	0	5…10
Photovoltaic (direct use)	7000	0	70
Photovoltaic (H_2 storage)	> 7000	0	> 250
Nuclear (old plants)	600	0.8	2
Nuclear (new plants)	1500	0.8	3.6

Overall it can be stated that nuclear energy will be competitive to all other options in the future, except for hydropower in large plants on some specific sites. The cost figures of coal and natural gas do not include external costs für CO_2 sequestration and final disposal or costs for penalties regarding CO_2 emissions. Especially the question how to store electrical energy in large quantities for longer times causes big problems and high additional costs for all regenerative energy systems. Only hydropower plants using pumping storage systems are a feasible solution today. However, their capacity is very limited.

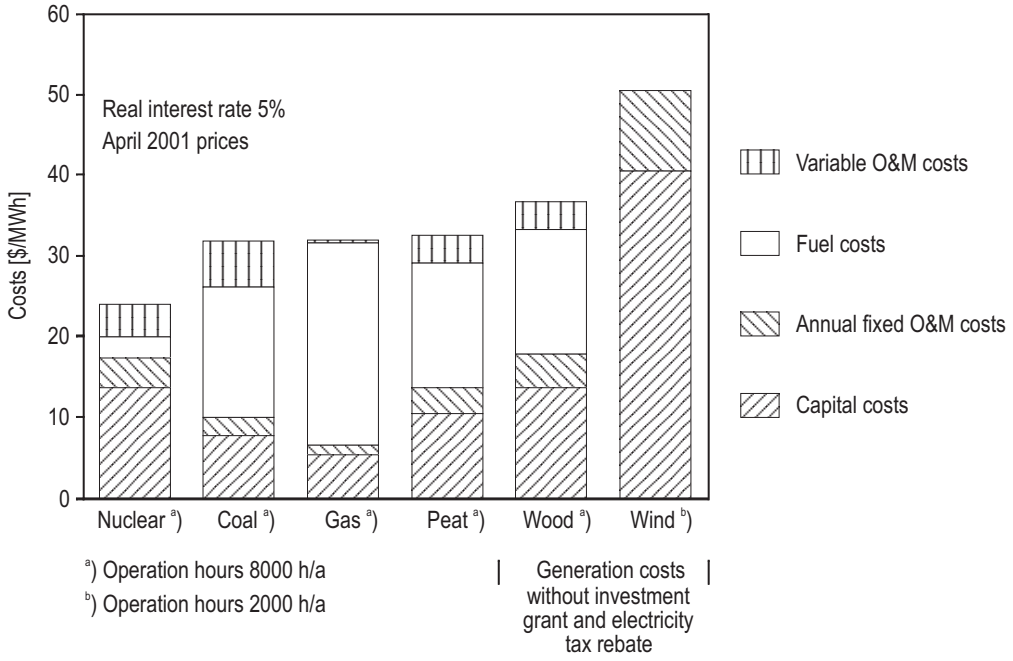


Fig. 1.135. Comparison of different options for electricity generation (Finland) [02Paa].

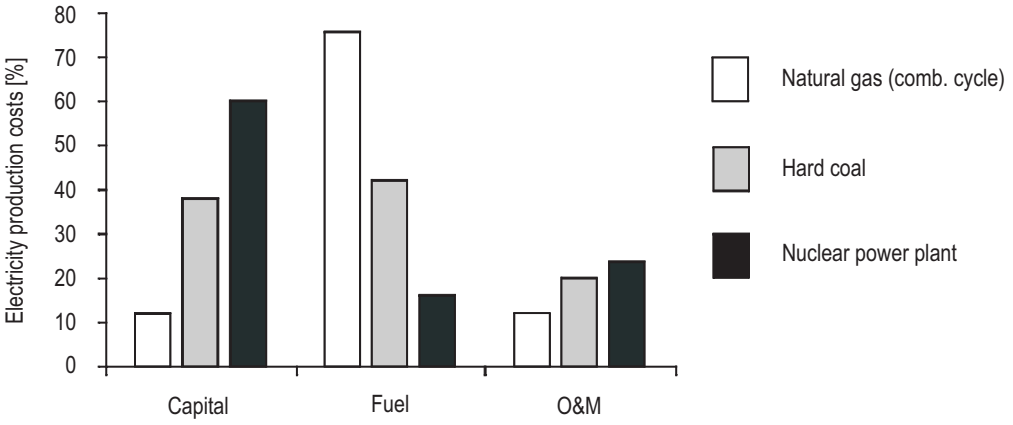


Fig. 1.136. Production costs of electricity (life-cycle costs) for different future options (Finnish conditions, start of operation in 2005, operation time: 35 years).

1.3.13 External costs

All processes of electricity production cause environmental damages. These effects can be included by the external costs of electricity generation, x_{ext}. In the future these have to be added to the generating costs described in the preceeding sections. The total costs are then represented by

$$x_{tot} = x_{gener} + x_{ext} \,. \tag{1.279}$$

Nowadays radioactive emissions during the normal operation of a nuclear power plant, for instance, or emission of dust, NO_x and SO_2 by fossil-fired power plants can be very effectively avoided by technical measures. These substances are retained inside the power plants and are deposited in well-suited places. The additional costs for this environmental protection are then already internal costs. The question how far emissions are reduced by technical effort is normally decided by legal requirements. In general a compromise must be made between economic and ecological aspects to find a sensible limit. Figure 1.137 presents a qualitative picture for the case of optimizing the technical effort for fission product retention in a nuclear power plant in normal operation.

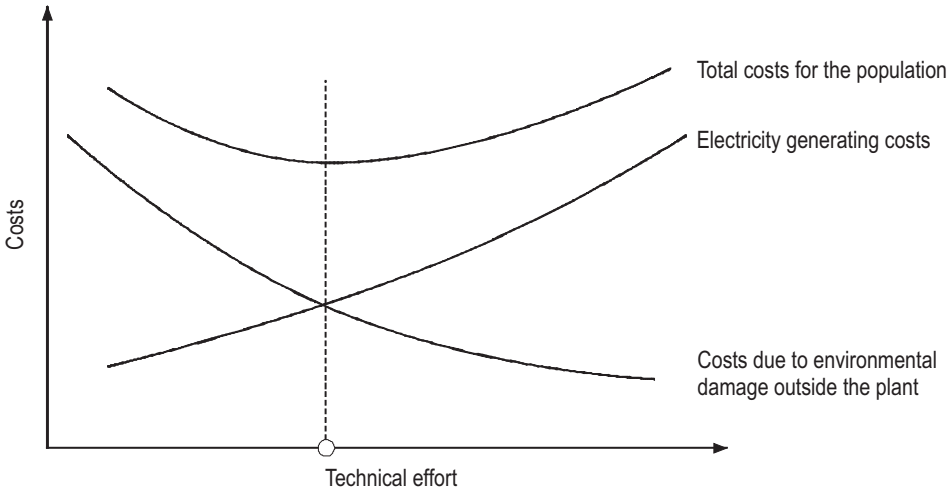


Fig. 1.137. Considerations on the optimization of technical effort for environmental protection in a case of nuclear power plants (example given: retention of fission products during normal operation).

The above considerations seem to be applicable to the normal operation of nuclear or conventional power plants. Quite different requirements exist for the evaluation of methods to calculate external costs for the CO_2 emissions by fossil-fueled power plants, or for estimating the consequences of extreme accidents in nuclear power plants with large damage by contamination of land. In many studies today the additional external-cost figure for accidents in nuclear power plants is calculated as follows:

$$x_{ext} = \frac{D \cdot W}{P_{el}^0 \cdot T} \,, \tag{1.280}$$

where D is the total damage [$], W denotes the probability of occurence of damage [1/year], and $P_{el}^0 \cdot T$ is the annual electricity production [kWh (el)/year]. The number D contains all monetary consequences of a severe accident, like loss of land, investments and infrastructure, as well as costs for casualties and late cancer in the population. Naturally those monetary numbers are handled quite controversially, and the results for the external costs are very different (see Table 1.32). The possible costs of damage and the probability of a large release of radioactivity are a controversial issue. Assuming the values $D \approx 10^{13}$ DM, $W \approx 4 \times 10^{-5}$/year, $P_{el} = 1300$ MW, $T = 8000$ h/year, the the result would be 4.3 Dpf/KWh (el) (see Table 1.30: Prognos, Ewers/Renning, 1992). Other authors assume different and partly much smaller values.

Table 1.32. Different estimates of external costs of electricity generation in nuclear and coal-fired plants, respectively. The numbers are given in the unit Dpf/kWh (el), where 1 Dpf ≈ 0.5 ct [00Kug].

Authors	Year of publication	Nuclear plants		Coal-fired plants	
		All effects	Only severe accidents	All effects	Only green-house effect
Hohmeyer	1989	9.7…24.5	3.5…21	2.94…8.54	
Friedrich/Voß	1989	0.5…0.6	0.02…0.07		
Ottinger et al.	1990	4.8	3.7	7	2.3
Friedrich/Voß	1992	0.03…0.7	0.01…0.07	0.44…2.35	
Prognos (Ewers/Renning)	1992		4.3		
Pearce et al.	1992	0.13…0.75	0.05…0.7	2.1	0.85
OECD	1992	0.02…0.2	< 0.05	0.3…1	
US DOE/ORNL/RFF	1993	0.05…0.07	0.01…0.02		

Clearly, a weakness of the above method is that today the number of nuclear power plants is too small, and the technical status of the plants is too different; moreover, especially the monetary damage by an extreme accident might be too high to carry out commonly accepted calculations of external costs. External costs would become calculable only if a nuclear technology with a limitation of the damage to the plant was realized (see Sect. 1.4). Just then it would be possible to have an insurance for the monetary damage in case of severe accidents, too.

Altogether the nuclear system, as other systems of energy technology, too, in the future needs an optimization between generating costs and external costs similar to the picture in Fig. 1.137:

$$x_{\text{tot}}(E) = x_{\text{gener}}(E) + x_{\text{ext}}(E)\,, \tag{1.281}$$

where the parameter E characterizes the efforts to gain nuclear power plants that are as safe as possible and as economical as possible.

1.4 Safety questions

1.4.1 Core melt incidents in light water reactors

The safety analysis of nuclear reactors covers a wide spectrum of accidents. One of the events playing a major role is the total loss of coolant and of cooling, especially in the light water reactors operating worldwide today.

Every reactor produces decay heat due to the β- and γ- decay of fission products. The heat production in the first hour after a shutdown amounts to several percent of the nominal thermal power (Fig. 1.138). The relative decay heat production can be described by the Way-Wigner equation, where t is the time after shutdown of the chain reaction, and t_o is the operation time before shutdown:

$$P_{\text{decay}} / P_{\text{thermal}} = 0.062 \cdot \left(t^{-0.2} - (t + t_o)^{-0.2} \right) \qquad (t, t_o \text{ in s}). \tag{1.282}$$

In case of a 3800 MW (th) plant between 250 and 20 MW (depending on the time after shutdown) must be removed by devices in order to avoid damages. During normal operation the steam generators of the primary system serve for this task. However, there are various possibilities to fail, so that additional safety systems are included in the layout.

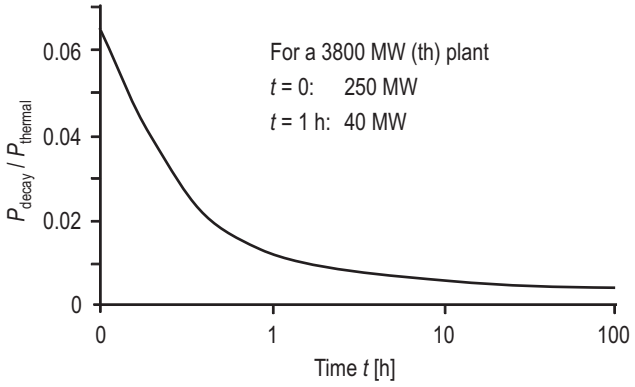


Fig. 1.138. Decay heat production in nuclear reactors versus time after shutdown ($t_o \to \infty$).

Every reactor is therefore equipped with very sophisticated systems to remove the decay heat released from the core. Figure 1.53a above shows the schematic component setup of a decay-heat removal system for a modern PWR [88Per]. In case of decreasing primary pressure caused by leaking primary components (distinguished as large, medium or small leaks), depending on the primary water pressure, several stacked water feeding systems are activated. Their common task is to ensure a flooded reactor core and to avoid core melting. If the primary pressure decreases below 11 MPa, the high-pressure safety injection system starts, takes borated water from flooding stores and feeds it into the cold legs of the primary coolant circuit. Continuously decreasing primary pressure initiates the low-pressure safety injection pump to start up (0.9 MPa), taking over the work of the aforementioned one. Finally, if the flooding store is empty, large amounts of water have been accumulated in the containment sump, so that the low-pressure system can be switched into the sump water circulation mode, which is the stable afterheat removal mode. If a large leak occurs, e.g. a rupture of the primary coolant pipe, the system pressure decreases so fast that the high-pressure injection system cannot start early enough to counteract that accident. In this case the pressurized-water containing stores come into operation (2.5 MPa), which serve as water supply until the low-pressure safety injection has started.

In modern PWR systems, this combination of equipment is connected to each primary loop, so that four times 100 % of the maximum afterheat power can be removed (1 of 4 redundancy). An intermediate water-to-water heat exchanger located in the sump water circulation piping serves for the final heat transfer from the entire system to the environment (cooling tower, lake, river). Highest quality levels during the fabrication of components, sophisticated supervision, repeated tests and the continuous replacement of

components assure a very low non-availability of the afterheat removal systems. The dominant cause of failures comes from the electric power supply for the pumps. Despite all efforts, a certain non-availability remains, e.g. 10^{-4} per demand. Therefore these sophisticated systems are not able to protect the environment completely against the consequences of a core meltdown.

Normally a system of stacked barriers retains the radioactivity inside the nuclear power plant: The most significant are the fuel element, the primary circuit and the containment. After the decay heat removal has failed, rising temperatures destroy this barrier system which is very effective during normal operation. A complete long-term failure of the decay heat removal results in a core meltdown in case of the current light water reactors. Associated with the course of the core meltdown a large-scale release of fission products into the containment and afterwards a release into the environment is possible with partially catastrophic effects. As a protection against accidents, especially a failure of decay heat removal, nuclear power plants are equipped with complex safety components. Owing to these complex precautionary measures, it is generally taken for granted, that accidents leading to a core meltdown will only occur very rarely. This is also the reason why hitherto no installations to control possible core meltdown accidents have been realized for the light water reactors currently in operation. A detailed description of core melt accidents is very difficult due to the numerous interrelated physical and chemical phenomena [80GRS, 89GRS]. Therefore only approximate views and models are feasible. A very simplified model which also allows the assessment of characteristic times up to the melting is presented here.

From the large variety of incident flows that can lead to a core meltdown, the following should be put in focus: a large leak occurs in one of the main coolant ducts, the emergency cooling works, the hydraulic accumulator and the low-pressure feeding from the flood vessel are operational. The “emergency and decay-heat cooling system” stops operating after some 20 minutes after switching to “cooling with sump water”. For the further course of the accident, the following models can be developed (Fig. 1.139): The decay heat production results in a heating and evaporation of the water inside the reactor pressure vessel (phase 1).

A lowering of the water level in the core area occurs, the upper zones of the core are heated and melt (phase 2). Additionally hydrogen is formed above a temperature of approximately 950 °C by a chemical reaction between the zirconium of the fuel element cladding and steam. This reaction is strongly exothermic and further heats up the core. In a third phase the reactor vessel is penetrated by the corium formed, and in a fourth phase the foundation of the containment could be destroyed.

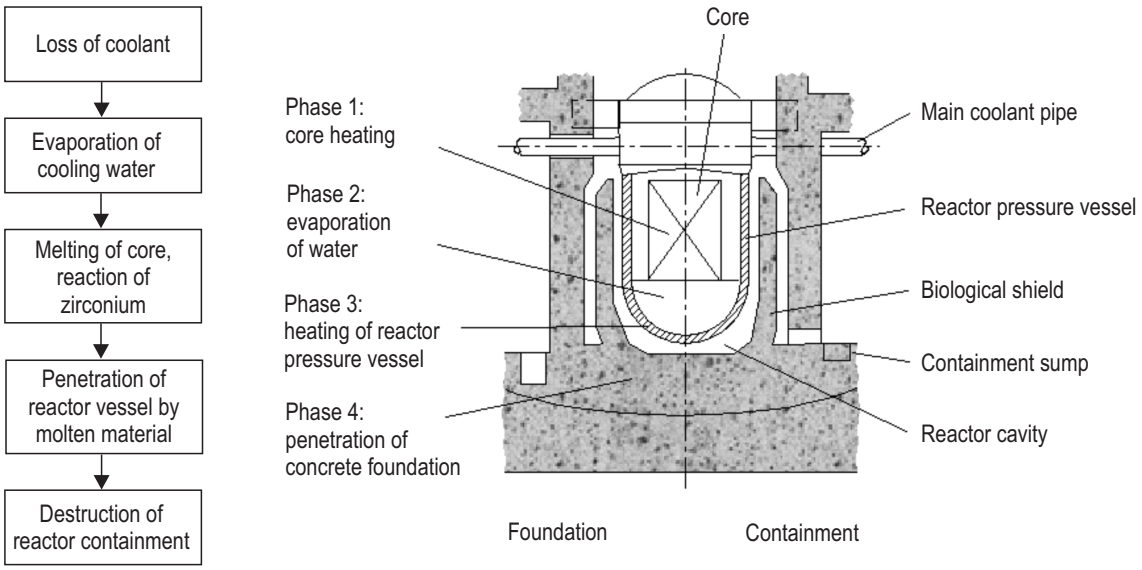


Fig. 1.139. Phases of core melt accidents in light water reactors.

Furthermore accidents are imaginable which lead to a melting of the core under full pressure inside the reactor cooling circuit. In this so-called high-pressure path, hydrogen is formed inside the core. At a sufficiently high pressure the reactor pressure vessel will burst, the discharging hydrogen will ignite and blow up the vessel parts, eventually leading to the destruction of the reactor safety building.

In order to visualize the characteristic occurrences a very simplified model can be introduced for the core melt accident. It is assumed that the decay heat, for the time being, serves to evaporate the primary circuit contents, then to melt the core and finally to produce the heat for the melting through of the reactor pressure vessel.

In reality the evaporation of water as well as the core melting would take place simultaneously. Moreover the strong exothermic heat of the zirconium-steam reaction has to be considered in the heat balance. An approximation for the first three stages leads to the following equations which already give a rough picture of the characteristic time spans of the processes.

– Phase 1: water evaporation,

$$\int_0^{\tau_1} P_D(t)\cdot \mathrm{d}t = m_w \cdot r, \quad P_D(t) \approx P_0 \cdot 6.22\times 10^{-2} t^{-0.2} \qquad (t \text{ in s}). \tag{1.283}$$

– Phase 2: melting of the core,

$$\begin{aligned}\int_{\tau_1}^{\tau_2} P_D(t)\cdot \mathrm{d}t = {} & m_{UO_2}\cdot\left[c_{UO_2}\cdot(T_{m_{UO_2}} - \overline{T}_{UO_2}) + \Delta h_{UO_2}\right] \\ & + m_s\cdot\left[c_s\cdot(T_{m_s} - \overline{T}_s) + \Delta h_s\right] + m_{Zr}\cdot\left[c_{Zr}\cdot(T_{m_{Zr}} - \overline{T}_{Zr}) + \Delta h_{Zr}\right].\end{aligned} \tag{1.284}$$

– Phase 3: melting through of the reactor pressure vessel,

$$\int_{\tau_2}^{\tau_3} P_D(t)\cdot \mathrm{d}t = m_{RPV}\cdot\left(c_s\cdot(T_{m_s} - \overline{T}_s) + \Delta h_s\right)\cdot \varepsilon\,, \tag{1.285}$$

with the decay heat power, P_D, the thermal reactor power prior to the shutdown, P_0, the mass of the evaporated water, m_w, the evaporation enthalpy of water, r, the specific heat, c, the melting point, T_m, the average temperature prior to the accident, $\overline{T}$, the melting heat, Δh, and the share of the melting mass of the reactor pressure vessel, m_{RPV}. The indices denote fuel (UO_2), steel (s), and zirconium (Zr). Characteristic numbers for a large PWR are given in Table 1.33. For the melting process at normal pressure the evaporation heat of water is $r \approx 2400$ kJ/kg. The factor ε includes the part of the reactor pressure vessel that is influenced by the melting process ($\varepsilon < 1$).

Table 1.33. Characteristic data of a large PWR core (1300 MW (el)) for estimating time spans of core melt accidents.

Material	Mass [t]	$\overline{T}$ in operation [°C]	T_{melt} [°C]	c [kJ/(kg·K)]	Δh_{melt} [kJ/kg]
Uranium oxide	100	1000	2850	0.33	250
Zircaloy	20	400	1850	0.25	260
Steel in core	50	300	1450	0.45	260
Steel of reactor vessel	500 · 0.3	300	1450	0.45	260
Water	150				

From these numbers the following time spans can be determined: $\tau_1 = 1.2$ h, $\tau_2 - \tau_1 = 1.1$ h, $\tau_3 - \tau_2 = 20$ min. This signifies that the core melt reaches the bottom of the reactor containment after approximately 2.7 h. This time schedule matches rather accurately with results that were achieved within the scope of complicated computer models (Fig. 1.140) [89GRS]. The reasons for these time spans being very limited are the high core power density of these reactors, as well as the limited storage masses inside the core.

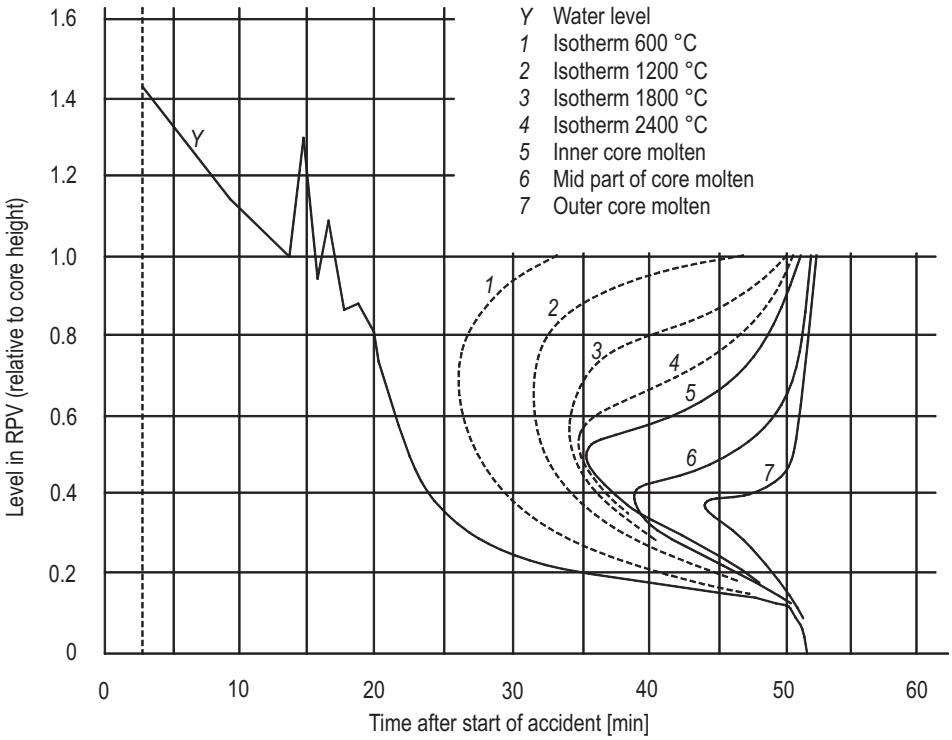


Fig. 1.140. Water level, temperature distribution, and molten areas in the core of a large LWR (PWR) in case of a core melt accident.

No such full-scale accident as described above has occurred until now. The accident at Three Mile Island (USA) in 1979 led to a partial melting of the core, which remained inside the reactor pressure vessel because the cooling circuits could be put in operation again. Figure 1.141 shows the status of the reactor after the accident [92OEC]. The probabilities for the occurrence of core meltdown accidents strongly depend on the single plant considered, and obviously they differ significantly on a global scale. The worldwide figures for the complementary probability of a core meltdown range between 10^{-3} and 10^{-6} per year. Furthermore, with respect to time spans of core meltdown, incidents of several variants can be distinguished.

Table 1.34 contains some characteristic time spans for different core melt accident scenarios: It is thought that the main process is the core meltdown under low pressure (low-pressure path) – after a loss-of-coolant incident the emergency cooling chain fails. The core then melts down under low pressure.

Core meltdown under high pressure (high-pressure path) is the second possibility – after an accident with transients or in case of an accident with coolant loss through a small leak the heat-removal via the steam generators completely ceases to operate. The reactor remains under high pressure. After meltdown of the core and hydrogen formation the reactor pressure vessel may fail, the hydrogen can ignite, and blown-up vessel parts may cause severe impact on the reactor containment. Core meltdowns can also happen by an uncontrollable leak in the steam generator pipes. In this case it is assumed that the high-pressure safety induction cannot be shut down, resulting in an overfeeding of the steam generator. The cooling of the core can only be maintained as long as water reserves are still available inside the flood tanks.

The grace periods until core destruction, corresponding to the accident assumptions, are within a range of one to five hours (see Table 1.34).

Core melt frequencies for light water reactors were obtained by several probabilistic risk studies (see Fig. 1.142). Dependent on the individual plants these values lie between 10^{-3} and 10^{-6} per year. The long-term goal for new plants is less than 10^{-6} per year.

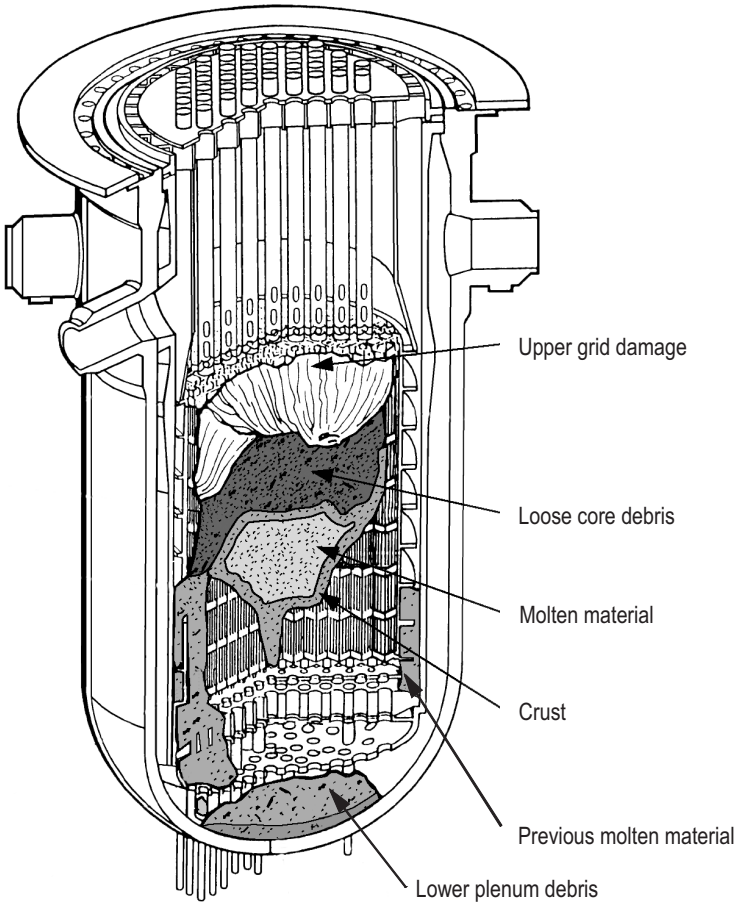


Fig. 1.141. Status of the core in Three Mile Island (USA) after a partial core melt accident (PWR, 1979).

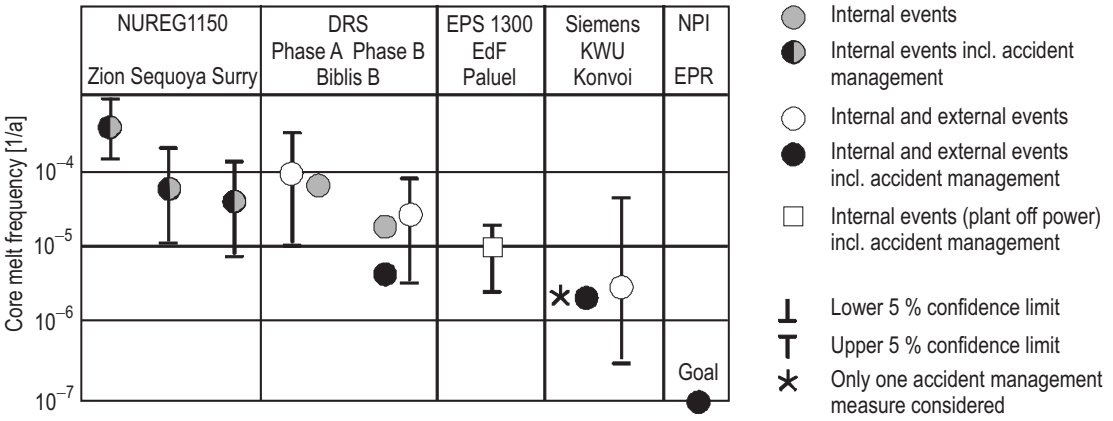


Fig. 1.142. Core melt frequencies of light water reactors from probabilistic analyses.

Table 1.34. Time spans for different core meltdown scenarios of a PWR.

Accident scenario	Start of core melting [min]	Failure of RPV [min]
Core meltdown (low pressure)	55	120
Core meltdown (high pressure)	110	140
Core meltdown (low pressure) [a])	330	410
Core meltdown (as a consequence of rupture of steam generator pipe)	540	710

[a]) Accident management included: transition of high-pressure path to low-pressure path by active depressurization of the primary circuit.

1.4.2 Consequences of core melt accidents

As a consequence of core meltdown accidents some distinct effects have to be considered, which have a substantial effect on the integrity of the reactor containment. The occurrences that need to be pointed out are presented in Fig. 1.143.

With the inner pressure being too high, failure of the reactor containment is possible. In a PWR a pressure of about 8.0 bar would be reached inside the reactor containment after some four days, leading to a high risk of failure of this component. The pressure is determined by the formation of steam, hydrogen, CO_2 and CO (Fig. 1.144). This overpressure can be reduced by specific pressure relief and filtering. Figure 1.145 shows a system which is already introduced to light water reactors at present. Solid fission products, iodine and aerosols are filtered out with the help of suitable beds of gravel, of water pools or of metal fiber filters at a separation rate of more than 99 %. Noble gases pass these systems without being retained.

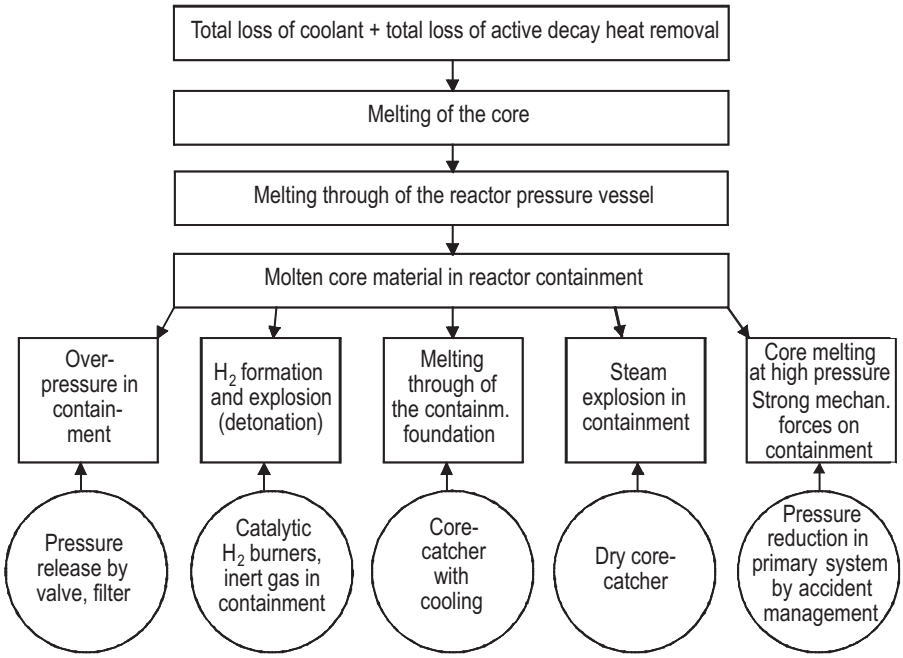


Fig. 1.143. Failure of the reactor containment due to different reasons after core melt accidents.

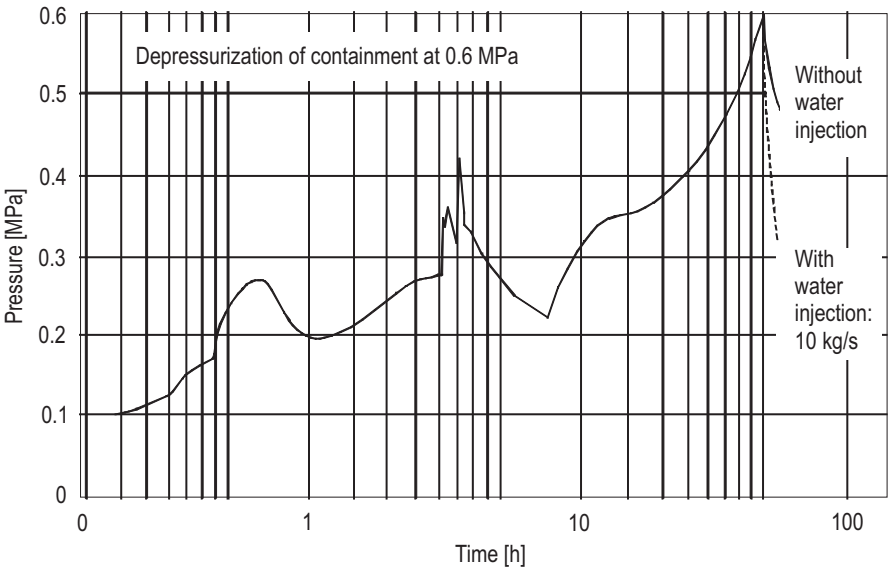


Fig. 1.144. Pressure rise inside the reactor containment after a core melt accident [87Hen].

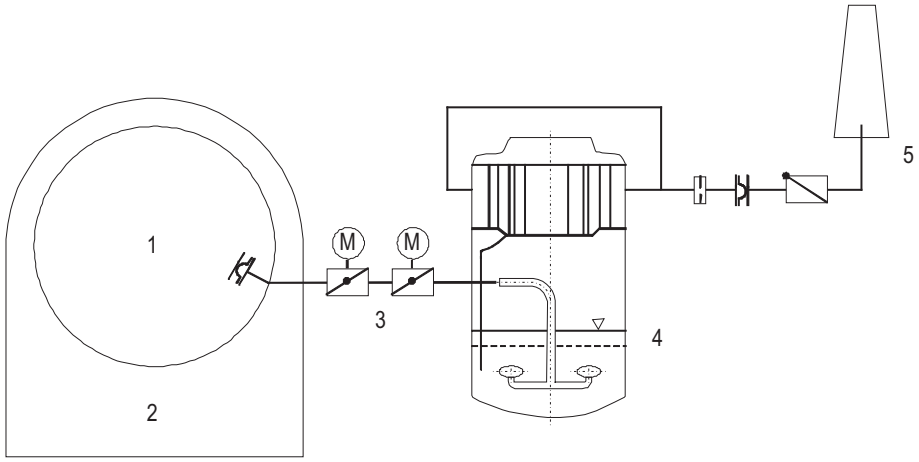


Fig. 1.145. System containing shut-off valve to avoid overpressure, and filter in a PWR (1: reactor containment, 2: outer part of reactor building, 3: pressure relief valve system, 4: filtering system, 5: stack) [93Kes].

Failure of the containment foundations by melting through of the corium is thinkable, too. The hot corium with a temperature of about 1500 °C can interact with the concrete of the foundations. These structures, depending on the strength of the foundations, can be severely penetrated within the course of several days. Fission products can reach the groundwater and cause widespread contamination. This would pose a very serious problem especially to sites on large rivers. According to the existing knowledge derived from experimental work, a penetration rate of approximately 1 m of concrete per day is to be expected. Naturally this depends on the type of concrete.

So-called core-catchers can be considered as an effective countermeasure to prevent a melting through of the foundations in future power plants. The core-catcher concept aims at maximizing the spreading of the corium in order to provide a better long-term effect of suitably arranged cooling components aimed at reducing temperatures (see Sect. 1.4.5).

Furthermore failure of the reactor containment by hydrogen explosion could happen. Hydrogen is formed during the core meltdown by interaction between the zirconium alloy of the fuel element cladding with steam, as well as by the later melting-through process of the concrete.

Hydrogen is formed from zirconium and steam according to the exothermic reaction

$$Zr + 2\ H_2O \rightarrow ZrO_2 + 2\ H_2 + \Delta H\ . \qquad (1.286)$$

The heat released during the reaction ($\Delta H = 586$ kJ/mol) is rather high. For each ton of zirconium converted, 550 m^3 (standard conditions) H_2 are produced, and an energy of approximately 7 MWh/t Zr is released in addition to the decay heat. Altogether around 10000 m^3 (standard conditions) H_2 are formed by the steam-zirconium reaction in a large PWR core (Fig. 1.146).

Hydrogen forms explosive mixtures when in contact with oxygen of the air. The explosion limits in air have been determined to be in a range of 4 to 75 volume % (Fig. 1.147). Beyond these combustion limits a self-reliant propagation of the flames inside the mixture is impossible. Beside the gas/steam ratios already mentioned the combustion limits also depend on the steam content, as well as on the original temperature of the mixture. The ignition energies needed to trigger the reaction, at a minimum of 0.02 mJ, can be considered as being very low and always available.

Under special conditions detonation combustion is also possible, in this case an overpressure of almost 15 bar can be reached, the detonation limits are in a range of 13 to 59 volume % H_2 in air. The entire explosion energy released amounts to some 2 kg TNT per 1 m^3 (standard conditions) of combustible gas. Due to these facts, severe damage to the reactor containment has to be expected in case of hydrogen explosions or even hydrogen detonations. The formation of the hydrogen from the zirconium reaction happens within a time span of less than one hour, because the reaction rate is very high.

As countermeasure against the hydrogen-related problems inside the reactor containment in BWRs almost the complete containment was made inert. As an alternative, inside some PWRs catalytic recombiners were installed, which can recombine 150 m^3 (standard conditions) of H_2 per hour in a single unit (Fig. 1.148). These systems have sufficient dimensions for small amounts of hydrogen. In the future massive failure incidents will require recombiners of significantly greater capacity, as well as precautions in order to command the huge amounts of heat resulting from the exothermic reaction.

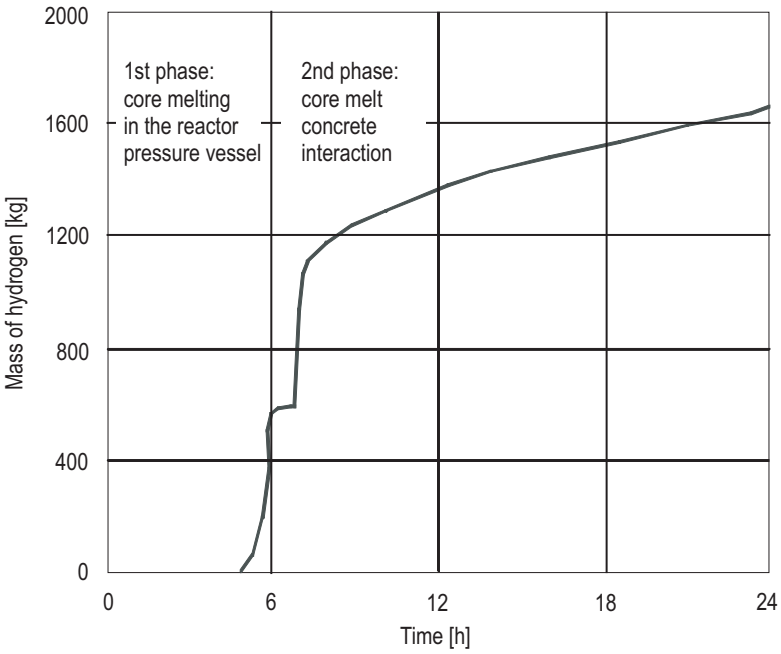


Fig. 1.146. Build-up of hydrogen inside the reactor containment (1st phase: zirconium reaction during meltdown, 2nd phase: reactions in the molten core with metals) [01Bro].

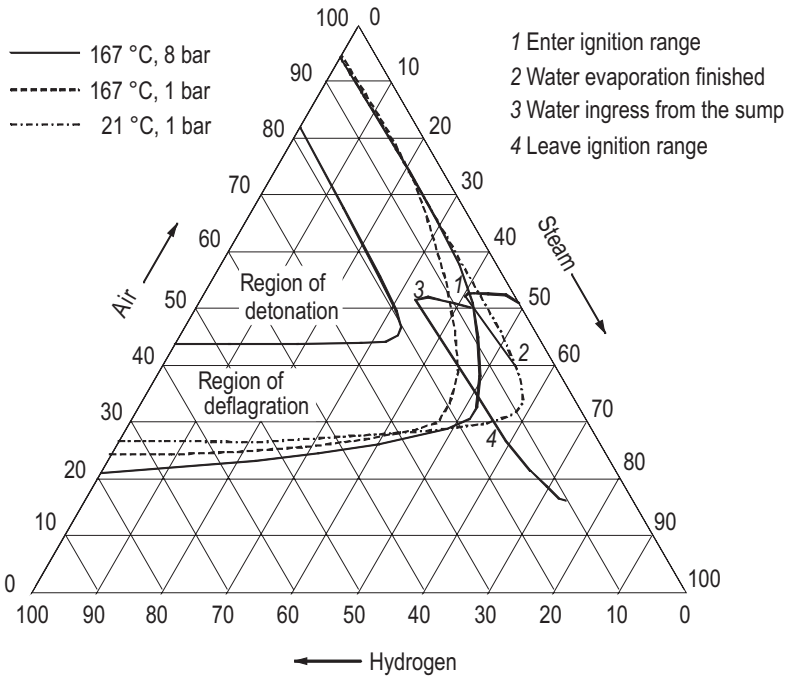


Fig. 1.147. Explosive limits of hydrogen in the system air/steam [96Bre].

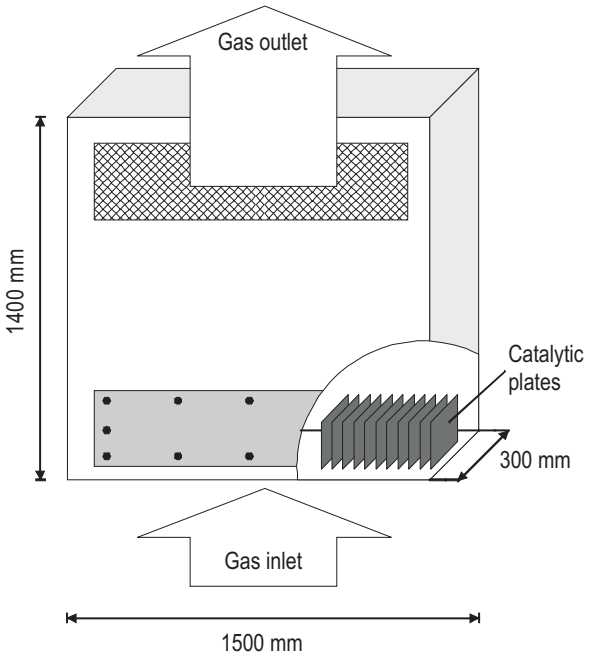


Fig. 1.148. Recombiner for hydrogen [95Kan].

Another failure mechanism of the reactor containment is caused by steam explosion: In case of contact between finely spread melt and water a sudden evaporation can occur with the initiation of a compression wave. The intensity of such an explosion depends on the share of heat stored inside the core melt which can be transformed into mechanical energy for the shock wave.

Preconditions for the occurrence of a shock wave which could damage the reactor safety building are for instance the following: sufficiently high mass of melt, intense heat transfer, small partition of the melt mass (10^{-2}…1 mm), extremely short time spans for fragmentation (10^{-2} s), and long contact times. According to the already existing results from experimental work, such a sequence is considered to be extremely unlikely.

In future reactors the bottom of the containment could be designed in such a way that a finely dispersed corium penetrating into a water sump becomes basically impossible.

A very extreme assumption on accidents is the failure of the reactor safety building after a core meltdown under high pressure: The reactor initiates a self-reliant shutdown assuming that the entire supply of electrical energy fails, including even the redundant diesel-fired emergency generator. Nevertheless the secondary-side feed of the steam generator ceases to operate.

First of all the decay heat from the primary side is conducted to the steam generators which then blow off their entire water content (approx. 250 m^3 in a 1300 MW plant) from the secondary side. After some 45 minutes the entire water is evaporated and the secondary-side heat source does not exist anymore. Pressure and temperature inside the primary circuit increase until the pressure relief valves at the pressurizer start to react. By blowing off first, the pressure within the primary circuit decreases up to the moment that the pressure relief valves close again. After that a renewed pressure increase occurs by evaporation inside the primary circuit until the armatures are opened again. This process is repeated as long as the water level inside the core is lowered to the point that parts of the core remain uncovered. The core is further heated up, which results in a melting process as well as in the formation of hydrogen. The partial melting of the core finally results in a failure of the core structure, and the core melt falls into the lower conical head of the reactor pressure vessel. After some 4 hours, with the pressure inside the primary system still being at a high level, the bottom of the cone is penetrated, the hydrogen leaks out of the primary circuit and ignites.

With the sudden pressure relief the vessel supports become extremely stressed. A massive energy and mass input from the reactor pressure vessel into the safety vessel takes place. If the actuation pressure of the hydraulic accumulator gets undershot, the water content can flood into the core melt. Consequently a further strain upon the safety vessel by evaporation of the hydraulic accumulator occurs. It needs to be examined whether the support of the reactor pressure vessel can fail in such a way that the upper part of the reactor pressure vessel can penetrate the containment on the upper side. A premature release of large amounts of fission products into the environment would be the consequence of such a core meltdown incident with high-pressure failure.

As counter measures against the high-pressure path, it is nowadays scheduled to change the incident progression (high-pressure path) into a low-pressure path through pressure relief of the primary circuit. Core meltdown frequencies have been evaluated in many probabilistic risk assessments. Characteristic numbers are 10^{-4} to 10^{-6} per year dependent on the specific plant. It is evident and easily understandable that nuclear power plants have different standards of safety all over the world. Even in the same country, the probalility varies for different plants because of their layout and construction date. It can be recognized that a continuous reduction of the core melt frequency was achieved during the last decades. Unfortunately, however, a great number of nuclear power plants with (intolerably) high core damage frequencies is still in operation worldwide. It is necessary to shut down these plants or at least to upgrade the plants to an adequate standard of nuclear safety.

The final result of risk analyses is the frequency (complementary) of early and delayed fatalities (Fig. 1.149). These studies show that the risk for private persons to be killed by a nuclear accident is very small compared to the risks arising from civilization. However, the results also indicate that a small (complementary) probability exists for large areas to be contaminated and a large number of people to be killed by radioactive products. This requires at least a resettlement for a long period combined with a loss of area. The consequences of the catastrophic accident in Chernobyl have demonstrated this problem very drastically [96Inf, 96Bir].

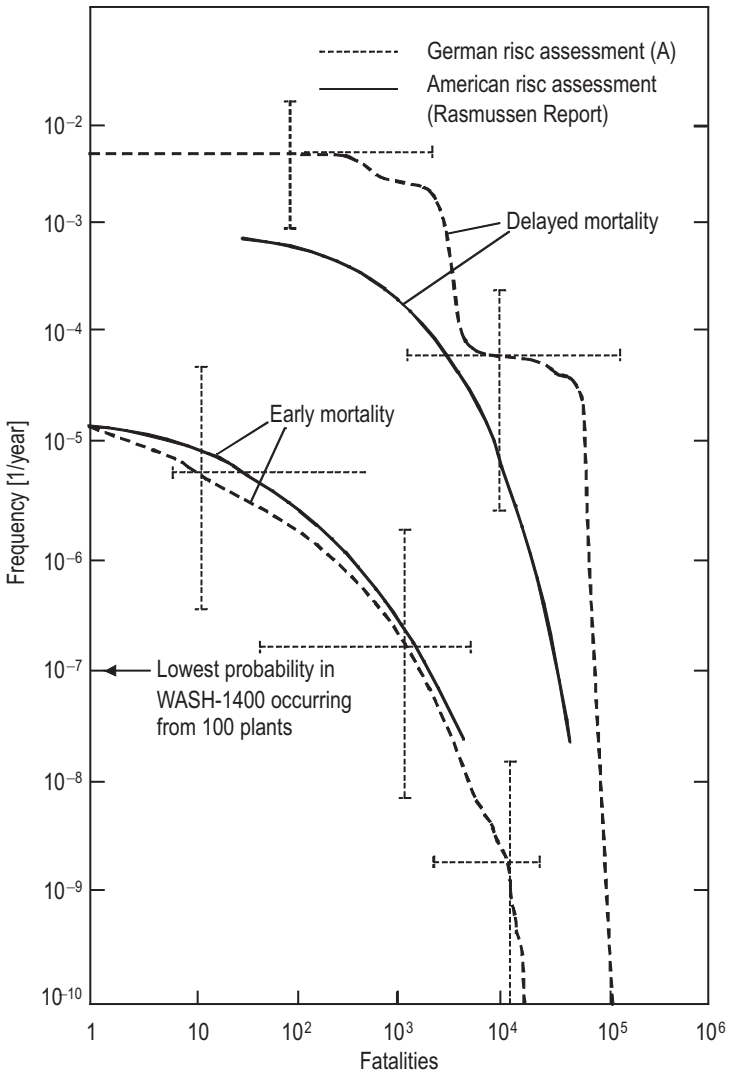


Fig. 1.149. Frequency (complementary) of fatalities from the German risk assessment for PWRs (25 plants) in comparison with the results of the American risk assessment (100 plants).

1.4.3 New safety requirements for future nuclear power plants

A further global development of nuclear energy use requires the preservation or the restoration of its acceptance within the society in some countries. This acceptance must be based on a secure prevention of catastrophic releases of radioactivity in case of severe accidents. These safety characteristics for future nuclear power plants were already stipulated by the state authorities in Germany in 1994. The modified German atomic law (§ 7 paragraph 2a) includes the following requirements for future nuclear power plants [94Bun]:

"… that also incidents, the occurrence of which is virtually excluded by measures taken to prevent damages, would not require incising measures to prevent harmful impact of ionizing radiation outside the plant itself."

In the scope of the solitary statement used for parliamentary discussion, the terms of the text are further determined. It is required that "incidents with melting of the core" can be controlled and that "evacuations are not necessary". This explicitly includes demands that no evacuations and resettlements

are allowed. The main requirements towards a future nuclear energy use therefore consist of the following points which clearly exclude potential impact enumerated in earlier nuclear risk studies: no immediate deaths among the population, no later deaths among the population, no land contamination and no related evacuations and resettlements. These are requirements of great importance that have been demanded in the field of reactor safety already for a long time.

It has to be added that these new safety characteristics of the plants have to be proved and that furthermore these requirements have to be assigned to all other activities such as supply and disposal in the nuclear energy economy.

Figure 1.150 clarifies the requirements related to a future nuclear technology: the retention of the radioactivity inside the reactor has to be efficient enough to limit the releases to the environment in case of an incident to an amount of less than 10^{-5} of the overall inventory. These new requirements mean that corresponding to the INES scale (Fig. 1.151) no incidents of the levels 5, 6 or 7 are allowed to be possible.

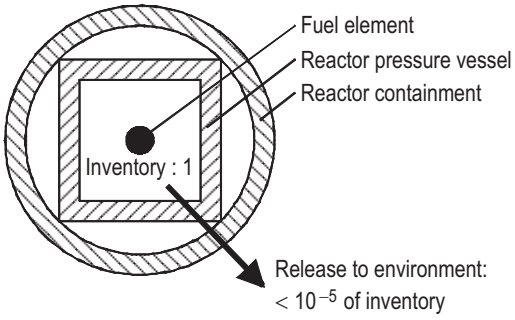


Fig. 1.150. Requirements related to fission product retention in future nuclear power plants.

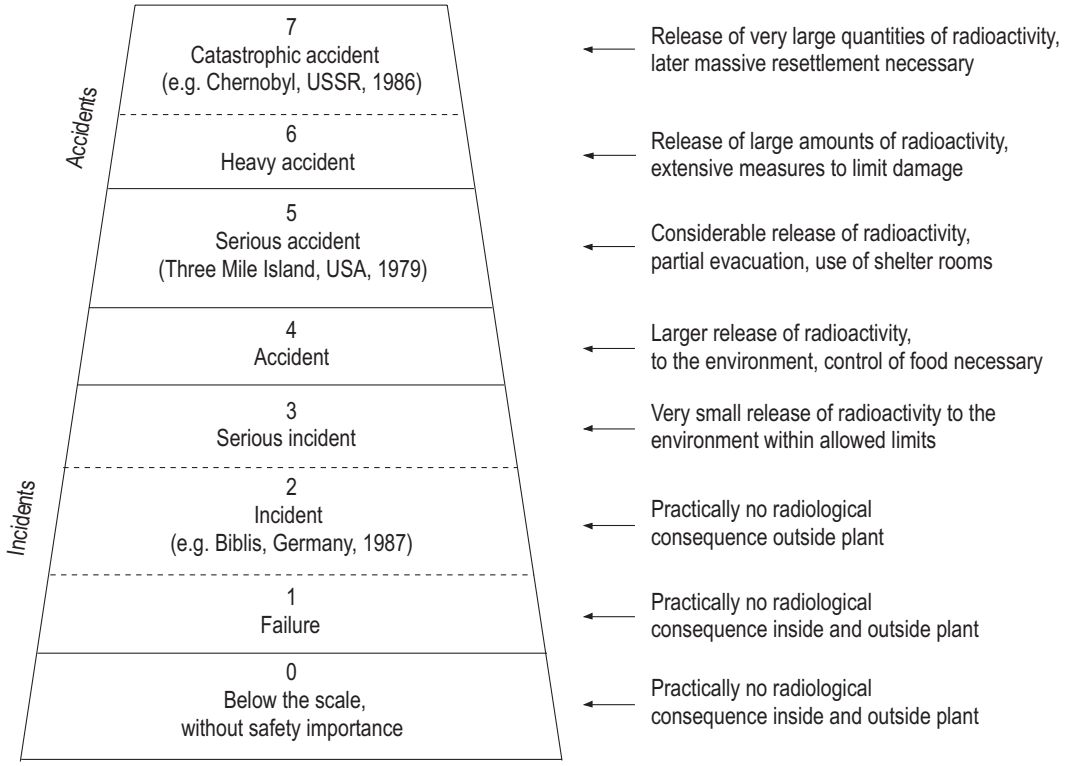


Fig. 1.151. The INES (International Event Scale) of the IAEA, 1992 [90Int].

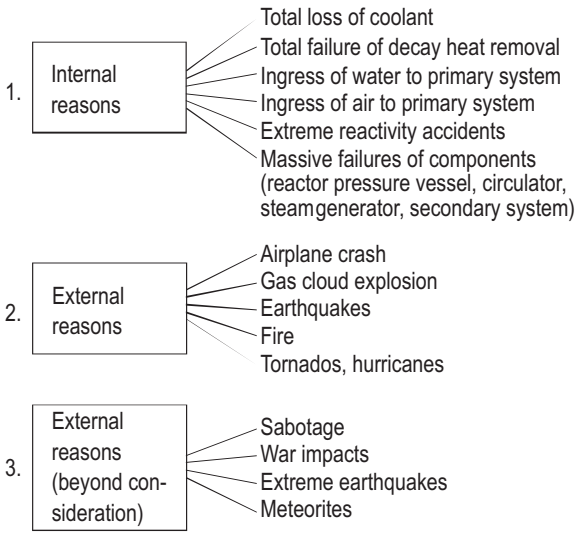


Fig. 1.152. Internal and external impact on nuclear reactors.

Of course, the details of the nuclear technology described above, also labelled as "catastrophe-free nuclear technology", have to be defined in further detail. As is shown in Fig. 1.152, incidents resulting (1) from an inner cause, or (2) from a predictable outer cause have to be distinguished. Basically the plant has to be designed to withstand the incidents mentioned above. Incidents or malfunctions of future plants are not allowed to cause any unallowed radiological impact outside the power plant itself.

Category 3 includes incidents that exceed by far the legal framework of the approval process. For this case, special examinations as well as other extraordinary measures could be necessary. For instance an underground location of the plant itself could be a reasonable option against impact from outside which has not been considered in the licensing process.

A typical distinction between accidents of category 2 and 3 is seen from earthquake assumptions. In many countries today the design against acceleration values of $a = 0.3\ g$ (3 m/s^2) is normal standard, naturally there is a small probability ($< 10^{-6}$ per year) that higher acceleration values than 0.3 g can occur. To be protected even against this extreme impact the design of nuclear power plants must be very robust, for instance the safety functions must not depend on machines.

1.4.4 Paths of development of new reactor systems

There is considerable worldwide effort to meet the advanced safety requirements of future power plants. To fulfill the demands of the German atomic law, two paths of development are basically imaginable (Fig. 1.153).

Current commercial light-water reactors could be upgraded in such a way that core meltdowns, which can occur in case of extreme incidents, are retained inside the reactor containment. Below the reactor pressure vessel a catcher component (core-catcher) has to be installed in order to catch and cool down the core melt. The reactor safety building has to be designed with additional strength in order to support weights and strains which can occur during the course of a core meltdown incident. The reactor safety building has to remain tight over a long period, and the radioactive materials have to be sufficiently retained, as to ensure that no unpermittable radioactive contamination can cause impact outside the plant. Therefore a reactor containment of appropriate dimensions and design is the decisive component for reactor safety, as well as for the protection of the environment. The EPR system, which is described in Sect. 1.4.5, represents such a new solution.

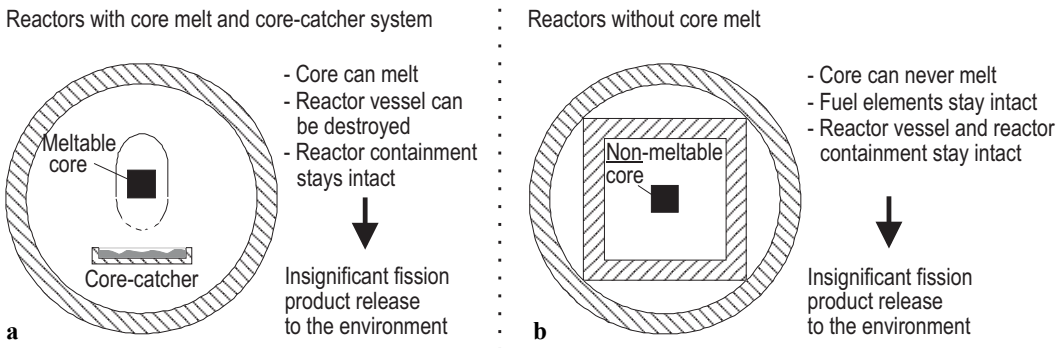


Fig. 1.153. Possible ways to meet future safety requirements: **(a)** reactor with a meltable core and core-catcher, **(b)** reactor with a non-meltable core and self-reliant decay heat removal.

As an alternative there are nuclear reactors which are designed such that a melting of the core after a complete loss of coolant and a failure of all active cooling systems is excluded by physical laws. High temperature reactors with specific design and layout of the fuel elements and the core fulfill this requirement (see Sect. 1.4.6). The decay heat is removed from the reactor exclusively by self-reliant mechanisms such as heat conduction, heat radiation and free convection of air, with the fuel elements remaining below a critical temperature. A complete failure of heat removal from the core is not possible in these plants. It has to be ensured that the fuel elements are not damaged or even destroyed by strong nuclear transients, unpermitted corrosion or mechanical strains that are too great. In the case of this reactor concept the quality of the fuel elements is of decisive importance. Apart from the fuel elements as "barriers" there are of course additional barriers like the primary circuit and the reactor building which also reduce the size of the release of radioactivity to the environment.

1.4.5 Reactor system with retention of the molten core

The European Pressurized Water Reactor (EPR) [92Hue, 99Fab, 95Bue] is designed with components to control meltdown accidents of the core as well as possible consequences resulting from such incidents because the residual risk of current commercial light-water reactors is mainly determined by core meltdown incidents and by the possible consequences. For the EPR (Fig. 1.154) it is planned to hold back an already occurred core melt inside the reactor containment, to cool it down, and thereby retain the complete fission products inventory inside the plant itself. The concept of this reactor ($P_{th} \approx 4900$ MW) is based on PWRs of the German convoi type and the French N4 design line.

Some important parameters of this plant are given in Table 1.35. The main characteristics of fuel element and core design, as well as the main thermodynamic data of the primary circuit and the steam cycle are similar to those of well-known PWR technology.

Table 1.35. Some important parameters of the EPR.

Thermal power	4900 MW
Electrical power	1700 MW
Primary coolant temperatures	292.5 → 330 °C
Primary pressure	160 bar
Secondary steam conditions	74.6 bar/280 °C
Average burn-up	60000 MW d/t
Max. enrichment	5 weight % U-235

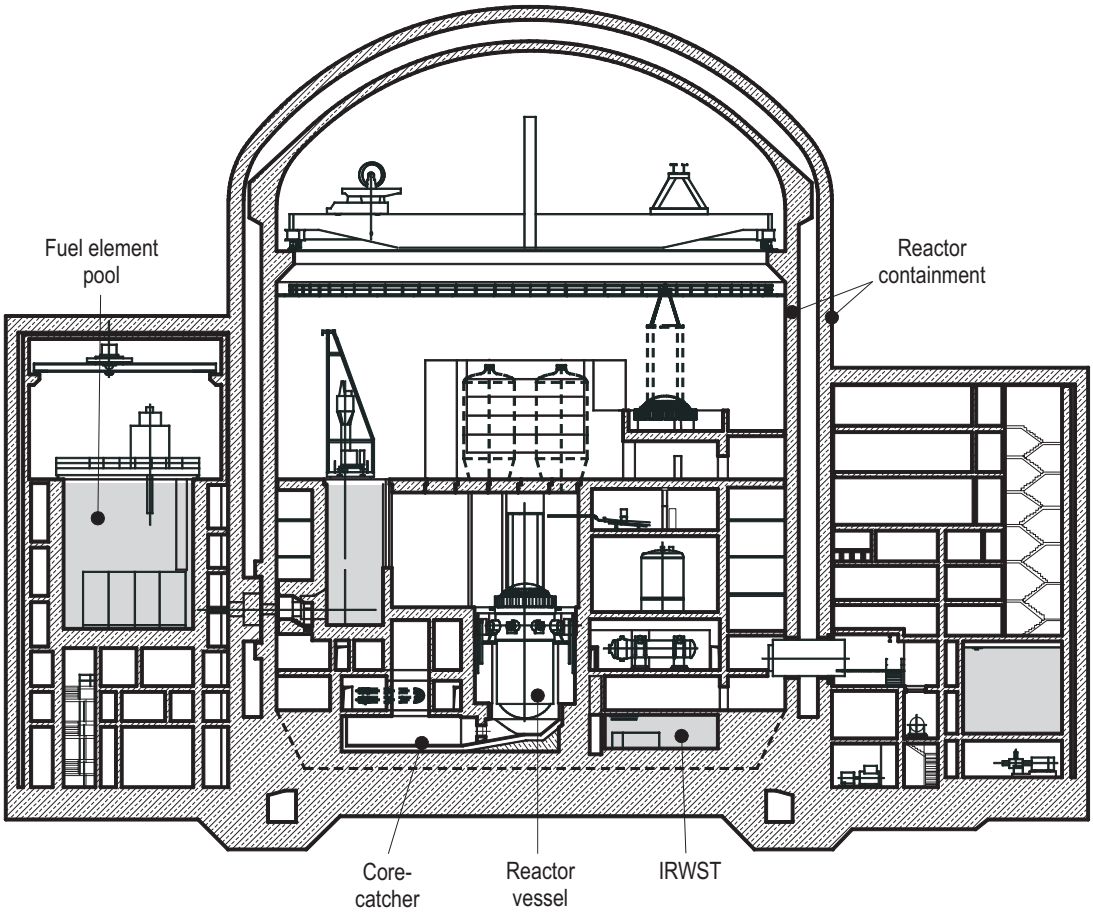


Fig. 1.154. Primary system of the EPR (1700 MW (el), IRWST denotes the internal refueling water storage tank). (Courtesy of Framatome ANP GmbH).

The decay-heat removal systems and the emergency cooling systems are supposed to include high-pressure and low-pressure feed systems with high redundancy, which were introduced for PWRs already long ago. For the reactor containment a double-walled construction is intended. The storage building for the burned fuel elements is arranged outside the containment in order to ensure free access to it in case of an incident. Thereby it is ensured that the unloaded fuel elements are coolable in the long term even after the most serious incidents.

The implementation of a water-cooled core-catcher at the bottom of the reactor containment (see Fig. 1.155) is planned as well. Additional safety installations are intended to exclude negative impact of a core meltdown on the environment. First of all there are components for pressure relief of the reactor containment, which are meant to achieve a decrease of the increased inner pressure after a couple of days in order to protect the containment against structural failure. This is already proven technology today. Linked incident-filters largely retain the fission products which mainly consist of solid fission products and aerosols.

Noble gases pass through the filters and are released to the environment via the chimney. The steam generators can be supplied with additional water or be pressure-relieved in case of an incident. Feedings via external auxiliary units into the secondary circuit are possible in order to raise the availability of the heat sink further.

The primary circuit can be pressure-relieved by opening the valves of the pressure vessel. Thereby the so-called high-pressure path can be transformed into a low-pressure path and the containment can be protected from a quick failure. Water can be fed into the primary circuit from additional water storage tanks.

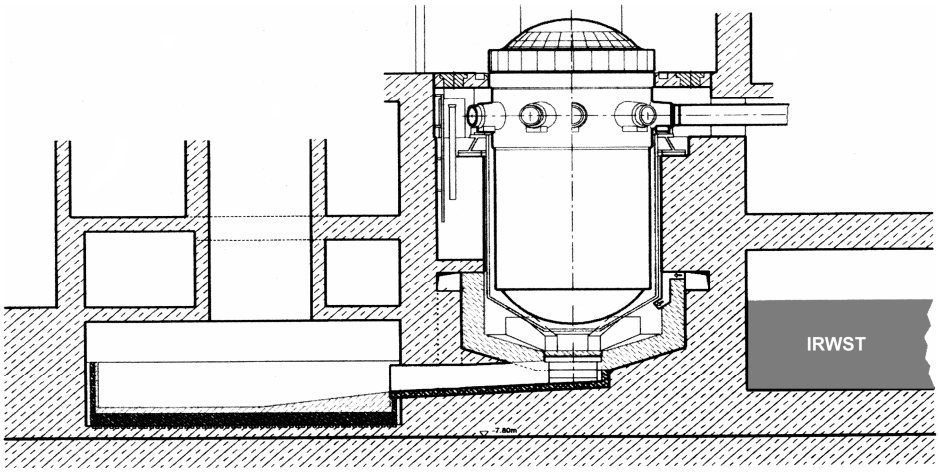


Fig. 1.155. RPV and core-catcher of the EPR. (Courtesy of Framatome ANP GmbH).

In the reactor containment, hydrogen recombiners are intended to recombine continuously the hydrogen which could be formed during the core meltdown incident from the reaction of the zirconium with the fuel element cladding. Explosive gas mixtures which could result in unallowably high pressures inside the containment shall be prevented. As an alternative an inertization of the containment is imaginable as was already introduced as a safety measure for boiling water reactors.

At the bottom of the reactor containment a new component, the so-called core-catcher, is installed in order to absorb the core melt. It has the task to cool the so-called corium efficiently over a long period. Figure 1.157 shows the course of the maximum fuel element temperatures after the beginning of the incident, assuming total failure of the active core cooling at the point $t = 0$.

Under certain incident assumptions, after 1…2 h the melting temperature of the fuel elements (2850 °C) is reached and the molten material, after melting through of the RPV's conical head, falls into the core-catcher inside which it needs to remain and be cooled for thousands of hours. With respect to the efficiency of the system, the proof will have to be provided, that interactions of the core melt with materials of the core-catcher, as well as possible steam explosions and the forming of large amounts of hydrogen inside the containment can be controlled.

The important parameters which determine the conditions of cooling down a molten corium spread over a large area can be derived from a simplified energy balance, as shown in Fig. 1.156.

Inside the corium layer there is a temperature profile with a temperature difference between the surface and the inner zone of the order

$$\Delta T_c \approx \dot{q}_c''' \cdot (S/2)^2 \cdot \lambda_c \,, \tag{1.287}$$

with $\dot{q}_c'''$ being the power density in the corium, s the thickness of the corium layer, and λ_c the heat conductivity of the corium. If direct water cooling is applied to cool the corium surface, the energy balance delivers:

$$\begin{aligned} &\rho_c \cdot c_c \cdot s \cdot A \cdot \frac{\mathrm{d}\overline{T}_c}{\mathrm{d}t} = P_{th} \cdot f_D(t) - \dot{m}_W \cdot r, \\ &f_D(t) \approx 0.062 \cdot t^{-0.2}, \end{aligned} \tag{1.288}$$

where ρ_c and c_c are the corium density and specific heat capacity, respectively; the mass flow and evaporation heat of water are denoted by $\dot{m}_W$ and r, respectively. If radiation from the surface and cooling at the bottom side of the corium are the main cooling processes one obtains:

$$\rho_c \cdot c_c \cdot s \cdot A \cdot \frac{\mathrm{d}\overline{T}_c}{\mathrm{d}t} = P_{th} \cdot f_D(t) - \dot{Q}_1 - \dot{Q}_2 \,. \tag{1.289}$$

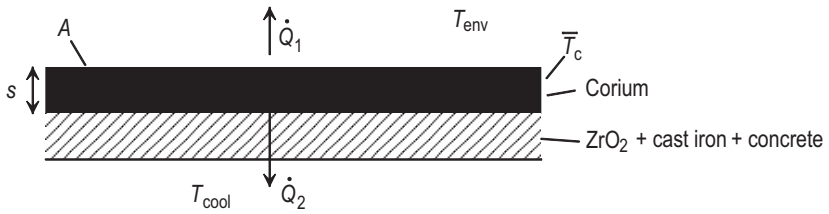


Fig. 1.156. Schematic arrangement of corium on the bottom structure of a core-catcher consisting of ZrO_2 + cast iron + concrete.

For the heat losses from the corium the following assumptions can be made:

$$\dot{Q}_1 = A \cdot \varepsilon \cdot C_s \cdot \left[(\overline{T}_c/100)^4 - (T_{env}/100)^4 \right],$$
$$\dot{Q}_2 = \overline{k} \cdot A \cdot \left(\overline{T}_c - T_{cool} \right). \tag{1.290}$$

For the decay heat production the approximation

$$P_D(t) \approx P_{th} \cdot 0.062 \cdot t^{-0.2} \qquad (t \text{ in s}) \tag{1.291}$$

is valid after a longer time. The differential equation,

$$\frac{d\overline{T}_c}{dt} = \frac{P_{th} \cdot 0.06}{\rho_c \cdot c_s} \cdot \frac{1}{s \cdot A} \cdot t^{-0.2} - \frac{\varepsilon \cdot C_s}{10^8 \, \rho_c \cdot c_c} \cdot \frac{1}{s} \left(\overline{T}_c^4 - T_{env.}^4 \right) - \frac{\overline{k}}{\rho_c \cdot c_c} \cdot \frac{1}{s} \left(\overline{T}_c - T_{cool} \right), \tag{1.292}$$

can be solved and shows the typical time dependence displayed in Fig. 1.157.

From (1.292) it is evident that the corium layer should be as thin as possible and that the spreading area should be as large as possible. In addition, the heat transfer number, $\overline{k}$, should be as large as possible to remove the heat to the cooling system effectively.

The currently favored EPR concept of a core-catcher is shown in Fig. 1.155. The entire amount of corium (approx. 150 to 250 t consisting of UO_2, ZrO_2, steel, fission products) is supposed to spread out on a surface of some 150 to 200 m^2. The layer thickness amounts to some 10 cm. From above the melt is supposed to be cooled in direct contact with water until it finally becomes solid. As far as aspects of construction are concerned, the core-catcher could consist of large plates of cast iron. An upper layer of ZrO_2 could take care of a spatial separation of the melt material and the cooling water in the lower part and thereby enable long-time cooling without direct contact to water.

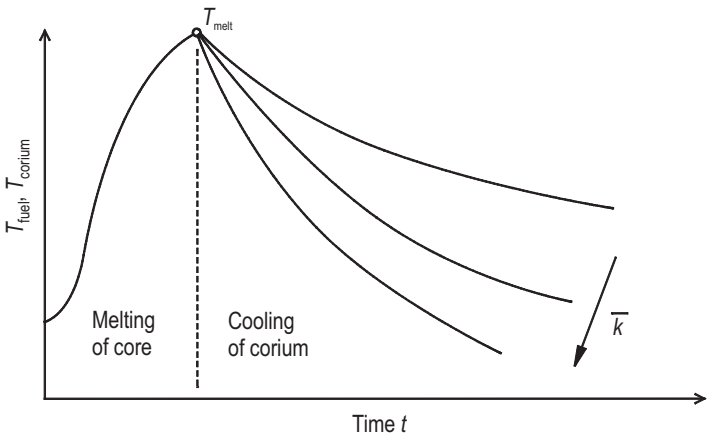


Fig. 1.157. Development of temperatures in the core, T_{fuel}, and in the core-catcher, T_{corium}, after a core melt accident.

This overall significantly modified pressurized-water reactor system can meet the requirements of a future catastrophe-free nuclear technology if the following demands can be fulfilled and demonstrated within the scope of large-scale experiments.

- Function of the core-catcher: Spreading of the corium on the surface as well as sufficient cooling of the core melt inside the installation over a long period have to be demonstrated. The preservation of the basement structure of the reactor containment over a long period has to be ensured.
- Ability to control the hydrogen formed during the incident: Immediate recombination of the H_2, or absorption of burdens from hydrogen detonations potentially occuring inside the reactor safety building with the help of corresponding pressure-tight inner components has to be proven.
- Exclusion of or ability to control a bursting of the reactor pressure vessel: This can be achieved, for instance by a burst protection around the reactor pressure vessel or by pre-stressed components for the reactor pressure vessel.
- Exclusion of or ability to control steam explosions: This can be achieved by a suited design of the containment bottom. This means that water, which has flown into the reactor swamp during the emergency cooling stage, has to be protected from contact with the molten and possibly finely-dispersed material.
- Ability to control the high-pressure core meltdown: With the help of accident-management measures the primary circuit gets pressure-relieved in time and at a high level of reliability.
- The aim to guarantee an integral tightness of the reactor safety building over long periods after a severe incident: This requires a great deal of work to provide the necessary proof. According to the German atomic law only less than 10^{-5} of the radioactive inventory is allowed to be released to the environment in the long term. Besides a double-walled design of the reactor containment, this requires distinct solutions concerning the construction of the reactor safety building.

The entire concept of the EPR is currently being planned in detail. First of all it has to be validated by experimental research and then be demonstrated by help of a prototype plant. Hitherto the cost analysis of the producers shows that in spite of all the additional safety components the investment costs remain in an acceptable range, considering that in the long term this technology will have to be competitive with power plants which are operated using hard coal or natural gas from the world market.

1.4.6 Principles of inherently safe reactors without core melt

A reactor core can never melt and cannot be destroyed by other accidents if four principles of stability are fulfilled (see Fig. 1.158): These are nuclear, thermal, chemical and mechanical stability. In this case, if the fuel temperatures are limited to allowed values by a suited design, the fission products are retained inside the fuel elements, and practically no radiological consequences occur outside the nuclear power plant.

With respect to the thermal stability demanded, the decay heat has to be removed from the core in a self-reliant way without the necessity of any applications of auxiliary machinery. The fuel temperatures must not exceed the allowed limits, ensuring that the fission products and fission material remain inside the fuel elements in all possible incident cases.

Nuclear stability means that in case of all imaginable incidents a self-reliant limitation of the nuclear power and the fuel element temperatures takes place. The fuel temperature, too, must stay below allowed values in these incidents.

The principle of chemical stability includes that the fuel elements are either designed to withstand any corrosion by media entering the primary system, or the primary circuit is designed in a way that guarantees that a massive assault of corrosive media from outside can be excluded. The fuel elements structures therefore remain intact in all incident cases.

Concluding, the principle of mechanical stability requires that no mechanical impact from an inner or predictable outer cause is possible which could alter the consistence or dimensions of the core. The principles enumerated are obviously feasible for a high temperature reactor (HTR) of suitable design and dimensions (see Sect. 1.4.7).

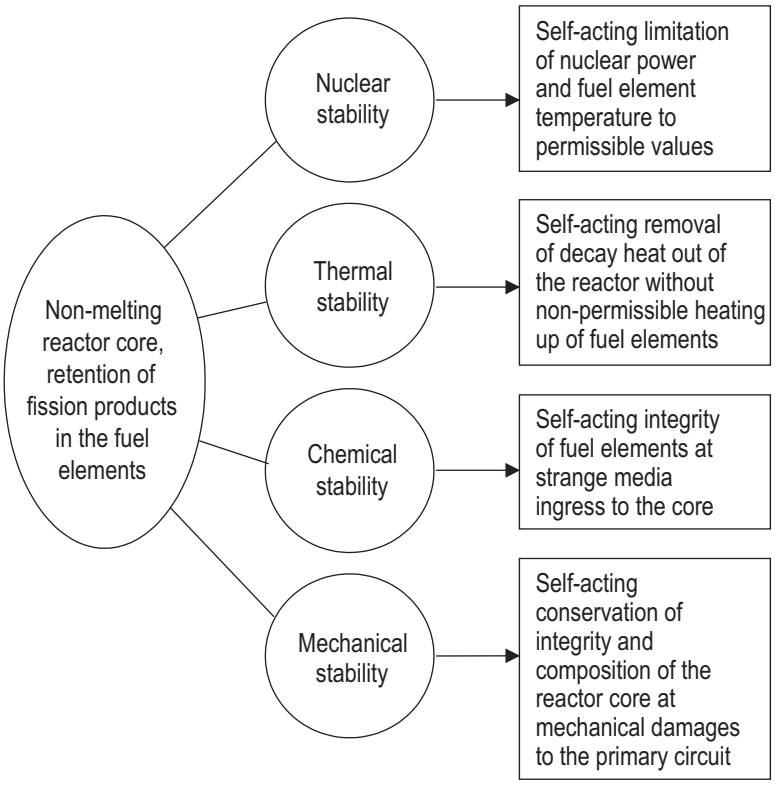


Fig. 1.158. Principles of inherently safe reactors and four principles of stability.

A necessary precondition for this behavior of the reactor is that the fuel elements can retain the fissile material and fission products up to high temperatures. In Sect. 1.4.7 coated-particle fuel will be explained, which is suited for this requirement.

In order to explain the concept of inherently safe reactors, first the principle of thermal stability can be described and mathematically assessed rather easily, and the important parameters can be identified. Referring to a very simplified model for the core area as shown in Fig. 1.159 the following energy balance equation can be set up:

$$\bar{\rho} \cdot \bar{c} \cdot V_c \frac{d\bar{T}}{dt} = P_D(t) - \dot{m}_c \cdot c_c (T_o - T_i) - \dot{m}_w \cdot r - \dot{Q}_v \,. \tag{1.293}$$

Here, $\bar{\rho}$ represents the material density, $\bar{c}$ the specific heat capacity in the core, V_c is the core volume, $\bar{T}$ is the average fuel temperature in the core, P_D denotes the decay heat power, $\dot{m}_w \cdot r$ includes the evaporation of water, and $\dot{Q}_v$ characterizes the heat losses from the surface of the core:

$$\dot{Q}_v = \bar{k} \cdot A_c \cdot (\bar{T} - T_u) \,, \tag{1.294}$$

where $\bar{k}$ is an average heat transport coefficient which covers the effective heat conductivity in the core, the heat transport in the reactor structures, and the heat transfer from the surface of the reactor vessel to the environment; A_c denotes the surface area of the reactor vessel, and T_u is the temperature of the environment. The decay heat production is described by the approximation (1.291), given a sufficiently long operation of the reactor.

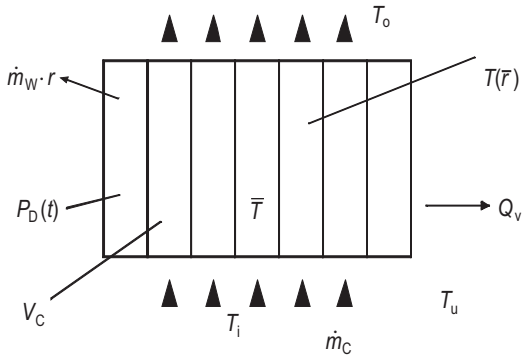


Fig. 1.159. Model for the principal possibilities of decay heat removal from the core (heat balance of the core).

Altogether the differential equation describing the time-dependent average fuel temperature inside the core has the form:

$$\frac{\mathrm{d}\bar{T}}{\mathrm{d}t} = \frac{P_{\mathrm{th}}}{\bar{\rho}\cdot\bar{c}\cdot V_{\mathrm{c}}}\cdot 0.062\cdot t^{-0.2} - \frac{\dot{m}_{\mathrm{c}}\cdot c_{\mathrm{c}}(T_{\mathrm{o}} - T_{\mathrm{i}})}{\bar{\rho}\cdot\bar{c}\cdot V_{\mathrm{c}}} - \frac{\dot{m}_{\mathrm{w}}\cdot r}{\bar{\rho}\cdot\bar{c}\cdot V_{\mathrm{c}}} - \frac{\bar{k}\cdot A_{\mathrm{c}}}{\bar{\rho}\cdot\bar{c}\cdot V_{\mathrm{c}}}(\bar{T} - T_{\mathrm{u}}) \ . \tag{1.295}$$

Three important cases can be discussed on the basis of (1.295):

(a) The normal case is that the active decay heat system works ($\dot{m}_{\mathrm{c}} \neq 0$), therefore $\mathrm{d}\bar{T}/\mathrm{d}t \leq 0$ is possible. For the PWR the heat losses, $\dot{Q}_{\mathrm{v}}$, are very small compared to the decay heat production, P_{D}; the evaporation of water ($\dot{m}_{\mathrm{w}}\cdot r$) is excluded in the normal decay heat procedure. In total the fuel temperature is reduced by the mass flow ($\dot{m}_{\mathrm{c}}$) through the core, which is driven by a redundant and diverse decay-heat removal system. This is the normal procedure of decay heat removal for all types of reactors.

(b) The function of the active decay heat removal is not available ($\dot{m}_{\mathrm{c}}=0$). In LWRs, because of $\dot{Q}_{\mathrm{v}} << P_{\mathrm{D}}$, the temperature of the core rises up until melting, as explained in Sect. 1.4.1:

$$\frac{\mathrm{d}\bar{T}}{\mathrm{d}t} = \frac{0.062\cdot\bar{\dot{q}}'''\cdot t^{-0.2}}{\bar{\rho}\cdot\bar{c}} - \frac{\dot{m}_{\mathrm{w}}\cdot r}{\bar{\rho}\cdot\bar{c}\cdot V_{\mathrm{c}}} \ . \tag{1.296}$$

Here, $\bar{\dot{q}}''' = P_{\mathrm{th}}/V_{\mathrm{c}}$ is the average power density in the core. The fuel temperature increases up to a value of $T_{\mathrm{F}} = T_{\mathrm{melt}} = 2850$ °C. The gracetime for this process is:

$$\tau = \left\{\left[(T_{\mathrm{melt}} - \bar{T}_{\mathrm{o}}) + \frac{m_{\mathrm{w}}\cdot r}{\bar{\rho}\cdot\bar{c}\cdot V_{\mathrm{c}}}\right]\frac{0.8\cdot\bar{\rho}\cdot\bar{c}}{0.062\cdot\bar{\dot{q}}'''}\right\}^{1.25} \ . \tag{1.297}$$

This time span, τ, is in the order of 1…2 h in today's light water reactors. Large quantities of water (m_{w}) in the primary system and a smaller average power density in the core ($\bar{\dot{q}}'''$) would help to achieve longer gracetimes. The PIUS concept, which uses a very large water pool inside a prestressed concrete reactor vessel for a PWR core, is an example for this possibility.

(c) The third case is characterized as follows: There is a total failure of active decay heat removal ($\dot{m}_{\mathrm{c}}=0$), no water being available in the primary system which could be evaporated ($\dot{m}_{\mathrm{w}} = 0$), but the heat losses through the surface of the reactor pressure vessel are in the same order as the decay heat production ($\dot{Q}_{\mathrm{v}} \approx P_{\mathrm{D}}$).

The last case cannot be realized for a large LWR core (P_{th} = 3000…4000 MW) but for the core of an HTR with P_{th} = 200…400 MW. The differential equation for the fuel temperature is now:

$$\frac{d\bar{T}}{dt} = \frac{0.062 \cdot \bar{\dot{q}}'''}{\bar{\rho} \cdot \bar{c}} \cdot t^{-0.2} - \frac{\bar{k}}{\bar{\rho} \cdot \bar{c}} \cdot \frac{A_c}{V_c} (\bar{T} - T_u) . \quad (1.298)$$

This equation can be solved with the initial condition $\bar{T}(t = 0) = \bar{T}_o$. A qualitative solution is shown by curve 3 in Fig. 1.160. Curve 1 represents the normal decay heat removal corresponding to case a), curve 2 belongs to case b) for the melting accident. The decay heat in a first phase after start of the accident (loss of coolant and loss of active decay heat removal) is mainly stored inside the fuel elements and the structure of the reflectors, and in a second phase the heat is given off from the surface of the reactor vessel just by radiation, conduction and free convection of air. The maximum temperature in the core, T_{max}, must never exceed an allowed value, which is 1600 °C for spherical HTR fuel elements with TRISO-coated particles (see Sect. 1.4.7).

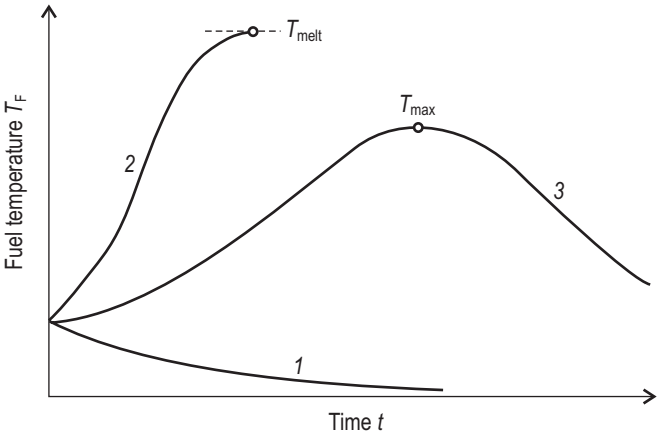


Fig. 1.160. Time dependence of fuel temperatures for different decay-heat removal scenarios.

From these considerations it can be stated that the following features of the reactor core are essential for the self-acting decay heat removal:

- low power density in the core ($\dot{q}'''$),
- high heat conductivity in the core (λ_{eff}),
- short distance for the transport of heat inside the core,
- high heat storage capability of the core ($\bar{\rho} \cdot \bar{c}$),
- high temperature stability of the core materials,
- high heat transfer numbers through the core structures ($\bar{k}$),
- high ratio of surface to volume of the core (A_c/V_c),
- permanent outer heat sink outside the reactor vessel.

Additionally there are some requirements for the integrity of the fuel elements:

- corrosion protection of the fuel elements, or limitation of the amount of corroding substances,
- no unallowed temperature rise by nuclear transients.

The differences between a reactor core which needs active cooling and a system which follows the principle of self-acting decay heat removal are pointed out in Fig. 1.161.

It can be stated that not only the overall heat balance for the fuel elements must be fulfilled, but that there must also be enough mass to store the decay heat as soon as possible in the fuel and in the surrounding material. Here, coated-particle fuel with very small diameter (0.5 mm) surrounded by a graphite matrix shows big advantages compared to pellet fuel (typical diameter 10 mm) in a canning surrounded by

water. The latter type of fuel, which is operated with high power density in the core (PWR: 100 MW/m^3), always needs water to fulfill the heat balance.

Typical time spans for the storage of decay heat in the structures of the core can be gained from a simple energy balance for the first phase after the loss-of-cooling accident:

$$m_c \cdot c_c \cdot \Delta\bar{T}_c = \int_0^{\tau_1} P_D(t) \cdot dt \,, \tag{1.299}$$

which gives for the HTR cores (with ≈ 3MW/m^3 power density) discussed below a value of τ_1 of around 20…30 h. In the second phase the decay heat must be released mainly from the surface of the reactor vessel,

$$\dot{Q}_v \approx P_D = \bar{\alpha} \cdot A \cdot (T_w - T_u) \,, \tag{1.300}$$

with the average heat transfer number, $\bar{\alpha}$, the surface of the reactor vessel, A, and the wall temperature of the vessel, T_w. After a time span of 30 h this condition can be fulfilled by a well-designed HTR with a thermal power of 200 MW, a vessel diameter of 5.5 m, and a wall temperature of around 400 °C.

Naturally the radial temperature profile in the core and the heat transport in the core structures have to be taken into account as well. The radial temperature profile is influenced especially by the effective heat conductivity in the core, λ_{eff}. The temperature difference, ΔT_c, between the surface and the center of the core depends on the average power density, $\bar{\dot{q}}_c'''$, the core radius, R, and the time factor of the decay heat function (f_D = 3 ‰ after 30 h):

$$\Delta T_c = \frac{\dot{q}_c''' \cdot R^2}{4\lambda_{eff}} \,. \tag{1.301}$$

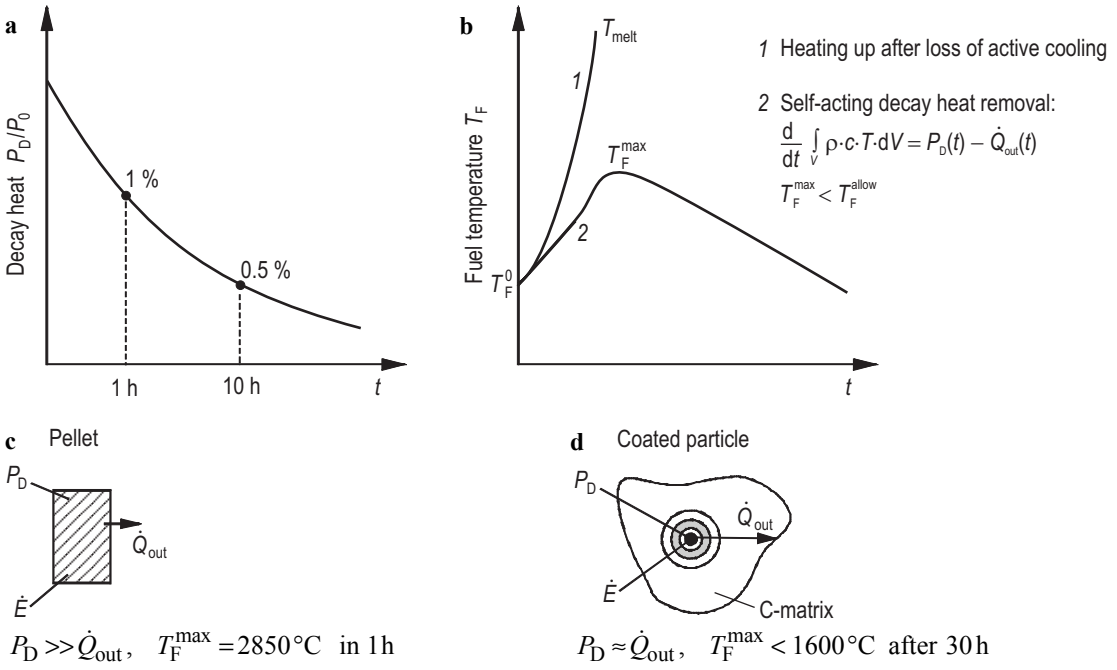


Fig. 1.161. Comparison between reactors with different principles of decay heat removal. **(a)** Time dependence of decay heat production. **(b)** Qualitative time dependence of the maximum fuel temperature in an accident (loss of coolant and total loss of active cooling) for two types of nuclear reactors. **(c)** Heat balance and conditions of fuel stability for a reactor with core melting, and **(d)** for a reactor without core melting.

All other structures inside the reactor pressure vessel cause an additional temperature difference,

$$\Delta T^* = \frac{\dot{q}''}{k^*}, \qquad \frac{1}{k^*} = \sum_i \frac{1}{\alpha_i} + \sum_j \frac{1}{\lambda_j / s_j}, \tag{1.302}$$

which has to be added to estimate the maximum temperature in the core. Here, α_j are heat transfer coefficients, λ_j are heat conductivities, and s_j are thicknesses of different layers in the radial core structure. Moreover, $\dot{q}''$ denotes the heat flux.

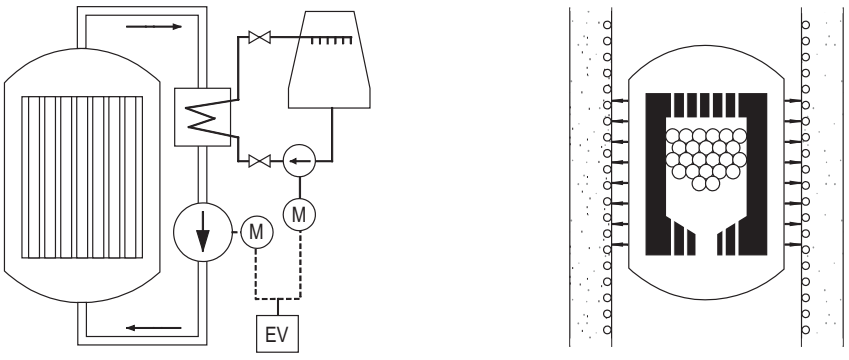
Naturally today all these calculations are carried out with 3-D computer programs including all details of space-dependent heat production and heat transport in all layers and regions of the reactor (see Sect. 1.4.7).

The differences between the principle of active cooling and the principle of inherently safe decay heat removal are are depicted in Fig. 1.162. Active systems can be carried out with a very high degree of redundancy and diversity, but there remains a very small probability that they can fail. On the other hand, inherent mechanisms of decay heat transport out of the reactor can never fail because it is only processes like heat conduction and radiation, as well as natural convection of air that are involved.

Self-acting limitation of the nuclear power and the fuel temperatures in all cases of disturbance of the neutron balance is a prerequisite for nuclear stability of the reactor system.

The neutron balance in the reactor can be disturbed by failures in the shutdown system or changes in the core composition. As an example for an extreme accident the sudden movement of all rods of the first shutdown system out of the core can be considered. In this case, when the reactivity value, $\Delta\rho_0$, is larger than the amount of delayed neutrons, β, large problems normally occur in present power reactors because of the damage to fuel elements. In the following it will be shown that nuclear technology offers solutions to resist even these extreme accidents if the design and layout of the fuel elements and the core meet the necessary conditions.

A simplified estimate on prompt excursions will demonstrate the reactor characteristics that are important in this context. It is assumed that a prompt excursion occurs with a gain of reactivity, $\Delta\rho_0$; there will be a negative feedback by a strongly negative nuclear Doppler coefficient (Γ) of the U-238 or Th-232 resonances of the breeding material inside the fuel elements. The temperature rise of the fuel will be $\Delta T(t)$, the energy produced during the transient time, t, will be $E(t)$. It is assumed that all the energy produced will stay inside the fuel (mass m_F, specific heat c_F) in a first phase. In order to simplify the calculation the kinetic equations employ the point approximation [64Mee].



a Afterheat removal by active loops

b Afterheat removal by conduction, radiation, and natural convection (self-acting)

Fig. 1.162. Concepts of decay heat removal: **(a)** present reactors, **(b)** inherently safe reactors.

For the reactivities, $\rho(t)$, the energy, $E(t)$, and the kinetic equations, one finds the following relations:

$$\rho(t)=\Delta\rho_0-\Gamma\cdot\Delta T(t)\,, \tag{1.303}$$

$$\Delta E(t)=m_{\mathrm{F}}\cdot c_{\mathrm{F}}\cdot\Delta T(t)=\int_0^t P(t')\cdot\mathrm{d}t'\,, \tag{1.304}$$

$$\bar{l}\cdot\frac{\mathrm{d}\phi}{\mathrm{d}t}=[(1-\beta)\cdot k-1]\cdot\phi+\lambda\cdot C,\qquad \frac{\mathrm{d}C}{\mathrm{d}t}=\lambda\cdot C+\beta\cdot\phi\,. \tag{1.305}$$

Because only short times are considered, the approximation $\mathrm{d}C/\mathrm{d}t = 0$ shall be assumed in the second kinetic equation, and $\phi\approx\phi(0)$ as a consequence for the flux equation ($\bar{l}$ is the lifetime of the prompt neutrons). The neutron flux, the reactor period and the reactor power are defined by:

$$\frac{\mathrm{d}\phi}{\mathrm{d}t}\;=\;\frac{k(t)-1}{l}\cdot\phi\,, \tag{1.306}$$

$$\frac{1}{\tau}\;=\;\frac{\rho(t)}{l}\;=\;\frac{\Delta\rho_0}{l}\;-\;\Gamma\cdot\frac{\Delta T(t)}{l}\,, \tag{1.307}$$

$$\frac{1}{\tau}\;=\;\frac{1}{\tau_0}-\alpha\cdot\Delta E(t);\quad \alpha\;=\;\Gamma/(l\cdot m_{\mathrm{F}}\cdot c_{\mathrm{F}})\,, \tag{1.308}$$

$$P(t)\;=\;\bar{E}_{\mathrm{f}}\cdot\Sigma_{\mathrm{f}}\cdot\phi(t)\cdot V_{\mathrm{R}}\,, \tag{1.309}$$

$$\frac{1}{P}\cdot\frac{\mathrm{d}P}{\mathrm{d}t}\;=\;\frac{1}{\tau}\,. \tag{1.310}$$

From the above relations the following integral equation for the reactor power results:

$$\frac{1}{P}\cdot\frac{\mathrm{d}P}{\mathrm{d}t}\;=\;\frac{1}{\tau_0}-\alpha\cdot\int_0^t P(t')\cdot\mathrm{d}t'\,, \tag{1.311}$$

$$\frac{\mathrm{d}P}{\mathrm{d}t}=\;P\cdot\frac{1}{\tau_0}-\alpha\cdot P\cdot\int_0^t P(t')\cdot\mathrm{d}t'\,. \tag{1.312}$$

This is the well-known Fuchs equation describing prompt nuclear transients. A solution can be found by methods known from the theory of integral equations or by numerical calculations. A relatively simple way of finding an approximate solution, which gives insight into the important parameters during the transient, is given by the following consideration starting with the expression:

$$\frac{1}{P}\cdot\frac{\mathrm{d}P}{\mathrm{d}t}\;=\;\frac{1}{\tau_0}-\alpha\cdot E\,. \tag{1.313}$$

A maximum of $P(t)$ is given by:

$$\frac{\mathrm{d}P}{\mathrm{d}t}\;=\;0\;\text{ at }P_{\max}\,. \tag{1.314}$$

With $E_{\max}$ defined as the energy at maximum power, $P_{\max}$, one obtains:

$$E_{\max}\;=\;\frac{1}{\alpha\cdot\tau_0}\,. \tag{1.315}$$

For the power depending on the energy variation in time the following equation is valid:

$$P = \frac{1}{\tau_0} \cdot E - \frac{\alpha}{2} \cdot E^2 + P_0 \,. \tag{1.316}$$

Because of $P_0 << P$ the maximum power results as:

$$P_{\max} = \frac{1}{\tau_0} \cdot E_{\max} - \frac{\alpha}{2} \cdot E_{\max}^2 = \frac{1}{2\,\alpha\,\bar{\tau}_0^2} \,. \tag{1.317}$$

If the specific boundary condition, $P = 0$, $P = P_{\max}$, is chosen, the time dependence of the energy during the transient can be calculated from the differential equation,

$$\frac{\mathrm{d}E}{\mathrm{d}t} = \frac{1}{\tau_0} \cdot E - \frac{\alpha}{2} \cdot E^2 \,, \tag{1.318}$$

with the following solutions for energy and power:

$$E(t) = \frac{2}{\alpha \cdot \tau_0} \cdot \frac{1}{1 + \mathrm{e}^{-t/\tau_0}} \,, \tag{1.319}$$

$$P(t) = \frac{\mathrm{d}E}{\mathrm{d}t} = \frac{2}{\alpha \cdot {\tau_0}^2} \cdot \frac{\mathrm{e}^{-t/{\tau_0}^2}}{(1 + \mathrm{e}^{-t/\tau_0 2})} \,. \tag{1.320}$$

A qualitative representation of the time dependence of the reactivity, the reactor power, and the temperature is given in Fig. 1.163.

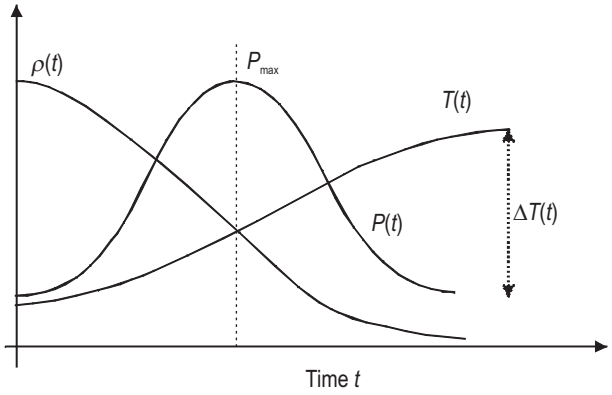


Fig. 1.163. Time dependence of reactivity, ρ, power, P, and temperature, T, during a prompt reactivity transient.

The total energy generated during the excursion is approximately:

$$E_{\text{total}} = 2 \cdot E_{\max} = \frac{2}{\alpha \cdot \tau_0} \,. \tag{1.321}$$

Substituting the characteristic parameters of the reactor, which have been mentioned above, the following results are obtained for the main parameters of the transient:

$$P_{\max} = \frac{1}{2\,\alpha \cdot \bar{\tau}_0^2} \approx \frac{m_{\mathrm{F}} \cdot c_{\mathrm{F}} \cdot \Delta {\rho_0}^2}{2 \cdot \bar{l} \cdot \Gamma} \,, \tag{1.322}$$

$$E_{\text{total}} = \frac{2}{\alpha \cdot \tau_0} \approx \frac{2 m_{\text{F}} \cdot c_{\text{F}} \cdot \Delta\rho_0}{\Gamma}, \tag{1.323}$$

$$\Delta T_{\max} \approx \frac{E_{\text{total}}}{m_{\text{F}} \cdot c_{\text{F}}} \approx \frac{\Delta\rho_0}{\Gamma}. \tag{1.324}$$

Related to these dependencies a nuclear reactor that can resist even extreme nuclear transients must fulfill the following characteristics:

- large negative reactivity feedback (esp. temperature coefficient Γ),
- small excess reactivity in the core ($\Delta\rho_0$),
- large permanent heat capacity in the core ($m_{\text{F}} \cdot c_{\text{F}}$),
- temperature-stable material in the fuel elements (ceramics),
- large span for storage of additional heat within the fuel; low fuel temperatures in normal operation,
- fission product retention in the fuel elements at higher temperatures (coated-particle fuel).

These conditions are fulfilled for the reactor design explained in the following section. This design employs entirely ceramic fuel elements and the disperged arrangement of fuel in the form of coated particles in a graphite matrix.

Figure 1.164 shows a comparison between LWR fuel (pellets) and pebble-bed HTR fuel (TRISO-coated particles) for a reactivity accident with $\rho > \beta$. The aspects to be considered are the reactivity balance and the heat balance for the fuel. In addition a fuel specification ($T_{\text{F}}^{\max} < T_{\text{F}}^{\text{allow}}$) must be fulfilled.

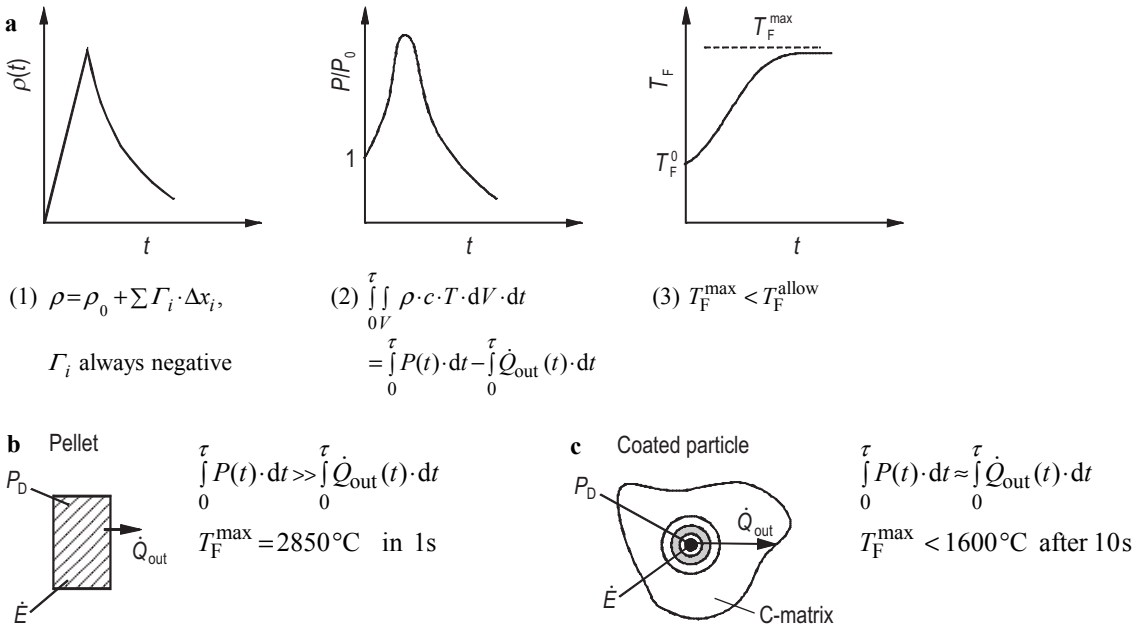


Fig. 1.164. Comparison of the behavior of nuclear reactors in case of extreme reactivity accidents. **(a)** Qualitative behavior of reactivity, power, and temperature. **(b)** Conditions of LWR fuel (pellets), and **(c)** of HTR fuel (TRISO-coated particles).

From this comparison one can state that it is not sufficient to have the negative feedback of the void coefficient or temperature coefficient. Rather it is also necessary in a first phase to store the heat, which is generated by the transient, in the fuel element; in a later phase the heat must be distributed in the core by self-acting processes (heat conduction, heat radiation, free convection) in order to limit the maximum fuel temperatures. On the one hand, the fuel must become hot as fast as possible to obtain the negative nuclear feedback, but on the other hand, the heat of the transient must be released as soon as possible from the

fuel region to realize low temperatures in these zones of the reactor. Coated particles with a very small diameter (0.5 mm) embedded in ceramic material with high heat conductivity (C, SiC) have the best preconditions to fulfill these requirements.

The damage rate of coated particles in irradiation experiments with high heat-up rates has been measured (see Fig. 1.165) and demonstrates the very good retention of fission products in these transients even if the additional energy deposition in the fuel is high [89Kug].

Overall it can be stated that it is possible to realize fuel elements and reactors which cannot be destroyed even in case of extreme reactivity accidents.

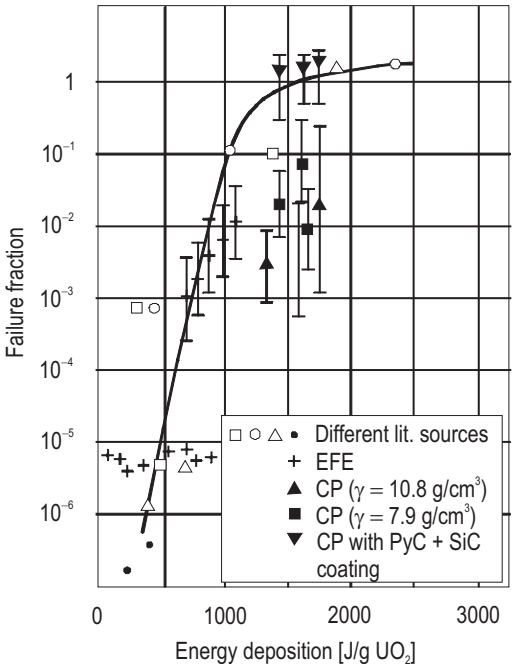


Fig. 1.165. Damage rate of TRISO-coated particles dependent on the additional energy deposition.

1.4.7 Inherently safe high temperature reactors without core melt

A reactor system which can fulfill the principles of stability, as well as the requirement of the core not melting in accidents and nearly no fission products being released is the high temperature reactor (HTR). One of its main safety-relevant features are the fuel elements. Today it is well known that TRISO-coated particle fuel with fuel kernels of very small diameter (0.5 mm) embedded in a graphite matrix retain the radioactivity almost completely for a long period up to a maximum temperature of 1600 °C in accidents. This is achieved by a very dense system of ceramic layers around the small kernel (pyrolytic graphite/silicon carbide/pyrolytic graphite). A representation of the coated particles inside the spherical fuel elements is given in Fig. 1.166. In the USA HTR fuel elements with prismatic structure have been developed.

Figure 1.167 displays typical results of heating experiments for spherical fuel elements which show that the retention of fission products in accidents up to a temperature of 1600 °C is very good [86Sch, 92Han].

The main requirement is to design a nuclear reactor in which the fuel temperature never exceeds a value of 1600 °C in all accidents and which fulfills the four principles of stability mentioned in Sect. 1.4.6.

An HTR which employs the principle of self-reliant decay heat removal and in which the maximum fuel temperature in severe accidents (loss of coolant and loss of active decay heat removal) is limited to values of less than 1600 °C has been designed by Siemens, featuring a thermal power of 200 MW [90Loh, 88Hoc, 82Reu]. The concept is depicted in Figs. 1.109a and 1.168. The cylindrical core has an

average power density of 3 MW/m^3, a diameter of 3 m and a height of 9.4 m. The fuel elements are spheres which contain the fuel in the form of TRISO-coated particles as explained above. The main technical parameters of this reactor are summarized in Table 1.36.

The core is cooled by helium flowing from top to bottom through the core, bing heated up from 250 °C to 700 °C at 60 bar. The core is arranged inside a thick graphite reflector and a core barrel. The connection to a steam generator vessel is carried out as a coaxial duct. The reactor is continuously loaded with fresh fuel elements during operation. Fuel elements which have reached the final burn-up are continuously removed from the reactor. Partly burned fuel elements are recycled through the reactor several times. By this type of fuel management no excess reactivity for burn-up compensation has to be installed in the core. Together with the always strongly negative temperature coefficient of reactivity, this is a very important feature of the reactor with respect to nuclear stability. The shut-off and control elements are arranged in holes inside the side reflector. These are rods for the first system, and small absorber balls which can fall into the holes for the second system. Mechanical interactions with the fuel elements are avoided.

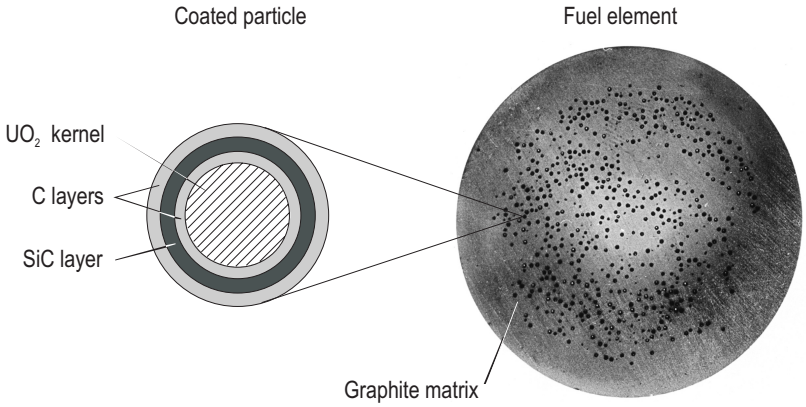


Fig. 1.166. HTR spherical fuel element (diameter: 60 mm) containing a large number (10 000…20 000) of TRISO-coated particles (diameter of vessel: 500 µm, thickness of C/SiC/C layers: 50/30/50 µm).

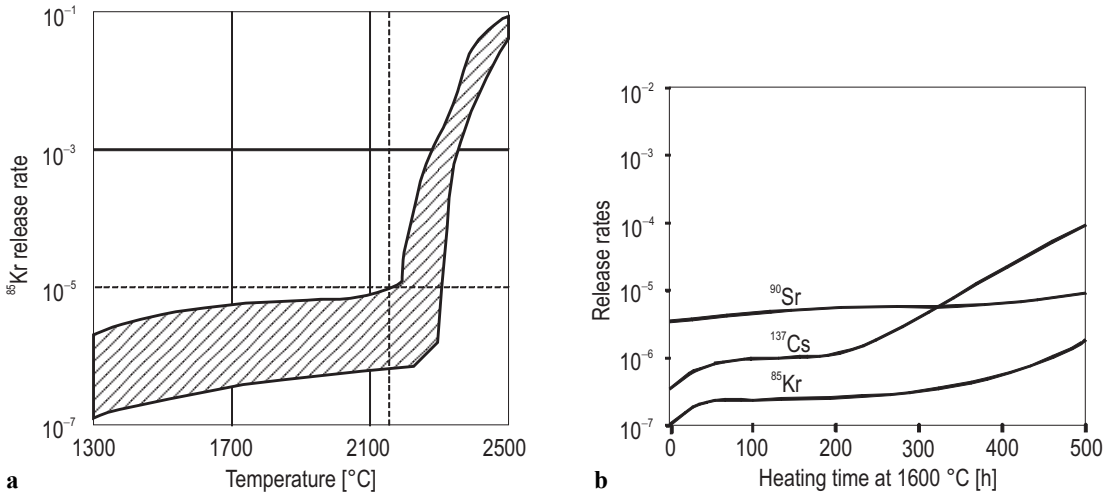


Fig. 1.167. Release rates (fractions) of important fission products from spherical fuel elements in accidents (TRISO-coated particles, LEU fuel): **(a)** release rate of ^{85}Kr vs temperature, **(b)** release rates at 1600 °C vs time.

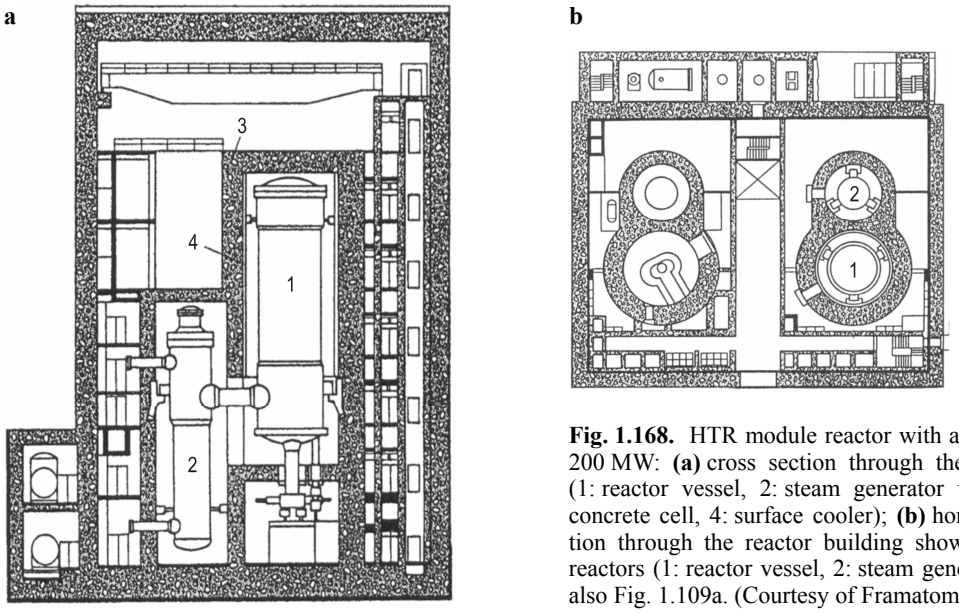


Fig. 1.168. HTR module reactor with a thermal power of 200 MW: **(a)** cross section through the reactor building (1: reactor vessel, 2: steam generator vessel, 3: primary concrete cell, 4: surface cooler); **(b)** horizontal cross section through the reactor building showing two modular reactors (1: reactor vessel, 2: steam generator vessel). See also Fig. 1.109a. (Courtesy of Framatome ANP GmbH).

The steam generator is arranged side by side to the reactor pressure vessel, the helium circulator is installed at the top of the steam generator. The primary system of the reactor is arranged inside a primary cell made from concrete. The surface of the concrete cell is equipped with a simple water cooling system, a so-called surface cooler. The reactor building must mainly protect the reactor against impact from outside. This reactor has a totally different behavior in severe accidents compared to today's reactors. The decay heat is removed from the core only by self-acting processes like heat conduction, radiation and free convection to the external surface cooling system via the reactor pressure vessel. If this water-cooled system fails as well, the decay heat is absorbed by the concrete in the primary cell around the reactor vessel and released from there to the environment.

Graphical representations of the temperatures during an accident are shown in Fig. 1.169. Even for the additional assumption that the very simple surface cooler outside the reactor vessel fails, the maximum temperature in the core stays below 1600 °C. Therefore this reactor can never melt, and the fission products stay inside the fuel elements. The reactor does not need any machine or installation for decay heat removal, with only the natural mechanisms of conduction, radiation and convection working. This behavior has been proven in the AVR reactor, as well as in large experiments, and it can be proven in the future in full scale in a real reactor. The principle of thermal stability is fulfilled for this reactor under all circumstances.

The maximum power which can be realized in a cylindrical core under these safety conditions can be estimated relatively easily. For the parabolic temperature distribution inside the core at the time t^*, at which the maximum fuel temperature occurs, one obtains:

$$\frac{1}{r}\cdot\frac{\mathrm{d}}{\mathrm{d}r}\left(r\cdot\lambda(T)\cdot\frac{\mathrm{d}T}{\mathrm{d}r}\right)=\dot{q}'''(r)\cdot f_{\mathrm{D}}(t^*)\Rightarrow\int_{T_{\mathrm{surface}}}^{T_{\max}}\lambda(T)\cdot\mathrm{d}T=\int_0^R\dot{q}'''(r)\cdot f_{\mathrm{D}}(t^*)\cdot\frac{r}{2}\cdot\mathrm{d}r,$$
$$\int_{T_{\mathrm{surface}}}^{T_{\max}}\lambda(T)\cdot\mathrm{d}T=\bar{\dot{q}}'''\cdot f_{\mathrm{D}}(t^*)\cdot\frac{R^2}{4}. \qquad (1.325)$$

The final result for the thermal power per length of the reactor is:

$$\frac{P_{\mathrm{th}}}{H}=\frac{1}{f_{\mathrm{D}}(t^*)}\cdot 4\pi\int_{T_{\mathrm{surface}}}^{T_{\max}}\lambda(T)\cdot\mathrm{d}T\,. \qquad (1.326)$$

With characteristic values of $\bar{\lambda}$ = 10 W/mK, $T_{\max}-T_{\mathrm{surface}}$ = 500 K, $f_{\mathrm{D}}(t^*)=3\times10^{-3}$ after 30 hours, the result is P_{th}/H = 20 MW/m.

Table 1.36. Technical parameters of the HTR module reactor.

Process	Steam turbine cycle
Thermal power	200 MW
Electrical power	80 MW
Net efficiency (wet cooling tower)	40 %
Average power density	3 MW/m^3
Helium pressure	60 bar
Helium inlet temp. into the core	250 °C
Helium outlet temp. from the core	700 °C
Core diameter	3 m
Core height	9.43 m
Control and shutdown rods	
1st system	6 rods
2nd system (KLAK)	18 positions
Number of fuel elements	375 000
Diameter of fuel elements	6 cm
Fuel	UO_2
Enrichment	8.6 %
Coated particle	TRISO
Burn-up	80 000 MW d/t
Steam temperature	530 °C
Steam pressure	180 bar
Feed water temperature	200 °C

Indeed the modular HTR mentioned above has a height of 9.4 m and a thermal power of 200 MW. For the release of decay heat from the surface of the reactor pressure vessel at time t^* the following relation is valid:

$$P_D = \bar{\alpha} \cdot 2\pi \cdot R_a \cdot H \cdot (T_{wall} - T_{enr.}) . \tag{1.327}$$

Therefore the maximum thermal power under this condition is:

$$\frac{P_{th}}{H} = \frac{1}{f_D(t^*)} \cdot \bar{\alpha} \cdot 2\pi \cdot R_a \cdot H (T_{wall} - T_{enr.}) . \tag{1.328}$$

With $\bar{\alpha}$ = 8 W/m^{2}K, R_a = 3 m, $T_{wall} - T_{enr.}$ = 400 K, and $f_D(t^*) = 3 \times 10^{-3}$ one obtains again P_{th}/H = 20 MW/m.

From these considerations including another temperature drop inside the core structures of 600 K the maximum temperature of 1500 °C in the hottest part of the core (see Fig. 1.169) is explained.

The requirement of self-reliant limitation of the nuclear power and the fuel element temperatures to guarantee nuclear stability is fulfilled, too. The core has virtually no excess reactivity to compensate for the burn-up; the temperature coefficients are highly negative. Also in case of ingress of water into the core there can be a negative back-coupling coefficient, because an overmoderation can be achieved by a suited fuel element design (50 % blind spheres inside the core).

In case of a fast removal of the shutdown elements from the HTR core, the latter would even withstand such sudden prompt reactivity incidents. A self-reliant limitation of the nuclear power and fuel element temperatures (< 1600 °C) are also ensured in this case. Figure 1.170 shows this behavior for the assumed incident, namely a sudden loss of the entire first shutdown system (1 s, 10 s) [94Sch].

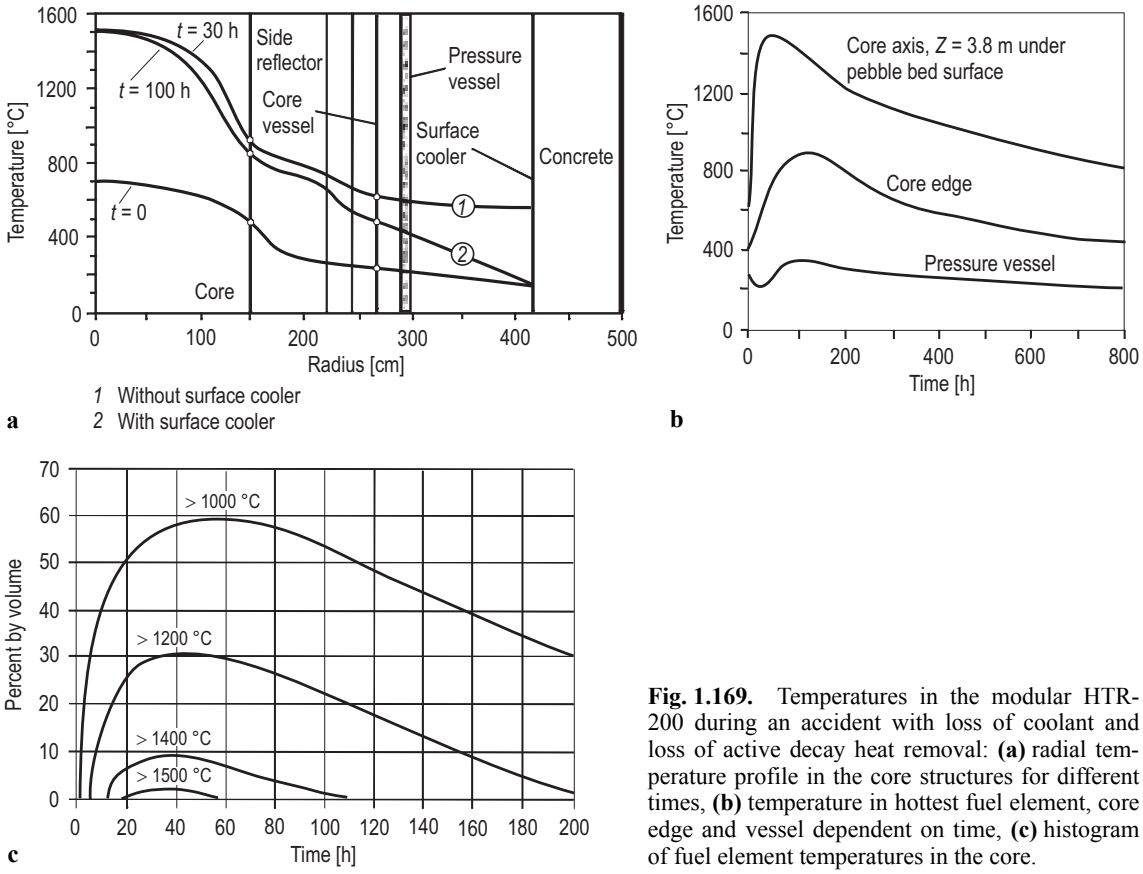


Fig. 1.169. Temperatures in the modular HTR-200 during an accident with loss of coolant and loss of active decay heat removal: **(a)** radial temperature profile in the core structures for different times, **(b)** temperature in hottest fuel element, core edge and vessel dependent on time, **(c)** histogram of fuel element temperatures in the core.

It was assumed that the reactor pressure vessel could never burst; therefore a deformation of the core and the ingress of large amounts of air would be excluded. This corresponds to the worldwide LWR safety philosophy. The power of the reactor can be higher and the assumption made before can be left out if prestressed primary enclosures are used. In contrast to the known cylindrical core with a thermal power of 200 MW, when using an annular core with an annulus width of 1.5 m and an average core power density of 3 MW/m^3 a reactor with a thermal power of 300 MW can be realized. The total primary system enclosure is prestressed with radial and axial tendons to prevent bursting of the vessel. The primary system is enclosed in an inner primary concrete cell, around which there is a reactor building to protect the reactor against impact from the outside (see Fig. 1.171). Underground siting even prevents dangerous consequences of terroristic attacks on the plant.

Cast steel is especially suited as a vessel material. Cast steel rings as well as caps and basement plates are welded gas-tight by outer welded lip seal gaskets. The application of spherical cast iron (with an inner liner) as a vessel material is also possible. Prestressed concrete vessels with conductive structures in the cylindrical area of the reactor pressure vessel are also feasible and applicable here. The thermal power of such a reactor unit according to present-day vessel technology amounts to some 300 MW [02Kug]. Larger power values can be achieved by parallel operation of several reactors. As it is the case for steam generators, gas turbine systems or the components of a combined cycle (gas/steam turbine system or cogeneration process) can also be arranged inside the second vessel. The principles of thermal stability and of nuclear stability are fulfilled as discussed above.

There is a self-reliant preservation of the structure of the fuel elements, too. The penetration of large amounts of air into the primary circuit causing unallowed damage to the fuel elements is impossible due to the burst-proof and prestressed design of the primary circuit. The possible openings of the primary system are very small (diameter 65 mm). As indicated by Fig. 1.172 the amount of air available for corro-

sion processes is limited by a tight inner concrete cell to ≈ 5000 m^3, which corresponds to a maximum corrosion of 500 kg of graphite. By intervention this number can be reduced to 100 kg.

The penetration of large amounts of water into the primary circuit is also controllable by means of the chosen concept. Normally the amount of water which can enter the primary circuit is limited by a real-time cut-off of the steam generator in case of damage and a stop to the feed water pump. Even if it is assumed that the entire water content of the secondary circuit reaches the primary circuit, in this case into the steam generator pressure vessel, no unallowed impact on the core can occur. The blower would immediately cease operation after a massive water ingress. Water flows into the core according to the relevant partial pressure and leads to a shutdown of the nuclear chain reaction. The overpressure is relieved and decreased by safety valves, rupture discs and finally by opening of welding lips of the prestressed vessel. Corrosion by hot steam inside the core is strongly limited and never threatens the 1600 °C concept.

For the self-reliant preservation of the integrity of the primary circuit systems the prestressed vessel system (see Fig. 1.173) plays a key role in the safety concept [92Fro, 00Fro]. In case of an unallowably high pressure increase, which was not decreased by safety valves or rupture discs already beforehand, a welding lip opens between the cast steel rings of the vessel and lowers the overpressure via this opening. The integrity of the reactor pressure vessel is never in jeopardy during this procedure. This fact is essential to the ensuring of nuclear and chemical stability. Damage to the core internals in case of coolant loss incidents is impossible if a prestressed vessel is used, because the pressure transients remain restricted to very small values. This overpressure behavior of prestressed pressure vessels was tested with a prestressed cast steel vessel: a two-fold overpressure was relieved by blowing off via a welding lip without any problem.

Prestressed reactor pressure vessels with the attribute to be burst-protected even in case of very high overpressure have often been realized in AGR plants and in HTR prototype plants. Figure 1.174 shows the THTR 300 reactor vessel which was built, licensed and operated (inner diameter 15 m, inner height 15 m, inner helium pressure 40 bar, prestressed axial and radial tendons, liner is insulated against 250 °C helium and cooled by water on the concrete side).

Outer impacts that are predictable and were also considered during the approval process include airplane crashes, gas cloud explosions, earthquakes, tornados and flooding. These are controlled by help of the very thick-walled outer confinement. The essential safety functions of the HTR such as shutdown, decay heat removal and enclosure of fission products are always ensured by inherently safe principles (Fig. 1.175). Some possible outer impacts which exceed the already known incidents,could mostly be controlled by underground plant location or by covering the confinement with suitable solid soil layers. These extraordinary incidents would include extreme earthquakes, sabotage, or war-related impact.

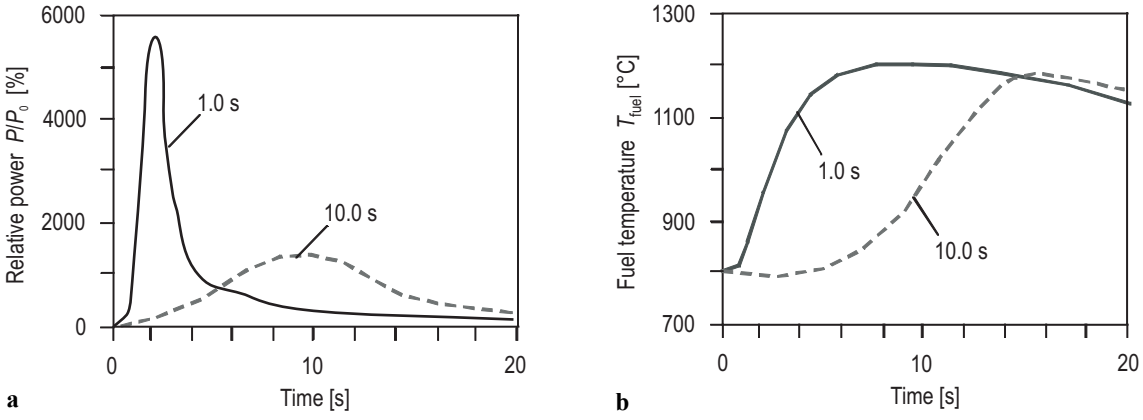


Fig. 1.170. Transient in case of total loss of the first shutdown system (ρ = 1.2 %): **(a)** thermal power versus time, **(b)** maximum fuel temperature versus time.

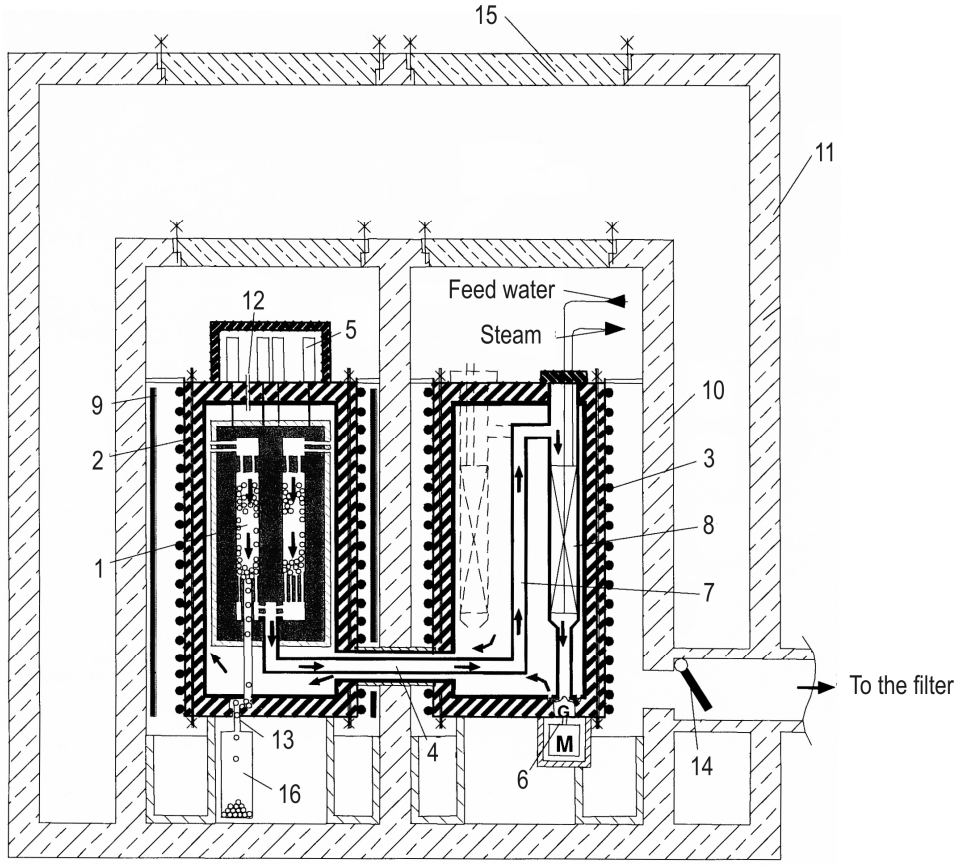


Fig. 1.171. Inherently safe HTR with prestressed primary circuit and annular core; thermal power: 300 MW, inner diameter of reactor vessel: 6 m (1: annular core, 2: prestressed reactor vessel, 3: prestressed vessel for steam generator, 4: prestressed connecting vessel, 5: control and shut-off system, 6: helium circulator, 7: hot-gas duct, 8: steam generator, 9: cell cooler, 10: inner concrete cell, 11: reactor building, 12: fuel loading device, 13: fuel discharge device, 14: flap, 15: concrete closure, 16: storage vessel for spent fuel elements).

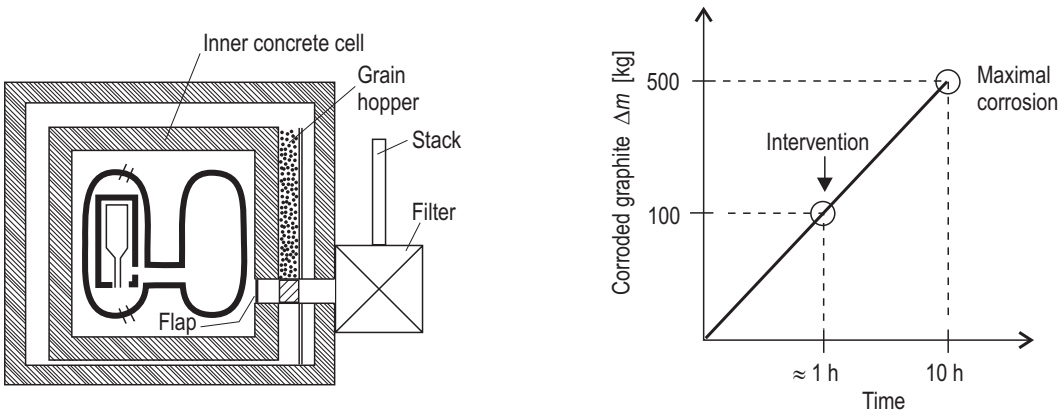


Fig. 1.172. Concept to avoid large corrosion effects in the HTR core structure in case of air ingress after a loss-of-coolant accident.

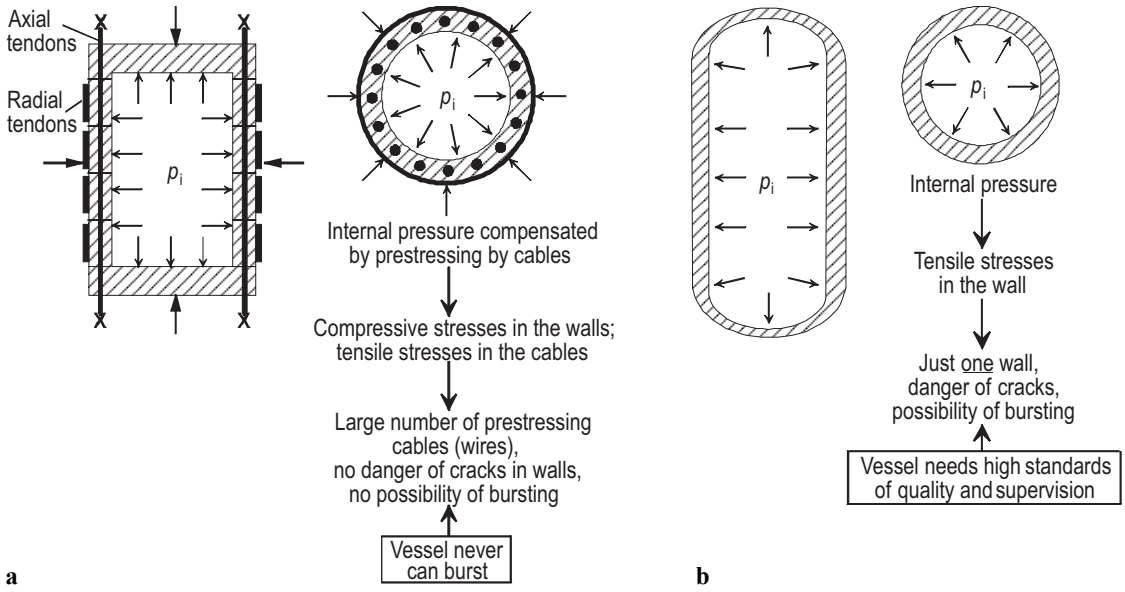


Fig. 1.173. Concept of **(a)** prestressed reactor vessel compared to **(b)** steel vessel.

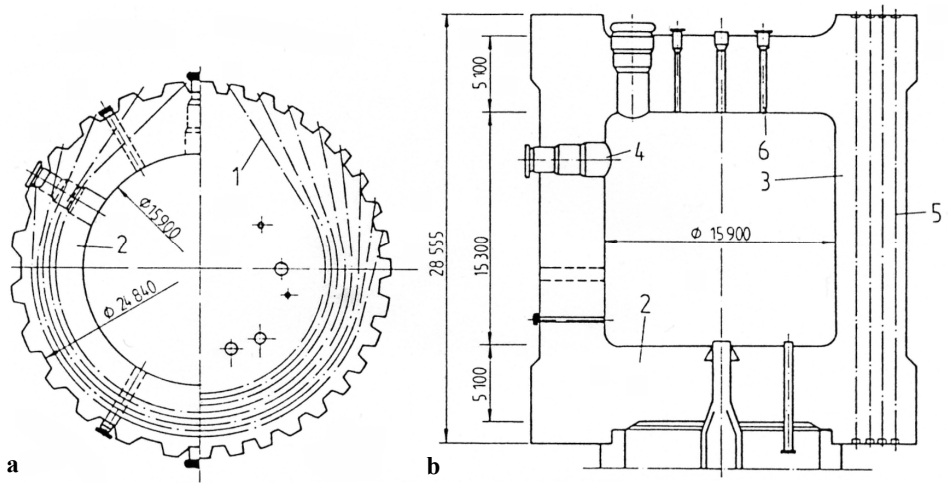


Fig. 1.174. Prestressed concrete RPV (THTR 300): **(a)** horizontal, and **(b)** vertical cross section; 1: radial tendon, 2: concrete structure, 3: liner with cooling and insulation, 4: shutter tube, 5: axial tendon, 6: penetration.

The HTR containment does not need to be as dense as that of an LWR (see Fig. 1.176). The real barrier for the fission products in all accidents are the 10^9 coated particles inside the fuel elements. They cannot be destroyed by accidents and have a very good retention capability for the fission products and fissile materials. The HTR containment building, however, has to be designed against outer impact, too (as the LWR containment). That means a wall thickness 2 m of concrete, as used e.g. for the most recently built nuclear power plants in Germany.

Overall the release rates of fission products are very small. The filter can additionally retain nearly all the solid and aerosol-type fission products which have been released from the fuel elements during heat-up accidents. Following the experience from HTR plants and the heating-up experiments for spherical fuel elements with TRISO particles, the final release to the environment is very small, as required in Sect. 1.4.6. The inventory of e.g. ^{137}Cs is 2.5×10^7 GBq, from which less than 10 GBq would be released to the environment including the action of the filter.

An overall assessment of the HTR safety characteristics leads to the result that no catastrophic incident consequences with large releases of fission products are possible. The system behavior described can be integrally demonstrated by means of a full-scale experiment [04Kug]. The decay heat can be simulated by help of electricity, and all enumerated incident situations can be modeled and verified in an experimental way. This is possible because it is a non-meltable system, so that all experiments can be carried out in a much more simple way than is the case for real corium in case of severe accidents of LWRs.

Obviously it can be demonstrated for the HTR that a catastrophe-free nuclear technology is possible.

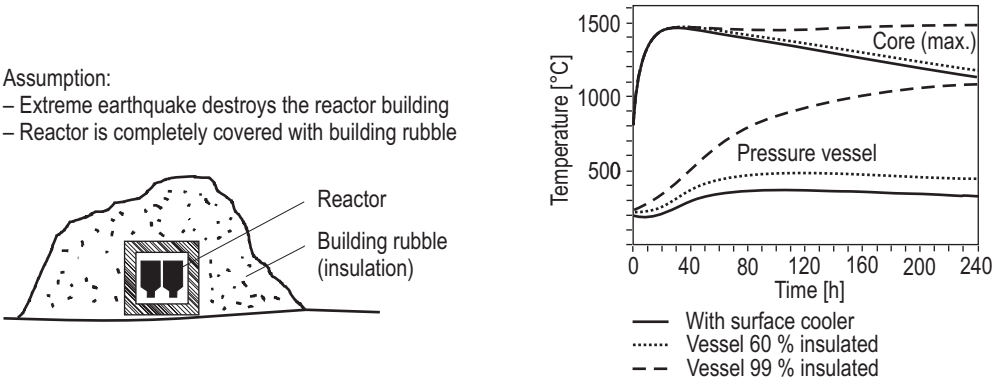


Fig. 1.175. Aspects of HTR safety: behavior of the reactor system after an extreme accident (reactor covered with rubble).

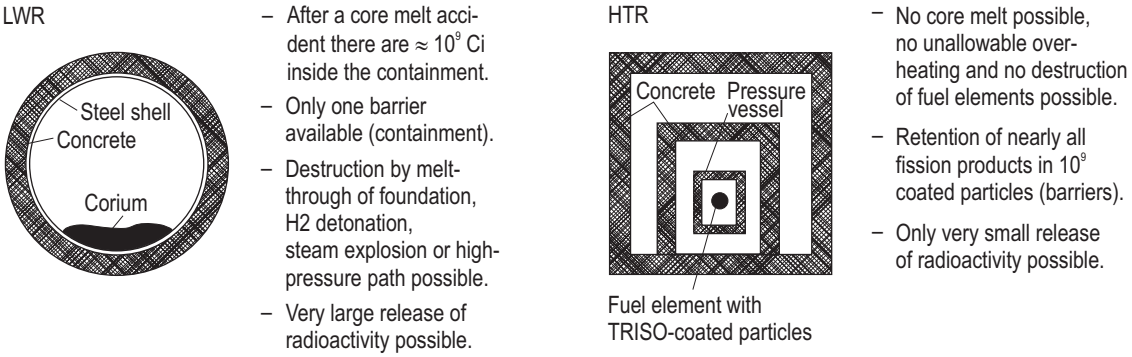


Fig. 1.176. Aspects of HTR safety: comparison of fission products retention for LWR and HTR.

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