9 NUCLEAR

Abstract

The world's demand for energy will increase because the quality of living is tied to its consumption and because the world's population is increasing. It is well known that high quality energy sources such as electricity and oil are directly linked to economic growth and quality of life (e.g., access to food, medicine, housing and education). As more countries participate in the benefits of globalization, world energy use will grow. For example, in the past decade prior to the financial collapse of world markets that occurred during October of 2008, two of the world's most populated countries, China and India, had unprecedented economic growth and this played a major role in the price of oil climbing to nearly $150 per barrel during the summer of 2008. The year 2008 also saw an precipitous drop in oil prices as well. It is well known that economic growth is cyclic and that during periods of recession, demand for energy decreases. After October of 2008 the price of oil dropped under $45 dollar per barrel. As of this writing the world was still in the recession of 2008, but history tells us that a recovery will occur and that during the next economic growth period oil prices will once again rise. The other factor which impacts growth in energy demand is growth in population. The US Census Bureau has projected that the world population will increase to 9 billion people by 2042 from approximately 6.6 billion people today. The future holds more uncertainty. The competition for scarce energy resources will only accelerate as more people in the world participate in the economic benefits of globalization and as world population grows.

One of the requirements for sustaining human life and progress is availability of a clean source of energy that does not harm the environment. The release of CO_2 into the atmosphere is the main cause of global warming and various associated climate changes. A major advantage of nuclear energy is that it doesn't put CO_2 into the atmosphere. In addition it provides a steady source of constant electrical power that does not suffer the whim of weather patterns. This is a critical feature because climate change does impact weather patterns. For example, areas

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which currently have an abundance of wind or clear skies for solar energy use today may experience a change over the 30 or so year lifetime of a wind farm or a solar energy plant.

Although nuclear energy generates electricity without releasing harmful gases; SO_2, NOx, CO_2, etc, the issues related to spent fuel (nuclear waste) from the nuclear reactors must be addressed. Reprocessing of spent fuels and the use of breeder reactors can substantially minimize its impact and even can lead to zero waste system. In this chapter, the complete nuclear fuel cycle is discussed.

9.1 Introduction

There are two fundamental ways by which energy is released from nuclear reactions: fission and fusion of atomic nuclei. The mass of a nucleus (a nucleus is made up of protons and neutrons) is always less than the sum of the individual masses of the protons and neutrons which constitute it. The difference is a measure of the nuclear binding energy which holds the nucleus together. During either a fission or a fusion reaction part of this binding energy is released and is utilized for electricity generation. This binding energy can be calculated from the Einstein relationship and is discussed below.

In 1905, a young patent clerk in Switzerland developed a theory that would forever change the world. That young patent clerk was Albert Einstein. His theory of relativity is one of the major achievements in modern physics (the other being quantum mechanics).

In the 1880s, Physicists believed that waves require a medium to propagate. Sound waves propagate in air, and waves propagate in water. It was believed that the earth traveled through an "ether wind," which allowed light to propagate through space. American Physicist Albert Abraham Michelson (1852–1931) and American Chemist Edward Williams Morley (1838–1923) designed an experiment that attempted to examine the motion of earth relative to the ether using Michelson's new instrument, an interferometer. The experimental arrangement is shown in Fig. 9.1.

A light beam is split. Part of the beam travels straight and the other part travels 90° up. Michelson and Morley assumed that the speed of light is c and the speed of the ether wind is v. Michelson and Morley reasoned that the part of the beam that travels anti-parallel and then parallel to the ether wind will take a longer time to traverse the distance $2d$, and the time is given by:

$$t_{\downarrow\uparrow} = \frac{d}{c+v} + \frac{d}{c-v} = \frac{2dc}{c^2 - v^2} \tag{9.1}$$

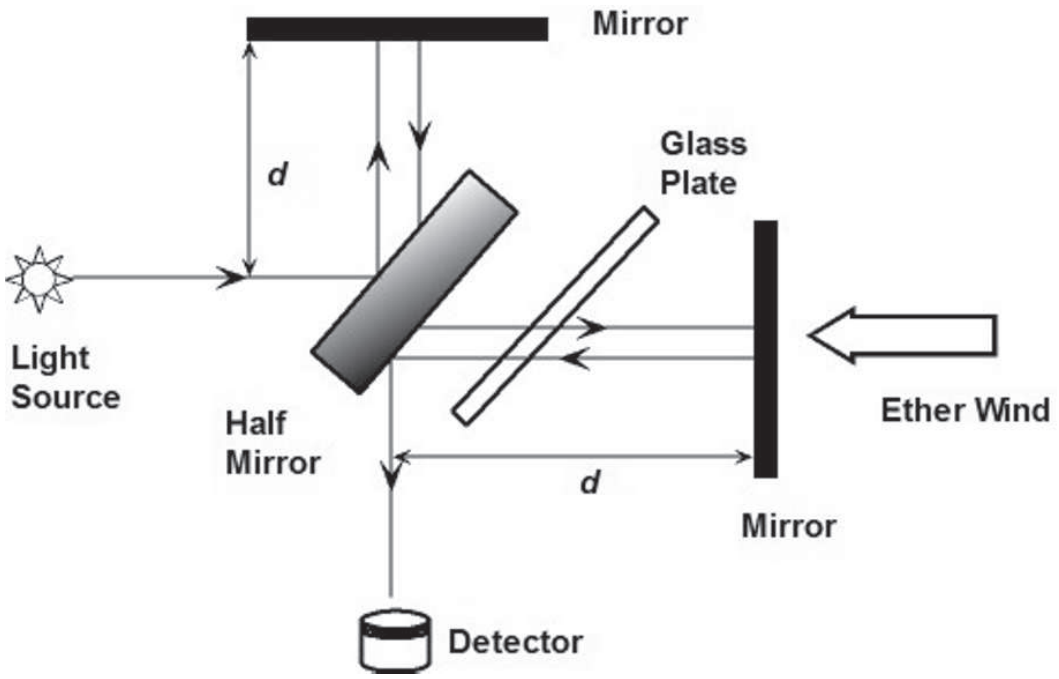


Fig. 9.1. Michelson interferometer.

Likewise along the path perpendicular to the ether wind, the time it takes the beam to traverse the distance $2d$ is given by;

$$t_{\perp} = \frac{2d}{\sqrt{c^2 - v^2}} \tag{9.2}$$

The ratio of the parallel time to perpendicular time can be represented as:

$$\frac{t_{\downarrow\uparrow}}{t_{\perp}} = \frac{1}{\sqrt{1 - \frac{v^2}{c^2}}} \tag{9.3}$$

If the light source has zero velocity with respect to the ether, then the ratio is 1. If the earth is in motion, the light source should have a velocity with respect to the ether so the ratio should not be 1. The two beams at the detector will then be out of phase and thus will interfere with one another. The interference fringes can then be measured.

Michelson and Morley ran thousands of experiments over many months. In each and every one of their experiments the ratio was 1. They had to conclude that there was no ether wind.

From this work, other physicists tried to explain the result. George Francis FitzGerald (1851–1901), an Irish physicist, developed the idea that objects grew

shorter in the direction of their motion because the ether wind would exert a pressure on the body. FitzGerald developed the FitzGerald length contraction formula from this assumption and used it to explain the Michelson–Morley experiment. The length is calculated from the following expression.

$$L = L_o\sqrt{1-\frac{v^2}{c^2}} \tag{9.4}$$

In Eq. (9.4), L is the length in the moving reference frame and L_o is the length at rest. The problem with the FitzGerald contraction is that when $v > c$, the L is an imaginary number.

Hendrick Antoon Lorentz (1853–1928), a Dutch physicist, extended the work of FitzGerald. He postulated that mass would have to increase with motion and is given below.

$$m = \frac{m_o}{\sqrt{1-\frac{v^2}{c^2}}} \tag{9.5}$$

Lorentz's work was able to predict the motion of high-speed charged particles in electromagnetic fields. Often times these contractions are called the Lorentz contractions.

Einstein assumed that the relative motion of bodies is impossible to sort out. How does one determine which body is absolutely at rest? The second assumption that he made in his new theory is that the speed of light in a vacuum is a constant regardless of frame of reference. This was a drastic change from the universally accepted Newtonian mechanics. In Newtonian mechanics if I am on a platform moving at the speed of light and I throw an object off that platform with a velocity of the speed of light, then the relative speed of that object to the rest frame is,

$$V = V_1 + V_2 = 2c \tag{9.6}$$

where $V_1 = V_2 = c$

According to Einstein, the velocity of the object would be,

$$V = \frac{V_1 + V_2}{1+\frac{V_1V_2}{c^2}} = c \tag{9.7}$$

This forced the object to have a velocity no greater than the speed of light.

Classical physicists believed that mass could not be created or destroyed. But the Lorentz contraction and Einstein's special theory of relativity differed in that mass was increased as velocity increased. The implication of this is very important to nuclear energy. We can make the following approximation (use a Taylor series expansion),

$$\frac{1}{\sqrt{1-\frac{v^2}{c^2}}} \approx 1+\frac{v^2}{2c^2} \tag{9.8}$$

So, the mass of a body in motion, m_1, becomes,

$$m_1 = m_o\left(1+\frac{v^2}{2c^2}\right) = m_o + m_o\frac{v_2}{2c^2} \tag{9.9}$$

The increase in the mass as a result of motion is given by;

$$m = m_1 - m_o = \frac{1}{2}m_o\frac{v^2}{c^2} \tag{9.10}$$

The term ½ $m_o v^2$ is the kinetic energy of the rest mass. This energy is denoted by E, therefore,

$$m = \frac{E}{c^2} \tag{9.11}$$

This equation can be rearranged to give the mass energy relationship,

$$E = mc^2 \tag{9.12}$$

The implication of this relationship can be seen in the calculation of binding energy of nuclei. If we assume that the mass of an atom is the summation of the mass of its components, then for an oxygen atom, we would assume it has eight neutrons, eight protons and eight electrons. The sum of these components would be,

$$m_{theory}^{oxygen} = 8m_p + 8m_n + 8m_e \tag{9.13}$$

where m_p, m_n and m_e are the masses of a free proton, neutron and electron. The problem is that the actual mass of an oxygen atom is less than the sum of its components.

$$m_{\exp erimental}^{oxygen} < m_{theory}^{oxygen} \tag{9.14}$$

This mass difference is,

$$\Delta m = m_{theory}^{oxygen} - m_{\exp erimental}^{oxygen} \tag{9.15}$$

The resulting mass change must go into energy according to the mass energy relationship,

$$E = \Delta mc^2 \tag{9.16}$$

This energy is known as the binding energy. In order to break apart the atom into its basic components, an energy equivalent to the binding energy of the atom must be supplied.

By convention, the number of protons in an atom is represented by Z, the number of electrons in an atom is equivalent to the number of protons and thus are represented by Z, the number of neutrons in an atom is represented by N, and the atomic number is the total number of protons and neutrons and is represented by A.

$$A = N + Z \tag{9.17}$$

It is useful to compare the binding energy of all atoms by looking at the binding energy per nucleon (a nucleon is either a proton or neutron). The binding energy per nucleon is given by the following expression.

$$\text{Binding Energy per Nucleon} = \frac{\Delta mC^2}{A} \tag{9.18}$$

We can plot a curve of the binding energy per nucleon for all known atoms as shown in Fig. 9.2.

The binding energy per nucleon curve tells us that the most stable element in the universe is iron. The second law of thermodynamics tells us that all systems tend towards their minimum energy. Therefore,

All matter in the universe will tend towards iron

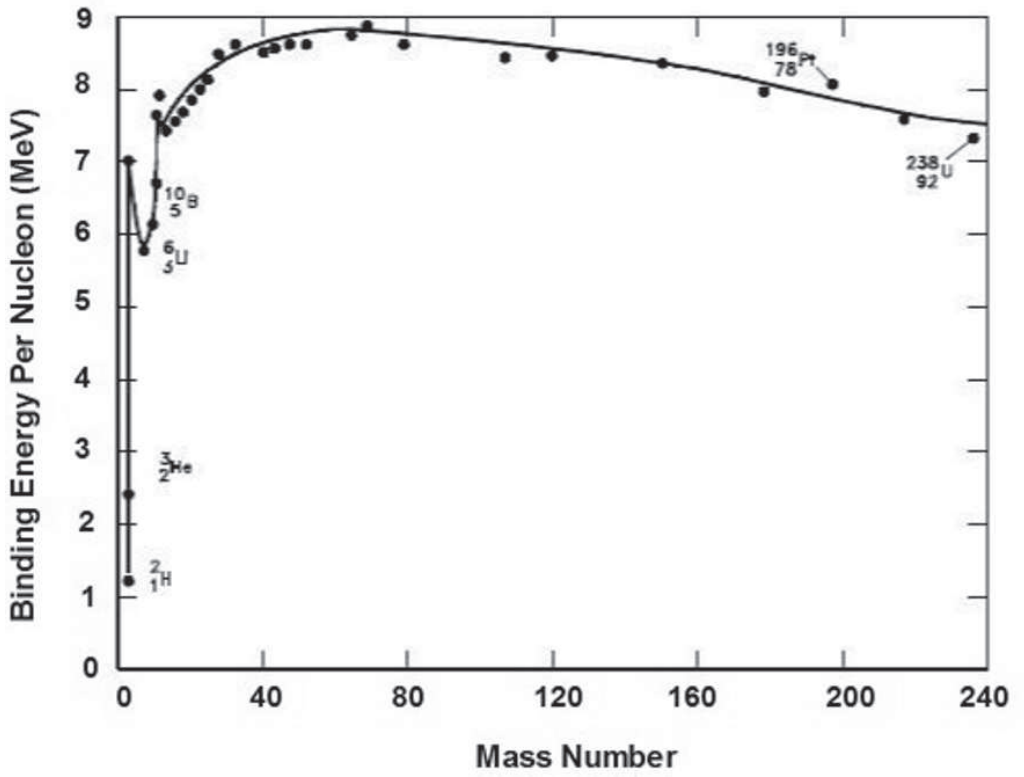


Fig. 9.2. Binding energy per nucleon as a function of mass number for the stable nuclides (Courtesy [1]).

9.2 The Fusion Cycle

According to the modern cosmology, the universe started with the big bang. A point in space with near infinite energy exploded. As the energy wave propagated, it cooled. Light particles such as baryons and leptons formed. Eventually hydrogen began to condense from this energy wave during the cooling process.

The hydrogen began to form density perturbations in space. This spatial variation of mass began to attract other hydrogen particles. The more particles that were attracted to the density variations, the more gravitation attraction there was to bring in more hydrogen. The mass began to build up. Eventually, the mass was large enough to form a proto-star. At this point, nuclear reactions that convert hydrogen into heavier elements produced sufficient energy that the energy produced by the fusion reactions balanced the energy lost by radiation. The mass grew larger and larger and eventually, the fusion energy produced was sufficient to create a high surface temperature and the surface glowed in the visible wavelength (e.g., stars were formed).

Some of the stars that were formed had enormous mass. The hydrogen fuel in these massive stars was consumed at rapid rates. Other elements formed from hydrogen fusion such as helium, and carbon began to fuse to form heavier elements. These fusion reactions created oxygen. Oxygen began to fuse creating silicon. Silicon began to fuse to form iron. The iron core remained inactive since it is the element with the largest binding energy per nucleon. As the iron core of the star grew larger and larger, the star could no longer support the pressure from the surrounding mass. This caused the star to collapse. As the star collapsed upon itself, the outer shells of silicon, oxygen, carbon, helium and hydrogen grew denser. This density increase caused an upsurge of fusion activity that resulted in an explosion. The massive star became a supernova. The massive energy release sent iron, silicon, oxygen, carbon, and helium and hydrogen particles in all directions at very high energy. Some of these particles collided (the supernova was like a massive celestial high-energy accelerator) and underwent complex nuclear reactions. These reactions produced elements heavier than iron. The remnants of the supernova began to condense. New stars were formed along with planets from the debris.

9.3 The Fission Cycle

Among the heavy elements that condensed to form planets was uranium-235. In the late 1930s Enrico Fermi had been bombarding uranium with thermal neutrons (the definition of thermal neutron is discussed later). He believed that he had created an element with an atomic mass of 93. Basically he was right, but his explanation was not very clear.

In 1938, German Physicist Otto Hahn and his coworkers Fritz Straussman performed an experiment planned by Hahn and Lise Meitner in which uranium was bombarded with thermal neutrons. The chemical makeup of the uranium had changed in that presence of barium was found. Because Meitner was a Jewish exile from Germany, Hahn and Straussman, who worked in the Kaiser-Wilhelm-Institut in Germany, could not include Meitner as a co-author of the paper announcing the result. Lise Meitner was widely credited as being the first to recognize that nuclear fission had occurred. She and Otto Frisch developed a proof that fission had occurred in these experiments and published a paper in Nature, January 16, 1939. In their paper they described how barium could be produced in a fission reaction along with krypton and additional neutrons.

Word of the discover spread like wildfire. Niels Bohr went to the US for a conference and discussed the possibility of fission with US physicists. Many of the physicists went back to their laboratories and verified the fission reaction.

Hungarian-American physicist Leo Szilard began to think about the nuclear chain reaction shortly afterwards. He pondered on the implications of the reactions and became concerned. Given that he was a Jewish refugee from Hitler's tyranny, he was fully aware of the danger that the Nazis posed to the world and what it would mean if Hitler was able to develop a weapon based on the nuclear chain reaction. Sizlard discussed his fear with physicist Eugene Paul Wigner and Edward Teller. They decided to bring their concerns to the attention of the US government. In order to do so they needed the help of the world's most prominent scientist, Albert Einstein. They were able to persuade Einstein to write a letter to President Franklin D. Roosevelt. In 1941, Roosevelt agreed to start a massive research program to develop a bomb based on nuclear fission. The order was issued on December 6, 1941, the day before the Japanese attacked Pearl Harbor.

The start of mankind's trek into nuclear energy began with fear while the world was in the death grip of World War II.

Nuclei of atomic mass up to 40 remain stable as the number of protons and the number of neutrons remain equal (Fig. 9.3). The strong force which holds the nucleus together is stronger than the Coulombic force which causes the protons to repel one another. The strong force has a maximum interaction length of about 1–1.5 fm, beyond which it is zero. Additionally, the strong force is repulsive as distances become shorter than 0.1 fm.

A nucleus is on the order of 1 fm in diameter. A proton and a neutron are about 0.01 fm in diameter. The density of nucleons in a nucleus is about constant throughout the volume. The importance of this fairly constant density is that the distance between nucleons remains constant regardless of the number of nucleons in the nucleus.

As the number of protons increases, the Columbic force gets larger and larger. The distances between protons must get larger. Neutrons fill in the space as protons move further apart. Eventually, the number of nucleons fills a volume so large that the strong forces can no longer act. At this point, the number of neutrons becomes larger than the number of protons.

The line of stability shown in Figs. 9.4 and 9.5 is very narrow. If there are fewer neutrons in a nucleus than the line of stability, the nucleus will undergo a radioactive process in which a proton in the nucleus is converted to a neutron and a positron. A positron is a positively charged electron.

$$ {}^{A}_{N}X_{Z} \rightarrow {}_{N+1}{}^{A}Y_{Z-1} + \beta^{+} + \nu \tag{9.19} $$

where β^{+} is a positron and ν is a neutrino.

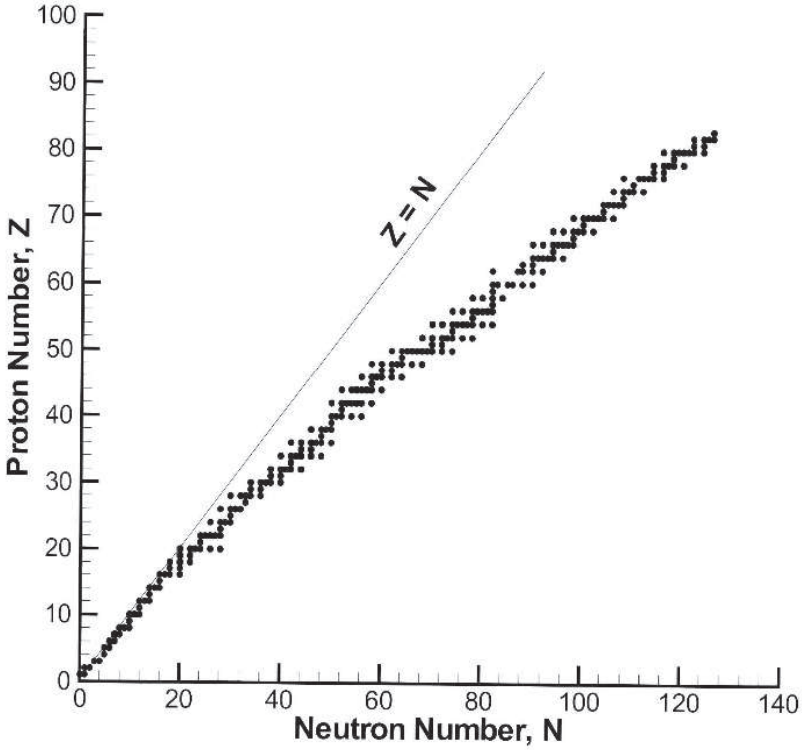


Fig. 9.3. Values of protons and neutrons for 266 stable nuclides. The line Z = N represents number of proton equals to number of neutron.

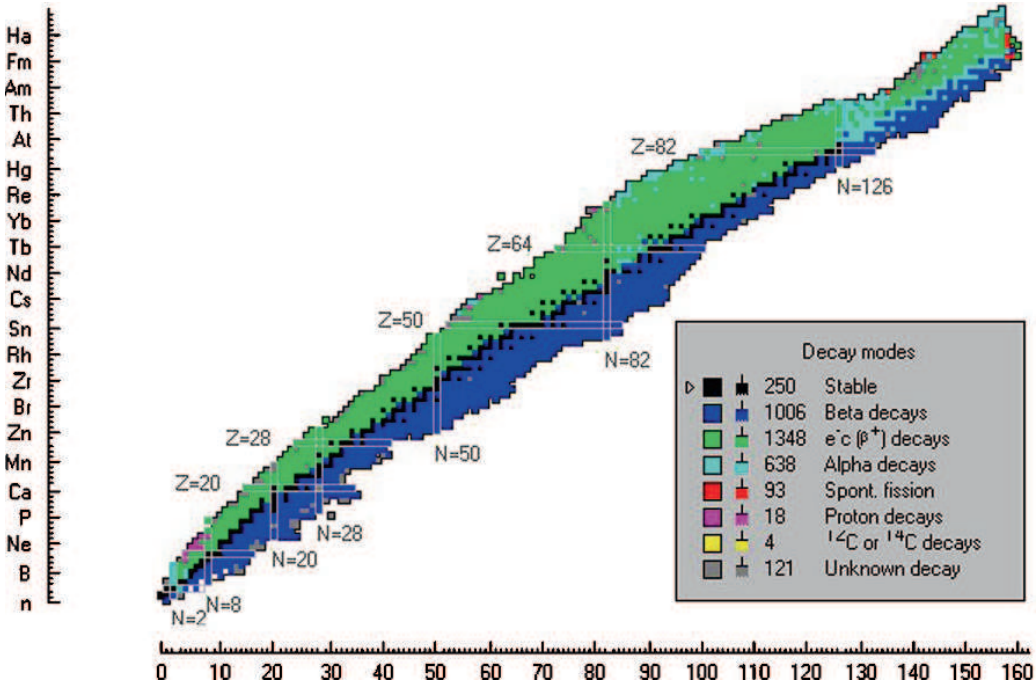


Fig. 9.4. A plot of all 250 stable elements vs. the atomic mass (Courtesy of [2]).

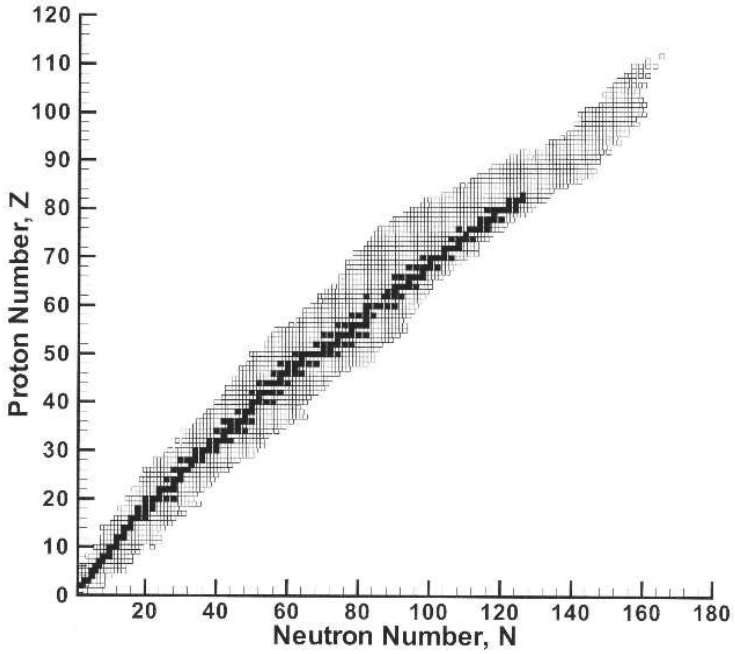


Fig. 9.5. Plot of neutron number versus proton number for all known nuclides (Dark squares represent stable nuclides and light squares represent unstable nuclides (Courtesy of [2]).

In this reaction, the element X is neutron poor and lies above the line of stability. If there are fewer neutrons than the line of stability, the decay mode is β^+. If there are more neutrons than the line of stability, the decay mode is β^-.

$$ {}^{A}_{N}XX_{Z} \rightarrow {}^{A}_{N-1}YY_{Z+1} + \beta^- + \bar{\nu} \tag{9.20} $$

where $\bar{\nu}$ is an antineutrino.

The farther a nucleus is from the line of stability, the faster it decays. In nuclear science, the rate of decay is illustrated by the half life. The half life is the time it takes N radioactive atoms to decay to $0.5N$ radioactive atoms. The decay of radioactive elements is a statistical process. This process can be modeled by a rate equation:

$$ \frac{dN}{dt} = -kN \tag{9.21} $$

where k is a decay constant.

The solution to this equation is,

$$ N(t) = N(0)e^{-kt} \tag{9.22} $$

The half life can be found by solving,

$$\frac{N\left(t_{1/2}\right)}{N(0)} = 0.5 = e^{-k\,t_{1/2}} \tag{9.23}$$

$$-k\,t_{1/2} = \ln(0.5) \tag{9.24}$$

$$k = \frac{0.693}{t_{1/2}} \tag{9.25}$$

Typically, in nuclear science, the decay rates of nuclei are given in half-lives. From the half-life, the decay constant is calculated using Eq. (9.25).

There are other types of radioactive decay processes. Some heavy nuclei can decay by giving off alpha particles (helium nucleus) and other types of ions.

One of the most interesting processes is fission. Elements like plutonium 238 can spontaneously fission. As discussed, fission is the splitting of a nucleus. On average, each fission gives off more than two neutrons (with a high energy distribution as shown in Fig. 9.6).

$${}^{A}_{N}X_{Z} \rightarrow {}^{A1}_{N1}fl_{Z1} + {}^{A2}_{N2}fh_{Z2} + \vartheta n \tag{9.26}$$

where ϑ is the statistical number of neutrons given off per fission. For U-235, ϑ is 2.44.

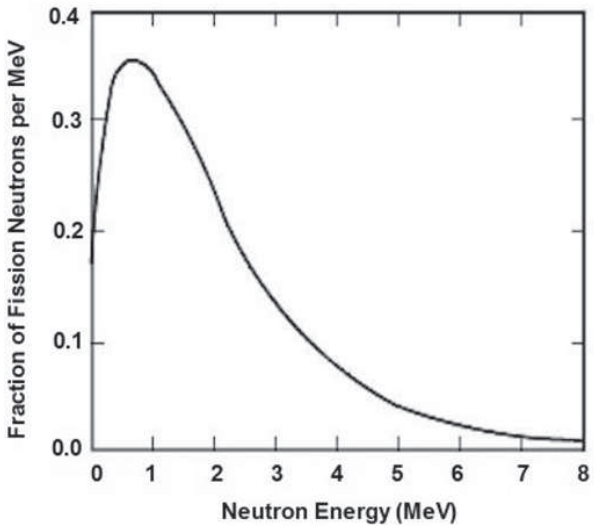


Fig. 9.6. Energy distribution of neutrons given off by the fission of U-235 as calculated (Courtesy [1]).

The release of more than one neutron per fission was key to Leo Szilard's idea of a chain reaction. If you consider neutron bookkeeping, you use one thermal neutron to initiate fission. In the process the fission gives off 2.44 additional neutrons. If each of these neutrons causes fission, then you have 2.44 × 2.44 (5.95) neutrons in the second generation. As you can see, each additional generation would have 2.44 more neutrons than the previous generation. This process is a geometrical process. However, such processes can not be sustained. There are natural processes which will cause the fission rate to plateau.

Nuclear fission occurs in several elements. The reaction begins with the capture of a neutron in the nucleus. In general, the probability of capturing a neutron with a low velocity is higher than capturing a neutron with a high velocity. The energy of a neutron is given by,

$$E_n = \frac{1}{2} m_n v^2 \tag{9.27}$$

where, E_n is the neutron energy (Joules), m_n (kilogram) is the neutron mass and v (meters per second) is the neutron velocity.

Energy is related to temperature. For an individual neutron the temperature/energy relationship is,

$$E_n = kT \tag{9.28}$$

where k is Boltzman;s constant (1.381×10^{-23} J/K) and T is temperature in Kelvin.

Because the energy of a high velocity neutron can be a very small number in Joules, it is common to express particle energy in electron Volts (eV).

$$1 \text{ eV} = 1.6 \times 10^{-19} \text{ J} \tag{9.29}$$

If we want to know the energy of a neutron, which is at room temperature, it can be calculated from Eqs. (9.28) and (9.29). Room temperature is about 20°C or 293 K. So a neutron of temperature 293 K has an energy of:

$$E_{293} = \frac{kT}{1.6 \times 10^{-19}} = \frac{\left(1.38 \times 10^{-23} \times 293\right)}{1.6 \times 10^{-19}} = 0.0253 \text{ eV} \tag{9.30}$$

Figure 9.7 shows the cross section for nuclear fission in U-235 and U-238 as a function of energy. The cross section is a measure of the probability of a reaction. Thus the higher the cross section, the higher is the probability of the reaction. As Fig. 9.7 shows, the cross section for fission in U-235 is very high at low energies. The unit of a barn is used which is 1×10^{-24} cm^2. This is about equal to the

cross sectional area of a typical nucleus. The absorption cross sections for all of the elements fall of as $1/v$ (where v is velocity) at energies below 2 eV. For light elements the $1/v$ behavior persists to high energies. For heavy elements there will be sharp peaks at energies above 2 eV which are due to enhanced absorption in the nucleus due to metastable levels. These sharp peaks are called resonances and they occur roughly between 2 eV and 100 KeV. Thermal cross sections are very important to modern light water or heavy water moderated power plants. Table 9.1 tabulates cross sections for important elements at 0.025 eV energies.

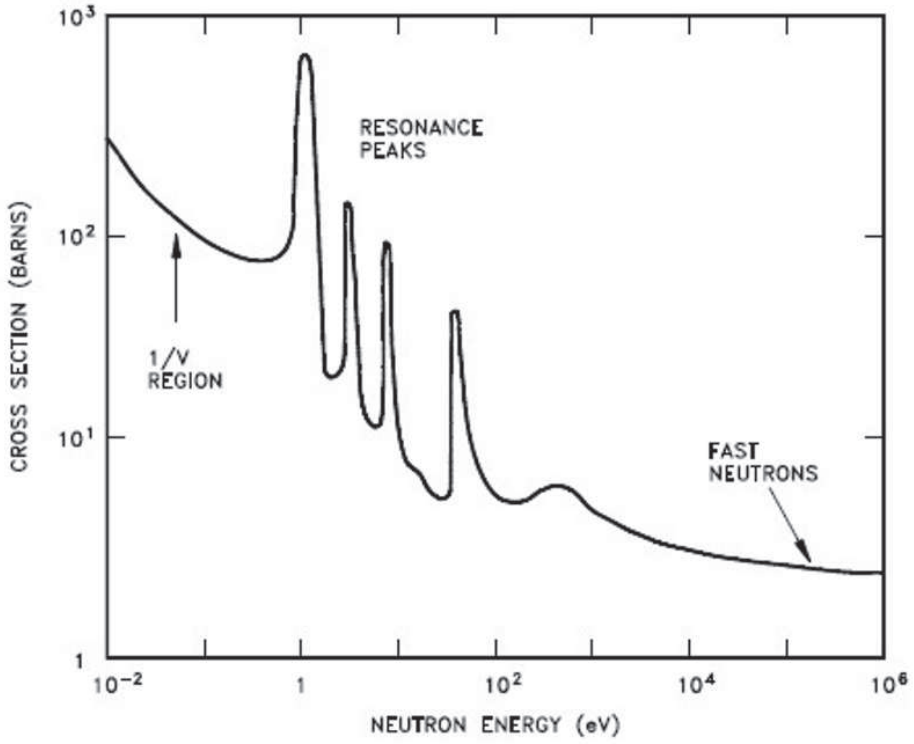


Fig. 9.7. Cross section for neutron capture (n, γ) and fission for U-235 and U-238 as a function of neutron energy (Adapted from [1]).

Table 9.1. Data for interactions with various elements at 0.025 Ev.

Element	Absorption cross section (barns)	Fission cross section (barns)
U-233	579	531
U-235	681	582
Natural U	7.59	4.19
Pu-239	1011	743
Pu-241	1377	1009

The concept of a cross section can be understood by looking at the illustration in Fig. 9.8 shown below,

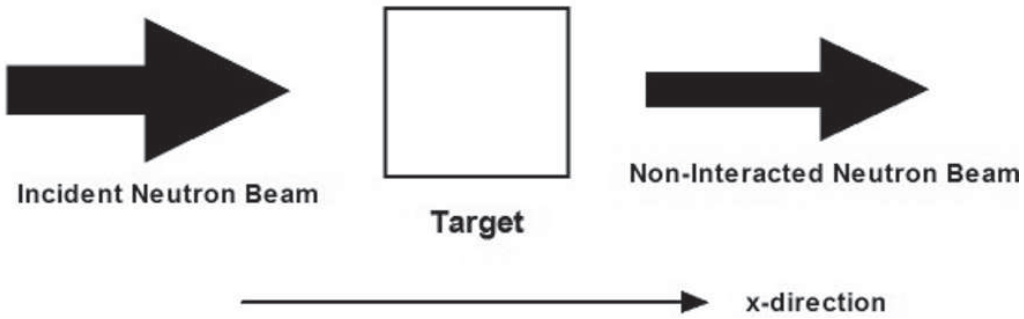


Fig. 9.8. Illustration showing concept of a nucleus cross section. An incident beam of neutrons intersect a target material and the non interacted beam of neutrons exits the target.

In this illustration, an incident beam of neutrons with intensity *I(x)* (neutrons per square meter per second) strikes the target and the non-interacted neutrons exit the target. The neutron beam intensity is a function of *x* (meter), *I(x)* and an equation can be set up to calculate it. We use the cross section to make this calculation by setting up the following relationship,

$$-dI(x) = N\sigma I(x)dx \tag{9.31}$$

where *dI(x)* is the differential neutron beam intensity at the point *x*, *N* is the density of target atoms (atoms per m^3), σ is the cross section (m^2), *dx* is a differential length.

The solution to this differential equation is,

$$I(x) = e^{(-N\sigma x)} \tag{9.32}$$

Terms can be grouped. The macroscopic cross section Σ is equal to $N\sigma$. Using the macroscopic cross section we can write the equation,

$$-\frac{dI(x)}{I(x)} = \sum dx \tag{9.33}$$

where this represents the fraction of neutrons at the point *x* which have interacted in a length of *dx*.

Thus, the term Σdx represents the probability that a neutron will interact in a length of *dx*. The macroscopic cross section is the probability per unit path length that a neutron will interact. Thus,

$$\frac{I(x)}{I(0)} = e^{(-\Sigma x)} \tag{9.34}$$

$e^{(-\Sigma x)}$ is the probability that a neutron reaches the point x without interacting with the target.

Define, $p(x)\,dx$ as the probability that a neutron which reaches x has its first interaction in a length dx. The mean free path (λ) is the average distance that a neutron travels before interaction. The mean free path is given by,

$$\lambda = \int_0^\infty x\,p(x)dx = \int_0^\infty x\Sigma e^{\Sigma x}dx = \frac{1}{\Sigma} \tag{9.35}$$

When a neutron is captured by a nucleus, a compound nucleus is formed. The additional neutron causes instability in the nucleus thus causing it to deform (Fig. 9.9). As the nucleus deforms it begins to split into two pieces. These pieces then emit gamma rays and neutrons leaving behind two fission fragments (Fig. 9.10). The fission fragments typically are unstable and eventually emit radiation over a time scale of seconds to years.

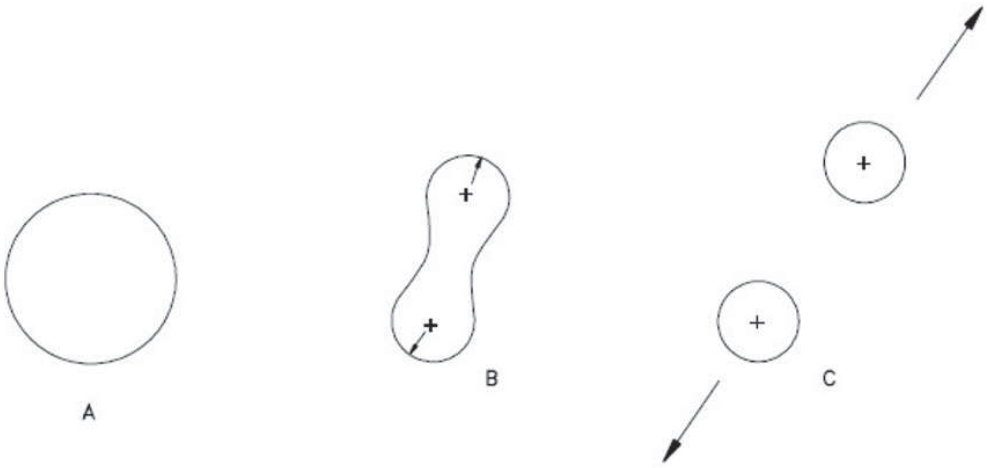


Fig. 9.9. Curve showing the potential energy versus distance of the two components of the compound nucleus as it deforms and splits into two segments. E_A is the activation energy required to split the nucleus into two fragments and the equation is the governing Coulombic repulsion equation (Adapted from [1]).

For fission to occur, it is possible to formulate a model which shows why certain elements fission. Begin by defining Potential energy (V), Kinetic Energy (KE) and total energy (M_AC^2). Total energy is equal to potential energy plus kinetic energy,

$$M_A\,C^2 = V + KE \tag{9.36}$$

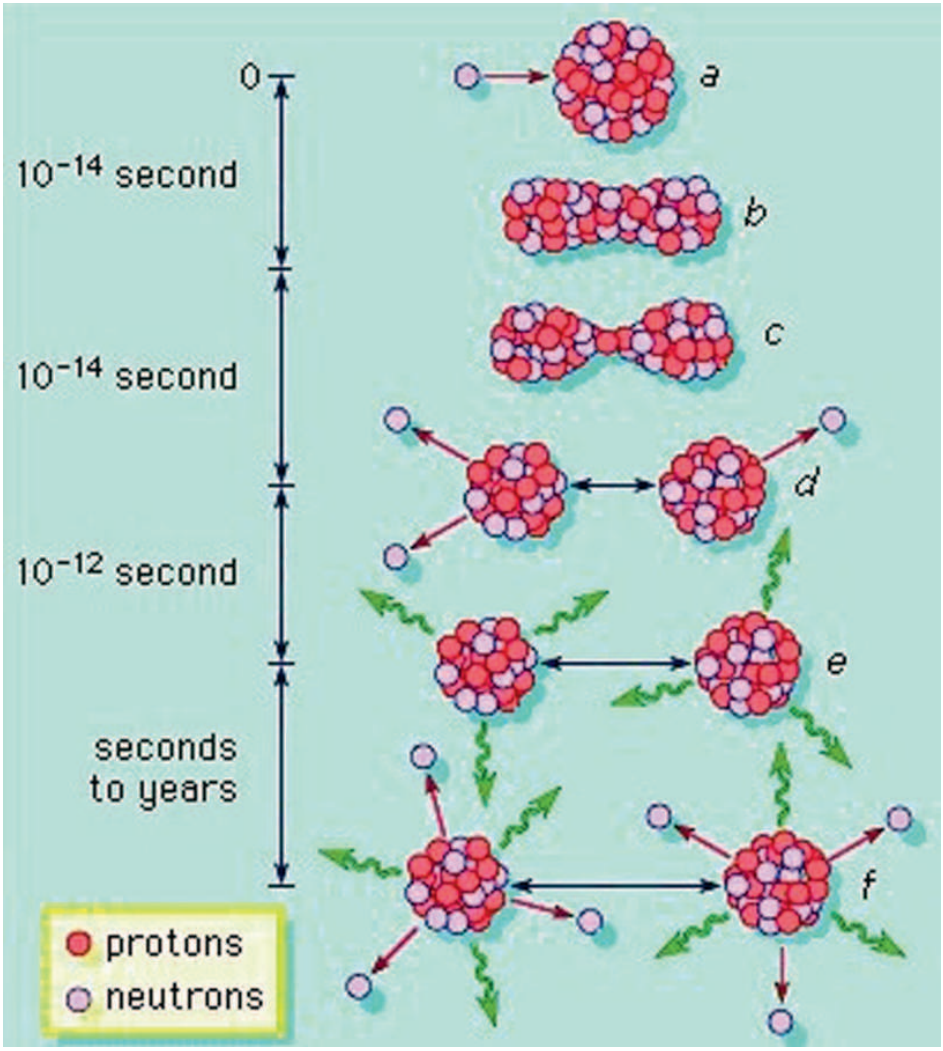


Fig. 9.10. An illustration of the phases of neutron capture leading to fission. In step *a* the neutron is captured by the nucleus. In steps *b* and *c* the nucleus begins to split. In step *d* the fission fragments form and fast neutrons emitted. The fragments promptly emit gamma rays in step *e* as they energetically stabilize. In step *f* the radioactive fission fragments decay (Adapted from [3]).

Therefore,

$$V = M_A C^2 - KE \tag{9.37}$$

Initially, before the two fragments split, the distance between the two fragments is *0* and the *KE* is *0*. As the nucleus begins to deform, energy must be added to the nucleus (E_{crit}) to overcome the strong force. So,

$$V = M_A C^2 + E_{crit} \tag{9.38}$$

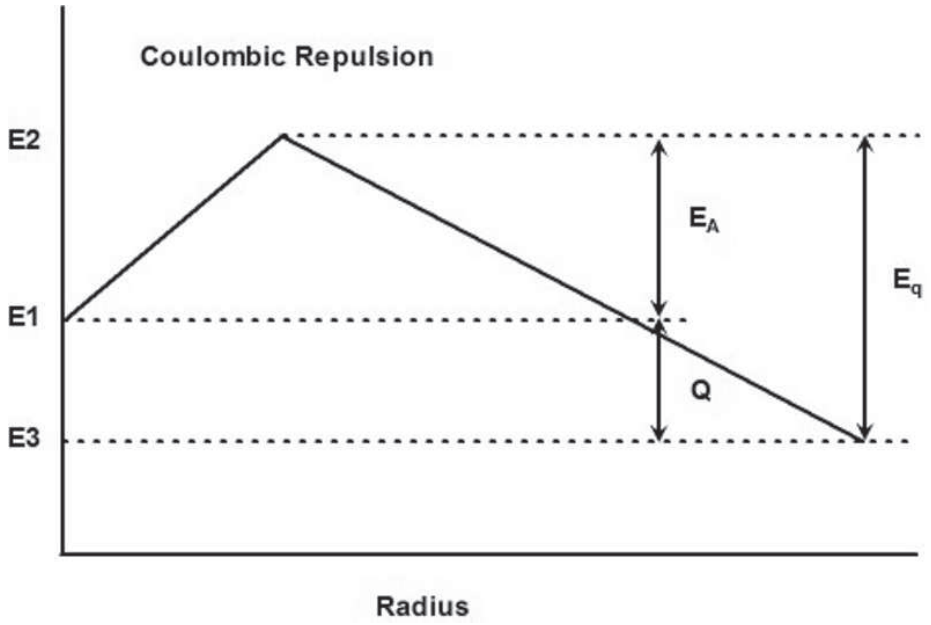


Fig. 9.11. This figure illustrates how the nucleus fissions. *E1* is the total energy of the nucleus before splitting, *E2* is the total energy plus the energy required to overcome the strong force, *E3* is the total energy of the two fission fragments, *Q* is the change in energy from the initial to final state and E_q is the coulomb repulsion energy.

From Fig. 9.11, the *Q* value is,

$$Q = M_A C^2 - M_{A1} C^2 - M_{A2} C^2 \tag{9.39}$$

where M_{A1} is the mass of the first fission fragment and M_{A2} is the mass of the second fission fragment.

E_{crit} can be estimated as,

$$E_{crit} = E_q - Q \tag{9.40}$$

and E_q by,

$$E_q = \frac{\left(Z_1 Z_2 e^2\right)}{\left(R_1 - R_2\right)} \tag{9.41}$$

where Z_1 is the number of protons in fission fragment 1 and Z_2 is the number of protons in fission fragment 2, *e* is the charge of a proton, R_1 is the radius of the nucleus before splitting and R_2 is the radius of the nucleus when it begins to split.

The radius of a nucleus is estimated by,

$$R = r_e \frac{A^{1/3}}{2} \tag{9.42}$$

where A is the atomic mass and $r_e = \dfrac{e^2}{m_e C^2}$ and m_e is the mass of an electron.

From Equations (9.41) and (9.42) it can be seen that,

$$E_q = \frac{Z_1 Z_2 e^2}{\left(\frac{r_e}{2}\right)\left(A_1^{1/3} + A_2^{1/3}\right)} \tag{9.43}$$

and,

$$E_q = \frac{2m_e C^2 Z_1 Z_2}{\left(A_1^{1/3} + A_2^{1/3}\right)} \tag{9.44}$$

The term $2m_eC^2$ is about 1 MeV. Also it can be assumed for the time being that $Z_1 = Z_2$, therefore,

$$E_q \approx \frac{0.16Z^2}{A^{1/3}} \tag{9.45}$$

For U-238, $Z = 92$ and $A = 238$, and

$E_q = 218$ MeV

Q for U-238 is about 212 MeV, thus

$E_{crit} \sim 6$ MeV

For lead, Pb-208, $Z = 82$ and $A = 208$. The value of E_q is given by,

$E_q = 182$ MeV

For Pb-208, Q is about 162 MeV, so,

$E_{crit} \sim 20$ MeV

In Table 9.2 is given the E_{crit} and binding energy of the last nucleon for various elements.

Table 9.2. The value of E_{crit} and the binding energy for the last nucleon.

Element	E_{crit} (MeV)	Binding energy last nucleon (MeV)
Th-232	5.9	*
Th-233	6.5	5.1
U-233	5.5	*
U-234	4.6	6.6
U-235	5.75	*
U-236	5.3	6.4
U-238	5.85	*
U-239	5.5	4.9
Pu-239	5.5	*
Pu-240	4.0	6.4

*Binding energy is not relevant for this analysis.

Fission will occur when the binding energy of the last nucleon is greater than E_{crit}. From Table 9.2, this condition applies to U-234, U-236 and Pu-240. Thus the nuclei which capture the neutron to trigger fission are U-233, U-235 and Pu-239. These materials are the primary practical fissile fuels.

The fission fragments do come out as a light fragment and a heavy fragment. However the fragments do not come out as a particular pair of nuclei. They come out in a distribution of paired nuclei. Figure 9.12 shows the distribution of paired nuclei as two distinct peaks.

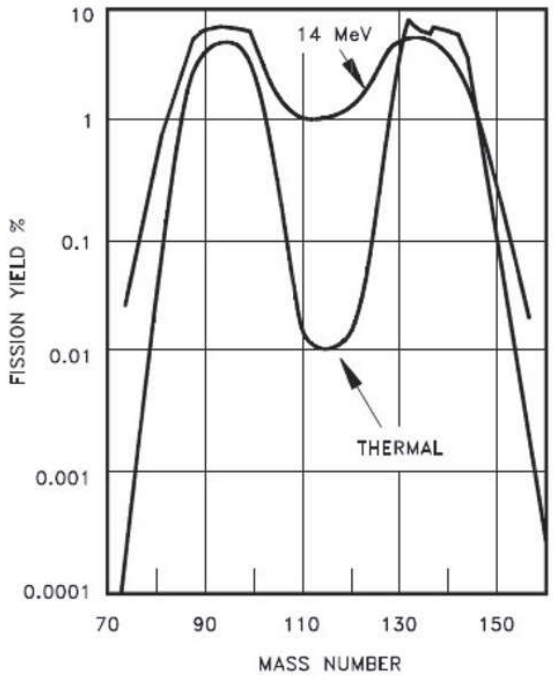


Fig. 9.12. The percent yield of fission fragments manifest as two peaks, one for the heavy fragment and one for the light fragment (Adapted from [3]).

The decay of radioactive fission products produces heat. Albeit, it is a small amount of the total heat produced in an operating nuclear power plant (about 0.7% of the total heat), it is enough heat to require cooling when the reactor is shut down. The Borst-Wheeler formula calculates the decay heat of a reactor after it has shut down,

$$P(t,T) = 4.10 \times 10^{11} \left[t^{-0.2} - (t+T)^{-0.2} \right] \tag{9.46}$$

where $P(t,T)$ is the total power emitted in the form of beta and gamma rays from the decaying fission products in a reactor which has operated T seconds at a power of 1 W, t is the time after shutdown in seconds.

For example, the University of Missouri Research Reactor is the most powerful University operated research reactor in the United States. Its operating power is 10 MW. During a typical week it operates continuously at full power for 150 h. The decay heat 10 s after shutdown can be calculated accurately by the Borst-Wheeler formula,

$T = 150$ h $= 5.4 \times 10^6$ s, thus
$P(10\text{s}, 5.4 \times 10^6\text{ s}) = 2.29 \times 10^{11}$ MeV/s/ W
Decay Heat in Reactor $= 2.29 \times 10^{11}$ MeV/s/W $\times$ 10,000 W
Decay Heat in Reactor $= 2.29 \times 10^{17}$ MeV/s $\times$ 1.6×10^{-13} J/MeV = 36,640 W

Once the nucleus fissions, one of the products is neutrons. Depending on which pair of fission fragments come out, the number of neutrons given off can vary. Over a large number of fissions, the statistical average of neutrons given off per event for example is about 2.44 for U-235. The decay chain of U-235 is shown in Fig. 9.13. The neutrons are born at fairly high energies (greater than 1 MeV) and as shown in Fig. 9.12, the energies have a distribution which is dependent on the fission fragment pair.

Nuclear reactors are designed to take advantage of the large cross section at thermal energies. In order to slow down the fast neutrons born in fission, materials have to be used to slow down the fast neutrons. These materials are called moderators and are based on light elements such as graphite, water, beryllium and heavy water.

Reactors are typically designed to use thermal neutrons (with neutron energies below 1 eV) and fast neutrons with energies from about 100 keV up to the top of range of the fission neutron energy spectrum. Thermal cross sections are very important to modern light water or heavy water moderated power plants. Table 9.1 tabulates cross sections for important elements at 0.025 eV energies.

In reality, the reactor materials are not entirely made up of fissile fuels and moderators. There are components such as structural elements, fuel cladding and there is leakage of neutrons from the reactor.

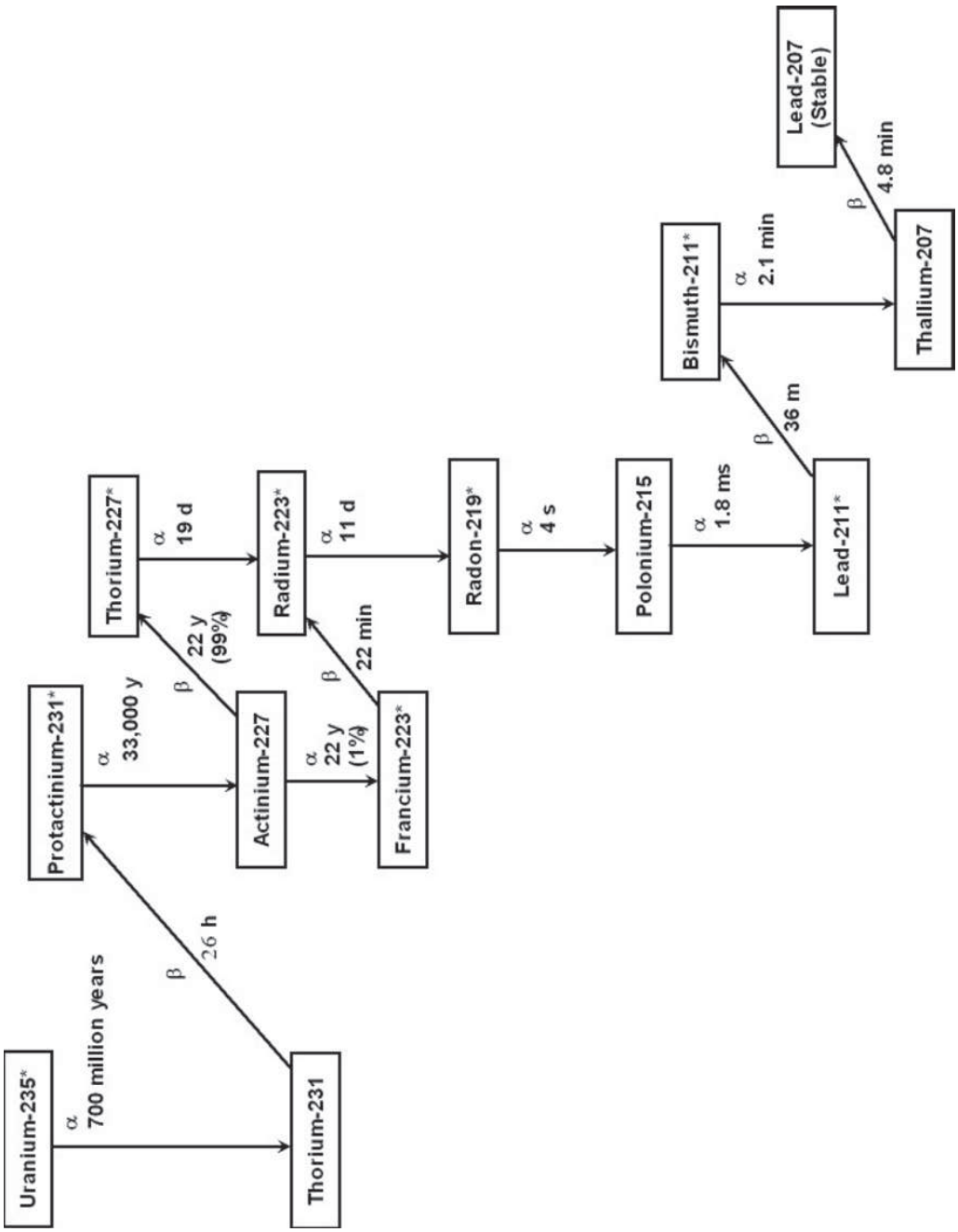


Fig. 9.13. Uranium-235 decay chain.

In reactor physics we describe the above process by the six factor formula.

$$k = \vartheta f p \varepsilon P_f P_t \tag{9.47}$$

In Eq. (9.47), ϑ is the number of fast neutrons produced by fission with thermal neutrons; f is the fraction of thermal neutrons absorbed in the fuel. Some of the thermal neutrons are absorbed in the fuel and some in the other materials that make up the reactor; p is the resonance escape probability. As the fast neutrons move through materials, there are absorption resonances at high neutron energies (2 to 100,000 eV in the materials that make up the reactor. This is the probability that the fast neutrons will slow down without being absorbed; ε is the fast fission factor. Fast neutrons can produce fission, ε is the number of additional fast neutrons that are created by fast fission; P_f is the probability that a fast neutron stays in the reactor. Some fraction of fast neutrons will leak out of the reactor volume; P_t is the probability that a thermal neutron stays in the reactor. Some fraction of thermal neutrons will leak out of the reactor volume. This process is shown in Fig. 9.14.

The factor k is a means of neutron bookkeeping. k neutrons produced in fission will survive to be absorbed by the fuel and will produce k additional fission reactions. If k is greater to or equal to one, then the reaction is self sustaining. This is a chain reaction. Reactors are designed to have $k > 1$. A nuclear reactor is controlled by the use of control rods. These rods are made up of materials that absorb thermal neutrons. When the control rods are inserted into the nuclear reactor, the k value of the reactor is less than one. When the control rods are taken out of the reactor, $k > 1$ and the reactor is able to undergo a chain reaction. The control rods allow the operator to control the reactor power level and to shut the reactor down.

The k value is a geometrical series which increases rapidly,

$$\text{Neutrons in Generation } n = k^n \tag{9.48}$$

Figures 9.15 and 9.16 is the neutron production over multiple generations for a k value of 2.44 and 1.1, respectively. Given that the time between generations is on the order of microseconds, and that energy release is directly proportional to the number of neutrons, it can be seen that energy output can multiply rather rapidly over a short period of time.

The total number of fission reactions can be very large in a short amount of time. Even though nuclear science had many applications which had saved millions of lives by the time Word War II started, people associate the word nuclear with nuclear weapons. A nuclear weapon is a formidable device which is designed to have a relatively large value of k for the purpose of releasing as much energy as possible in a short period of time. The weapon is designed to maintain a high k value until the energy released blows the device apart. We can see the potential of the weapon in the following example.

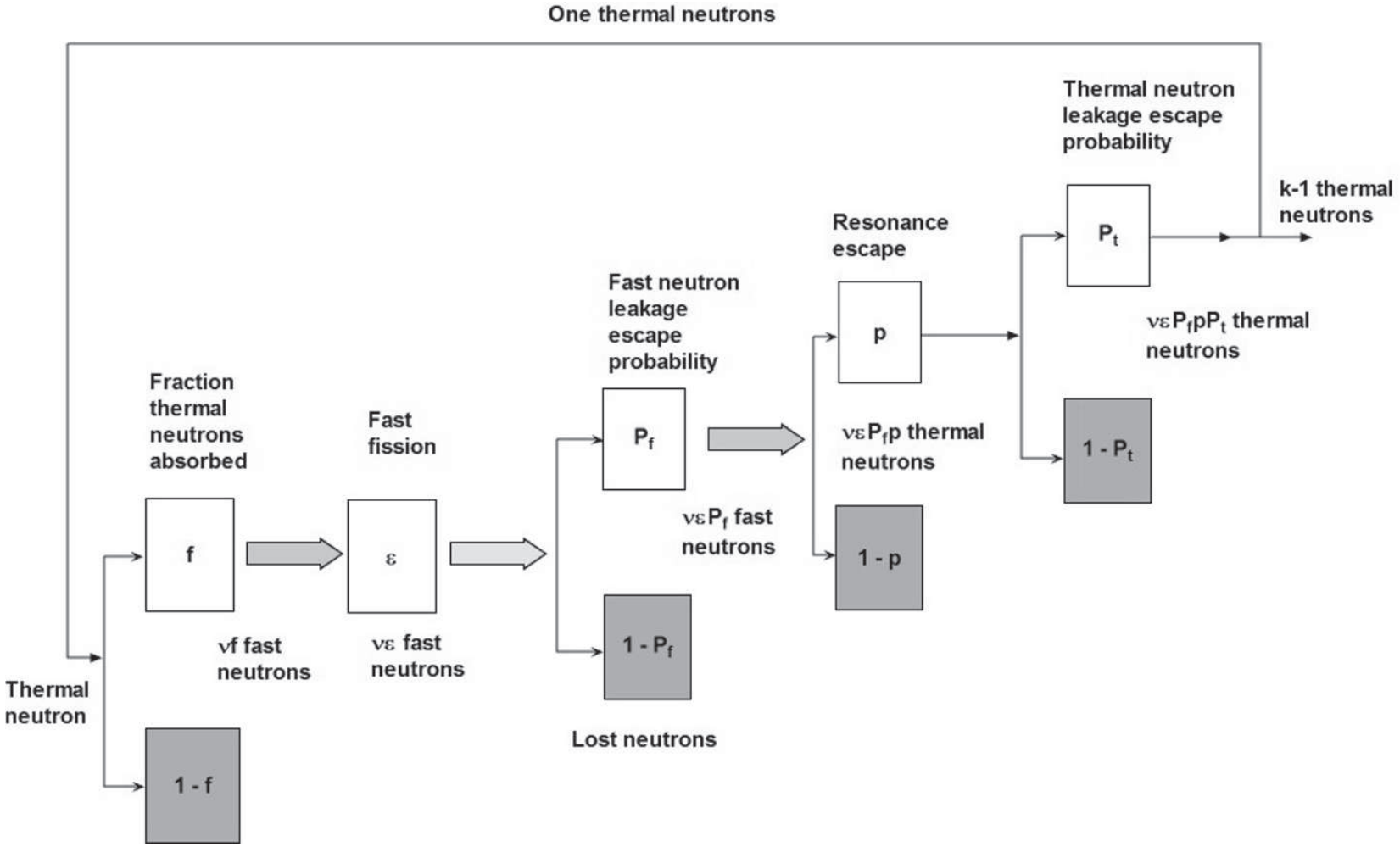


Fig. 9.14. Neutron bookkeeping in a reactor. Printed with permission from Prelas MA, Peck MS (2005) Nonproliferation Issues for Weapons of Mass Destruction. CRC Press [4]

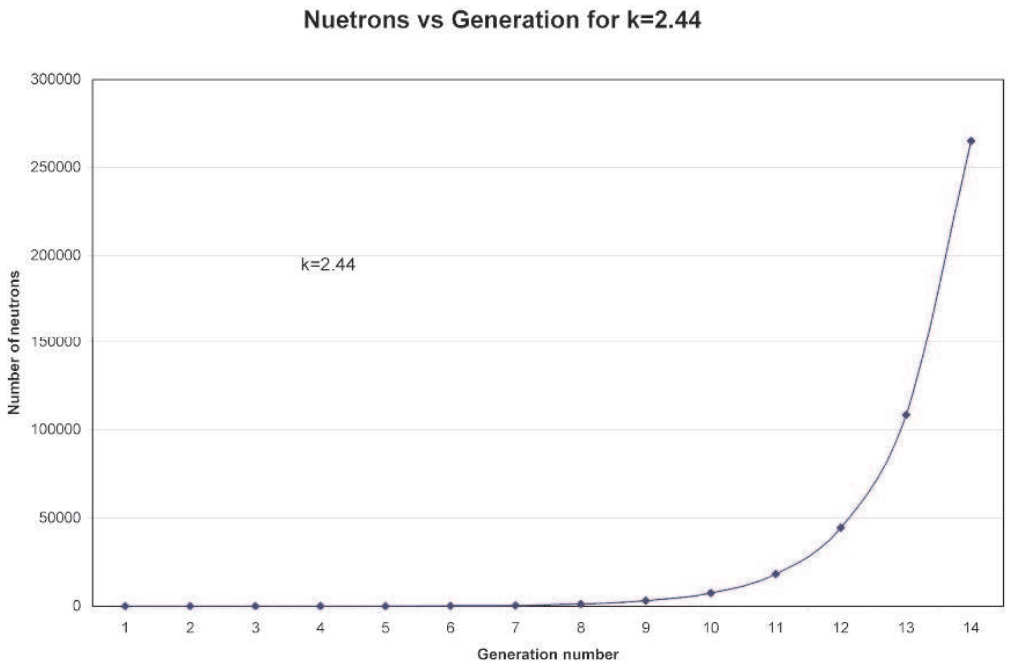


Fig. 9.15. Neutron production from generation to generation for a k value of 2.44 (Printed with permission from [4]).

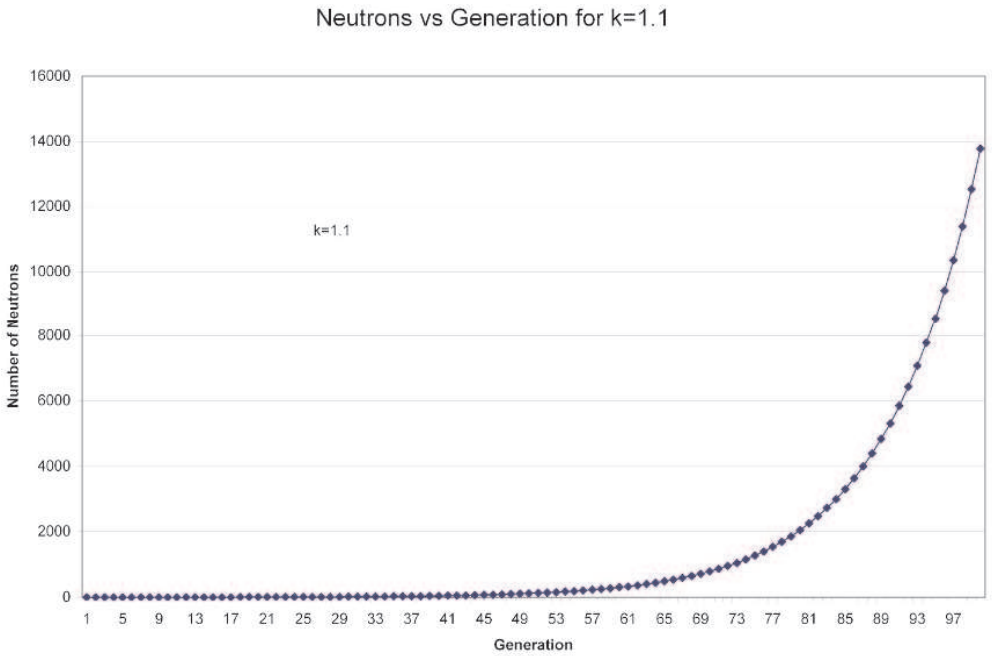


Fig. 9.16. Neutron production from generation to generation for a k value of 1.1 (Printed with permission from [4]).

Consider a nuclear warhead which has a k = 1.5 and holds together for 139 neutron generations, what is the total energy released? We begin by looking at the total number of neutrons (TN) that occur in 139 generations,

TN = neutrons in generation 1 + neutrons in gen 2 + … + neutrons in gen i

generation 0 = n_0
generation 1 = $n_0 k$
generation 2 = $n_0 k^2$
generation i = $n_0 k^i$

This is a power series which has an analytical solution,

$$TN(i) = n_0 \left[\frac{k^i - 1}{k - 1} \right]$$

Since each fission will yield ν neutrons in a generation the total number of fissions will be,

TF = fissions in generation 1 + fissions in gen 2 + … + fissions in gen i

$$\text{Fissions in generation 1} = \frac{n_0 k}{\nu}$$

$$\text{Fissions in generation 2} = \frac{n_0 k^2}{\nu}$$

$$\text{Fissions in generation } i = \frac{n_0 k^i}{\nu}$$

The formula for the total number of fissions will be,

$$TF(i) = n_0 \frac{\frac{k^i - k}{k - 1}}{\nu}$$

Thus, assuming that n_0 = 1, and ν = 2.44 (for U-235)

$$TF(139) = n_0 \frac{\frac{1.5^{139} - 1.5}{1.5 - 1}}{2.44} \sim 2.46 \text{ x } 10^{24} \text{ fissions}$$

Each fission gives off about 200 MeV or 3.2×10^{-11} Joules, so the energy produced after 139 generations is,

$$E = 2.46 \times 10^{24} \times 3.2 \times 10^{-11} = 7.9 \times 10^{13} \text{ J}$$

This is a large amount of energy. For an explosive, we can use the relationship that 1,000 t (or a kiloton) of TNT is 4.184×10^{12} Joules. Converting this energy to kilotons of TNT we find,

$$E = 7.9 \times 10^{13}/4.184 \times 10^{12} \sim 18.9 \text{ kt of TNT}$$

Fortunately, it is not easy to design a nuclear weapon. It is a sophisticated device that requires a great deal of technology. For instance, we know that Iraq invested more than 20 billion dollars in its nuclear weapons program and were not even close to getting any significant amounts of fissile materials.

We need to dispel a common concern about the relationship of nuclear weapons to nuclear power. A power nuclear power plant is nothing like a nuclear weapon. A nuclear power plant has a much different design criteria which does not allow large power excursions, thus k is kept at values below 1.1 allowing complete and safe control of the reaction.

It is not an easy task to design a critical assembly ($k > 1$) of fissile materials for a power producing reactor. There is much that goes into the design such as choice of materials, geometry, structural components, moderator, control systems, active safety systems, passive safety systems etc. The US for example requires that nuclear power plants have a negative temperature coefficient which means that the plant will shut itself down if the fuel exceeds a safe temperature. It is beyond the scope of this course to delve into this physics and system details. If the reader has a desire to learn more, there are courses on nuclear reactor engineering, nuclear reactor physics and transport theory that you may wish to take.

In addition to producing power, the fission reaction produces fission products. These products are radioactive (see Fig. 9.17) and produce residual heat even when the reactor is shut down. This residual heat is a problem which must be dealt with in the design of the power plant. The energy that is emitted by fission reactions is shown in Table 9.3.

Table 9.3. Emitted and Recoverable Energy for Fission of U-235

Energy form	**Emitted energy, MeV**	**Recoverable energy, MeV**
Fission fragments	168.00	168.00
Fission product decay		
β rays	8.00	8.00
γ rays	7.00	7.00
Neutrinos	12.00	0.00
Prompt γ rays	7.00	7.00
Fission neutrons (kinetic energy)	5.00	5.00
Capture γ rays	0.00	4.00
Total	207.00	200.00

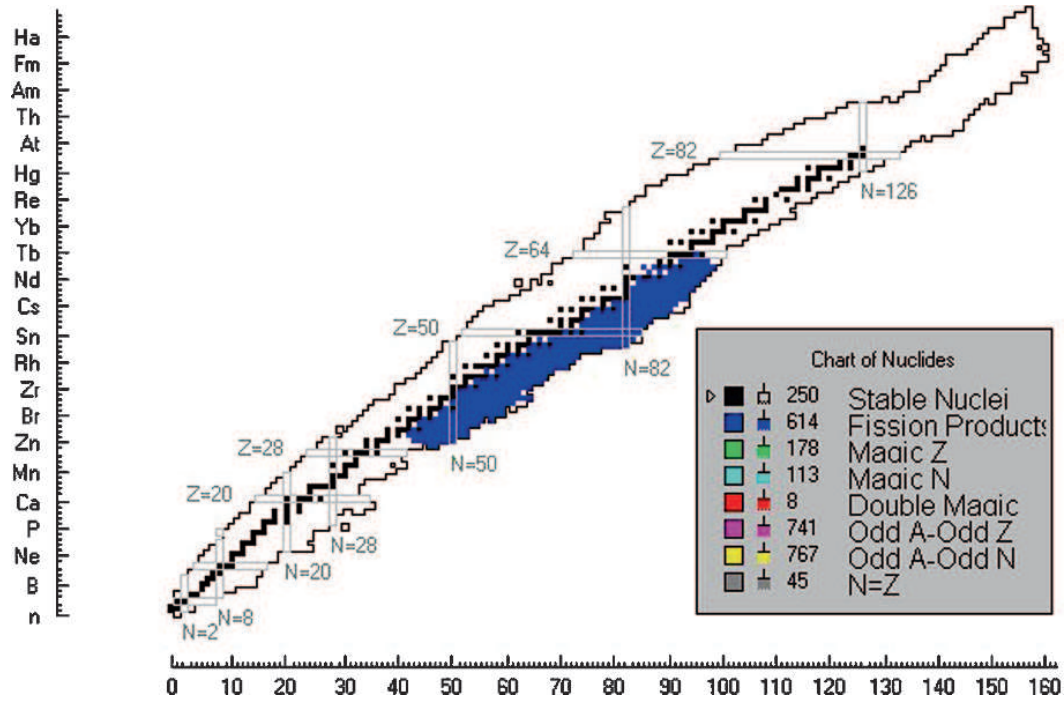


Fig. 9.17. Distribution of fission products. Most of the products are neutron rich and undergo radioactive decay (Courtesy of [2]).

Calculation of the amount of power that fission produces for a given amount of fuel consumed is very large. A simple calculation can be made relating the reactor power (P), to the fuel burnup, and fuel consumption:

$$\text{Fission Rate} = \text{P (MW)} \times \frac{10^6\, joules}{MW - \sec} x \frac{fission}{E_R\ MeV} x \frac{MeV}{1.6x10^{-13}\ joule} x \frac{86400\ \sec}{day}$$

$$\text{Fission Rate} = 5.40\,x10^{23}\,\frac{P}{E_R}\ fissions/day$$

The burnup rate can be found by assuming that mass of a fissile nuclei is A,

$$\text{Burnup Rate} = 5.40\,x10^{23}\,\frac{P}{E_R}\,x\,\frac{A}{0.602\,x10^{24}}$$

$$\text{Burnup Rate} = 0.895\frac{PA}{E_R}\ gm/day$$

For unranium-235, which releases 200 MeV per fission, the burnup rate is,

$$\text{Burnup Rate} = 1.05\ P\ \text{gm/day} \tag{9.49}$$

So, one MW-day of energy requires about 1 gm of uranium-235 per day.

The fissile material is consumed by both fission reactions and capture reactions (in which fission does not occur). The total absorption rate factor is 1.175. The total consumption rate is:

$$\text{Consumption Rate} = 0.895 \times 1.175 \times \frac{PA}{E_R}\ gm/day \tag{9.50}$$

Consumption Rate = 1.24 gm/day

The unit of megawatt-days per metric tonne of fuel, MWD/t, is used to describe consumption rate. A metric tonne is 1,000 kg or 10^6 gm. If it were possible to fission all uranium-235 the total energy release would be 1 MWD/gm which is equal to 106 MWD/T. The uranium-235 is not all consumed. There are parasitic absorptions that reduce this number slightly to 800,000 MWD/t.

9.4 Uranium Resources

About 439 commercial nuclear power reactors are operating in 30 countries, with 372,000 MWe of total capacity providing 16% of the world's electricity as base-load power (Fig. 9.18). Also there are 284 research reactors in 56 countries, and further 220 reactors that power ships and submarines.

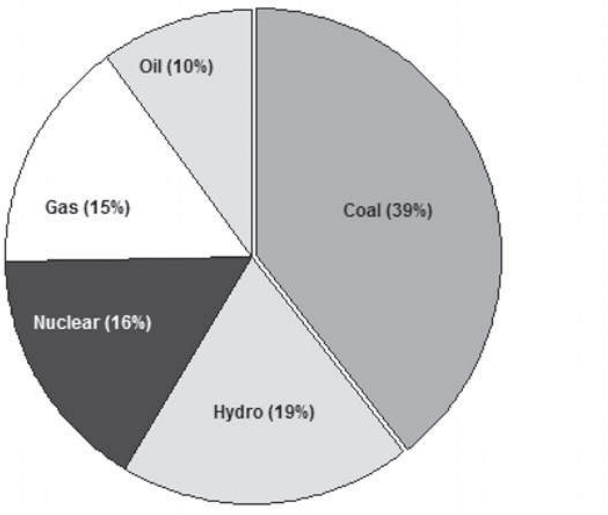


Fig. 9.18. World electricity generation by fuel type.

In Fig. 9.19 the share of electricity generated by nuclear energy by various countries is shown. France is leading all the countries with 80%.

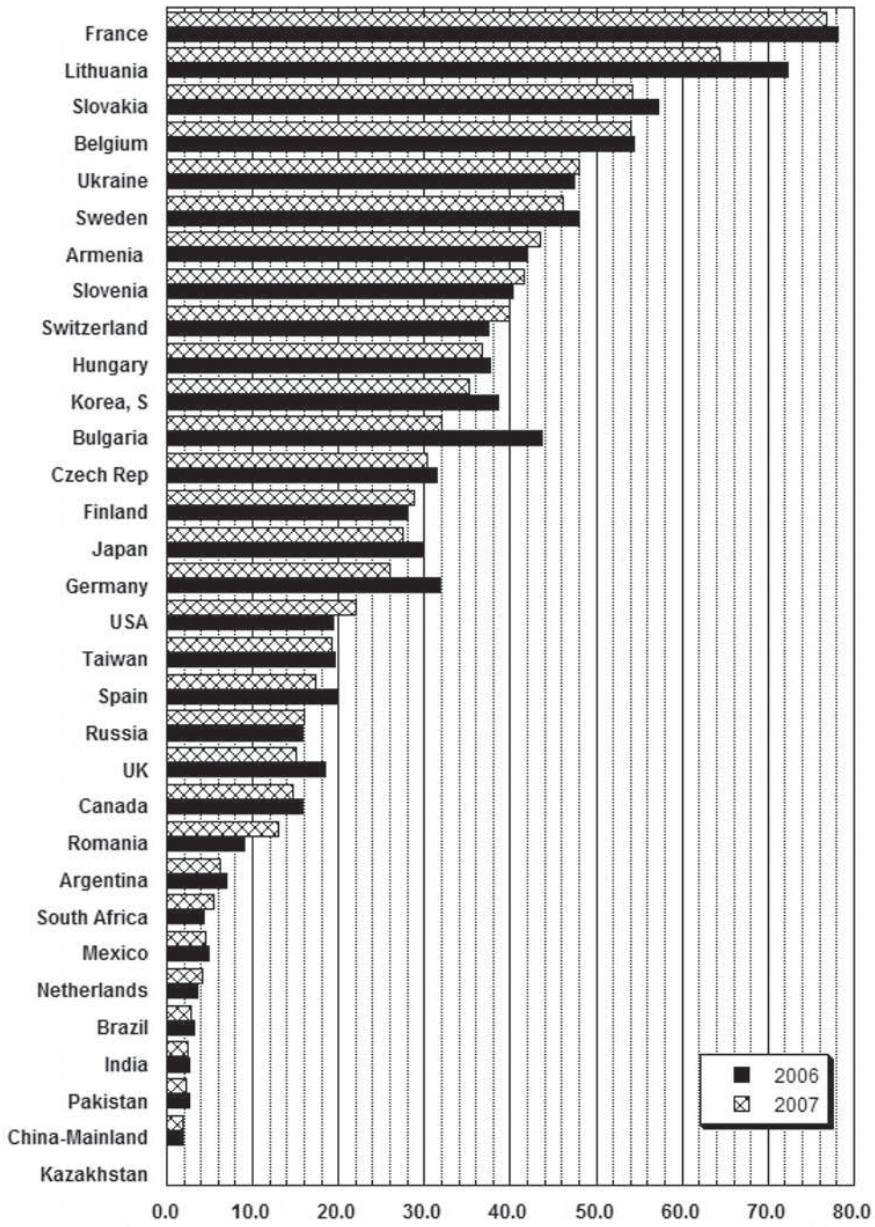


Fig. 9.19. Percentage of electricity generated by nuclear by various countries (Adapted from [5]).

As can be noted from Fig. 9.20, the share of nuclear towards total electricity production is steadily increasing. Interestingly, from 1990 to 2006, world capacity rose by 44 GWe, or 13.5%, due both to net addition of new plants and uprating of some established ones. Total electricity production increased to 757 billion kWh (40%). The relative contributions to this increase were: new construction 36%, uprating 7% and availability increase 57%. Nuclear power reactors that are under construction or planned for the near future are given in Appendix IX.

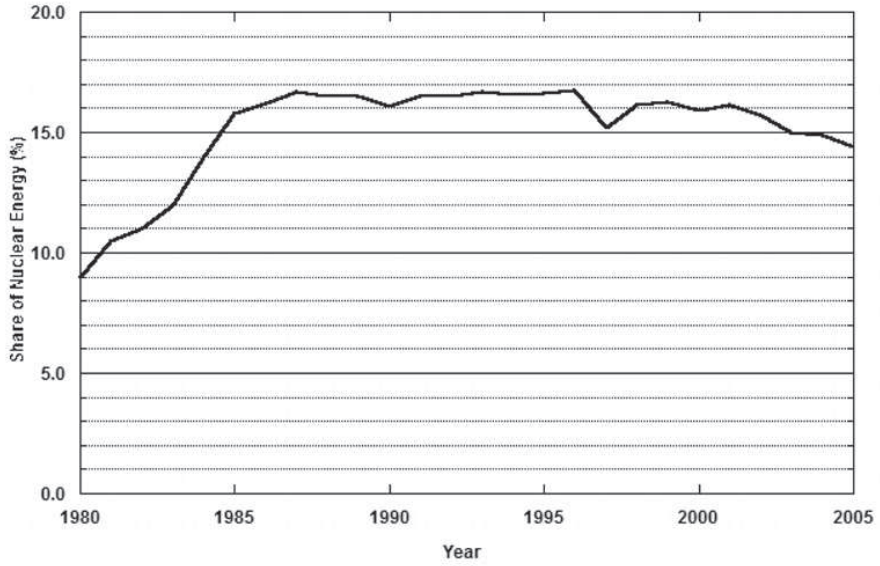


Fig. 9.20. Global share of electricity production by nuclear energy (Adapted from [5]).

The total generation of electricity by nuclear energy is steadily increasing worldwide. These data are shown in Fig. 9.21. A number of nuclear power plants are already under construction. Also several more plants have been planned worldwide which will further increase the share of total electricity production by nuclear energy. The plants under construction and that are planned are given in Appendix IX

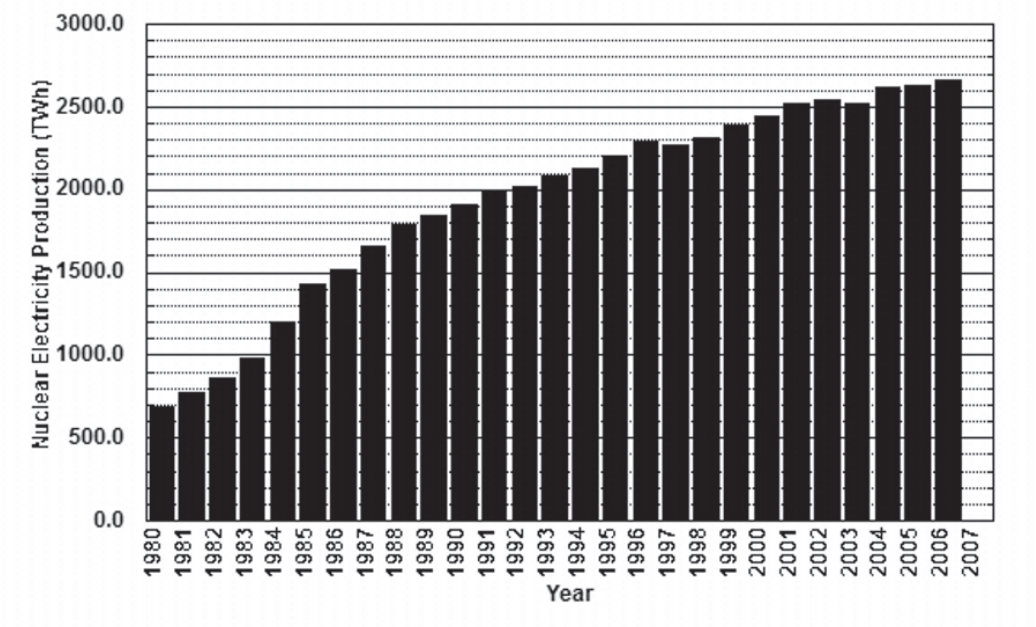


Fig. 9.21. Yearly nuclear electricity production in the world (Adapted from [5]).

9.5 Fuel Cycle

Sustainability of nuclear power production relies heavily upon the closing of the fuel cycle, i.e., reprocessing of the spent fuel. A number of countries including, Japan, France, and Russia have been reprocessing spent fuel from their light water reactor fleet for many years. The United States opted for the once-through cycle in the early 1970s, but has shifted its philosophy in recent years, as fossil fuel prices have continued to soar. Also, India and China are planning to reprocess the spent fuel from the nuclear reactors.

9.5.1 Once-Through Cycle

The fuel cycle begins with mining and milling of uranium, followed by conversion and enrichment. The fuel cycle is described in details later in this chapter. The enriched product is then sent to fabrication. The enriched product can be either in a metal or oxide form. The fuel is encased in a protective barrier known as cladding and arranged such that critical mass can be achieved within a core, under proper conditions. This fuel is then cycled through a reactor core, which usually amounts to three cycles of use, approximately 18 months each. Once the fuel has reached the end of its useful life, it is transferred to a storage pool, where it decays over a period of several years. Dry storage is also available for fuel that has decayed sufficiently and is necessary when spent fuel pools reach their capacity. This is the current stopping point of the current U.S. fuel cycle.

The next step in the once-through cycle would be the shipment of the spent fuel to a permanent repository as high level waste (HLW). However, no repository has been licensed for use and the only active project, Yucca Mountain, still has substantial political and scientific hurdles to overcome. The topic of waste generation and final storage is discussed later.

9.5.2 Closed Fuel Cycle

The front end of the closed fuel cycle is the same as that of the once-through cycle. However, the back end of the closed fuel cycle is substantially different. Instead of simply storing the spent fuel in a High Level Waste (HLW) disposal facility, the fuel is reprocessed for recycle back into the fuel cycle. The reprocessing involves separating the fuel material from the cladding material, retrieving the fissionable and fissile materials from the mixture, mainly plutonium and uranium, for recycle. The rest of the long lived fission products are vitrified for long term storage. The uranium and plutonium can be recycled back into fuel for light water

and fast reactor use. Additionally, both thermal reactors and fast breeder reactors can make use of the U-238, which is present in 99% of natural uranium. Thermal reactors will convert some of the U-238 into plutonium some of which is burned during operation of the reactor. About 1/3 of the energy produced in a light water power reactor is derived from the plutonium generated by the converted U-238. Fast breeder reactors on the other hand produce more fissionable material than they consume, providing a more efficient means of uranium use. The fuel-resources for nuclear reactors can be extended if reprocessing and fast breeder reactors are used instead of once-through cycles.

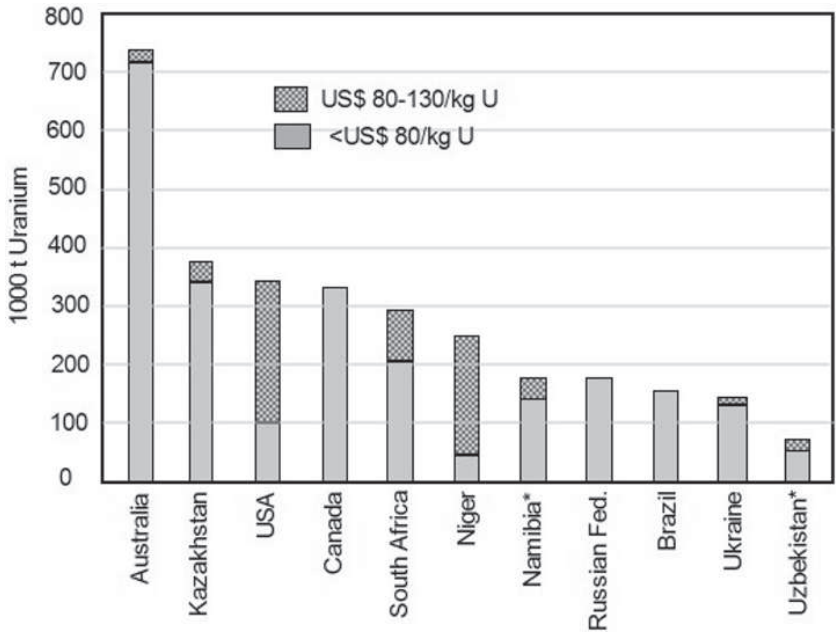
Annual requirements of uranium worldwide for existing power reactors are about 67,000 t of uranium. According to the authoritative "Red Book" [6] produced jointly by the OECD's (Organisation for Economic Co-operation and Development) Nuclear Energy Agency (NEA) and the UN's International Atomic Energy Agency (IAEA), the world's present known economic resources of uranium, exploitable at below $80 per kilogram of uranium, are about 3.5 million tonnes. This amount is therefore enough to last for 50 years at today's rate of usage – a figure higher than many widely used metals. Current estimates of all expected uranium resources (including those not yet economic or properly quantified) are four times as great, representing 200 years' supply at today's rate of usage. It can be seen from Table 9.4 that Australia has a substantial amount (about 23%) of the world's low-cost uranium, followed by Kazakhstan 15%, and Russia 10%.

Table 9.4. Known recoverable resources of Uranium (2007 data).

Country	Tonnes U	% of world	Country	Tonnes U	% of World
Australia	1,243,000	23%	Niger	274,000	5%
Kazakhstan	817,000	15%	Uzbekistan	111,000	2%
Russia	546,000	10%	Ukraine	200,000	4%
Canada	423,000	8%	Jordan	112,000	2%
USA	342,000	6%	India	73,000	1%
South Africa	435,000	8%	China	68,000	1%
Namibial	275,000	5%	Mongolia	62,000	1%
Brazil	278,000	5%	Other	210,000	4%
World total	5,469,000				

Source: World Nuclear Association, London, UK [5]. Reasonably Assured Resources plus Inferred Resources, to US$ 130/kg U, 1/1/07, from OECD/NEA & IAEA, *Uranium 2007: Resources, Production and Demand*, "Red Book" [6].

The amount of uranium that can be extracted from the ores depends on the uranium price. As can be seen from Fig. 9.22, in countries such as the USA and Niger, significant amount of uranium can be recovered at a cost of US$ 130/kg of U.



* IAEA estimate

Fig. 9.22. Reasonably Assured Resources plus Inferred Resources, to US$ 130/kg U [6]. Note: The Australian figure had risen to 1,558,000 t U as at August 2007 and other country figures would have risen also, but are not yet published. 1/1/05.

It is expected that the demand for uranium will increase as more and more reactors are constructed worldwide. With this being anticipated, the investment for exploration of uranium resources has increased significantly in recent years (Fig. 9.23).

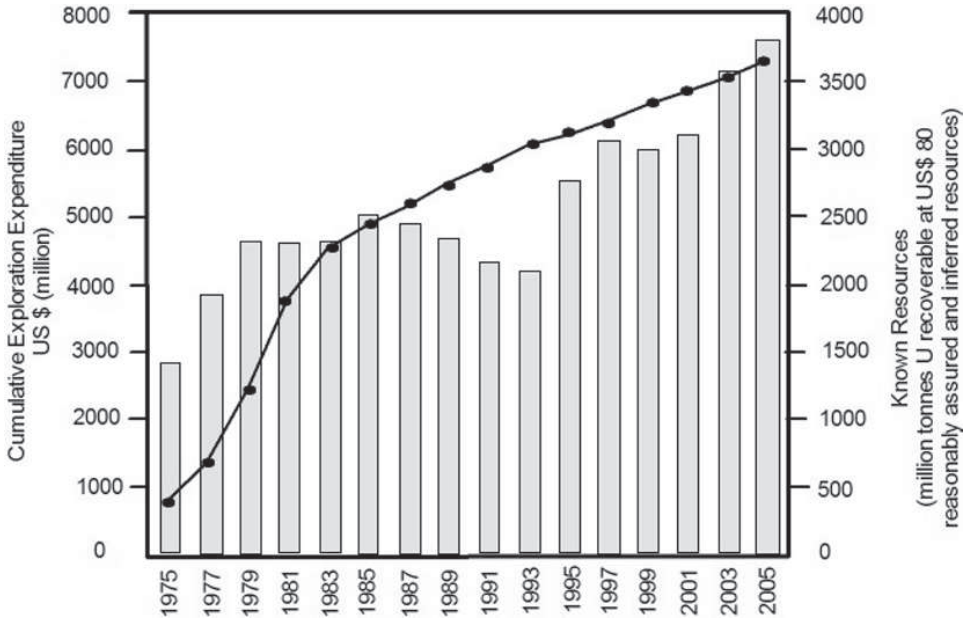


Fig. 9.23. Investment for exploration of uranium (Adapted from [5]).

The IAEA-NEA estimates that if all conventional resources are considered, there are about 10 million tonnes of uranium beyond the 4.7 million tonnes known economic resources, which is enough to go beyond 200 years' at today's rate of consumption. However, if unconventional resources such as phosphate/phosphorite deposits (22 Mt U recoverable as by-product) and seawater (up to 4000 Mt) are considered, the available resources would be even greater. These unconventional resources are currently not economical, but as the price of uranium increases these resources may become attractive.

The use of the fast breeder reactor would increase the utilization of uranium 50-fold or more. This type of reactor can be started up on plutonium derived from conventional reactors and operated in closed circuit with its reprocessing plant. Such a reactor, supplied with natural or depleted uranium for its "fertile blanket", can be operated in such a way that each tonne of ore yields 60 times more energy than a conventional reactor. However, reprocessing and the use of fast breeder reactors can increase the operation of nuclear reactors significantly (Table 9.5).

Table 9.5. Years of uranium availability for nuclear power.

Reactor/Fuel Cycle	**Years of 2004 world nuclear electricity generation with identified conventional resources**	**Years of 2004 world nuclear electricity generation with total conventional resources**	**Years of 2004 world nuclear electricity generation with total conventional and unconventional resources**
Current once through fuel cycle with light water reactors	85	270	675
Pure fast reactor fuel cycle with recycling	5, 000 – 6, 000	16, 000 – 19, 000	40, 000 – 47, 000

The values in the last row assume that fast reactors allow essentially all U-238 to be bred to Pu-239 for fuel, except for minor losses of fissile materials during reprocessing and fuel fabrication. Adapted from Rogner H-H, McDonald A (November 2007) Nuclear Energy - Status and Outlook, 20th World Energy Congress, Rome, Italy [7].

9.6 Uranium Supply and Demand

Uranium is currently extracted in 19 countries, and Canada and Australia account for more than 50% of the worldwide production. Approximately 40,000 tons of uranium is produced each year, supplying approximately 60% of the uranium needed for the existing reactors. The rest of the uranium is made up by secondary uranium sources, such as natural and enriched stockpiles, re-processing of spent fuel and enrichment of uranium tails [7].

Demand for uranium has gone up slowly over the past three decades. The upward trend has been a result of expansion of nuclear power overseas and more recently, the pending demand from mass global expansion of nuclear power. Uranium prices have risen more dramatically in the past few years, generating renewed interest in new exploration. Figure 9.24 shows recent trends in uranium requirements and production.

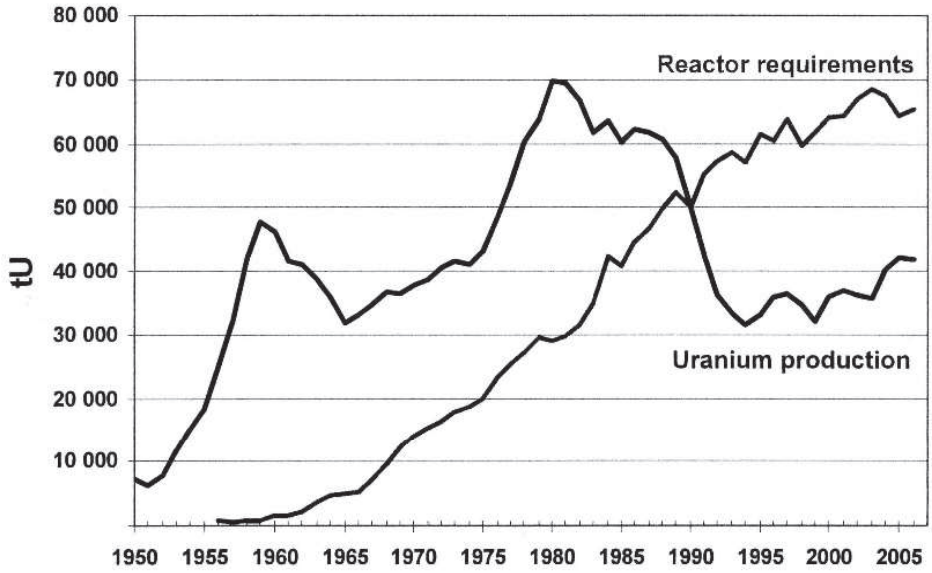


Fig. 9.24. Global Annual Uranium Production and Reactor Requirements, 1950–2006 (Adapted from [7]).

It is assumed that the need of uranium will be between 78,000 and 129,000 t of uranium by 2030 [7] due to proposed and anticipated new reactors. This will certainly force uranium prices up and encourage even more mine developments worldwide.

9.7 Electricity Generation from Nuclear Energy

The steps or processes involved in generation of electricity from nuclear energy may be divided into following categories. This is also called the Nuclear Fuel Cycle.

- Uranium mining and extraction
- Enrichment
- Fuel Fabrication
- Power production
- Waste management
- Spent fuel reprocessing (If the objective is to recycle unused/unburned uranium)

The nuclear fuel cycle is shown graphically in Fig. 9.25.

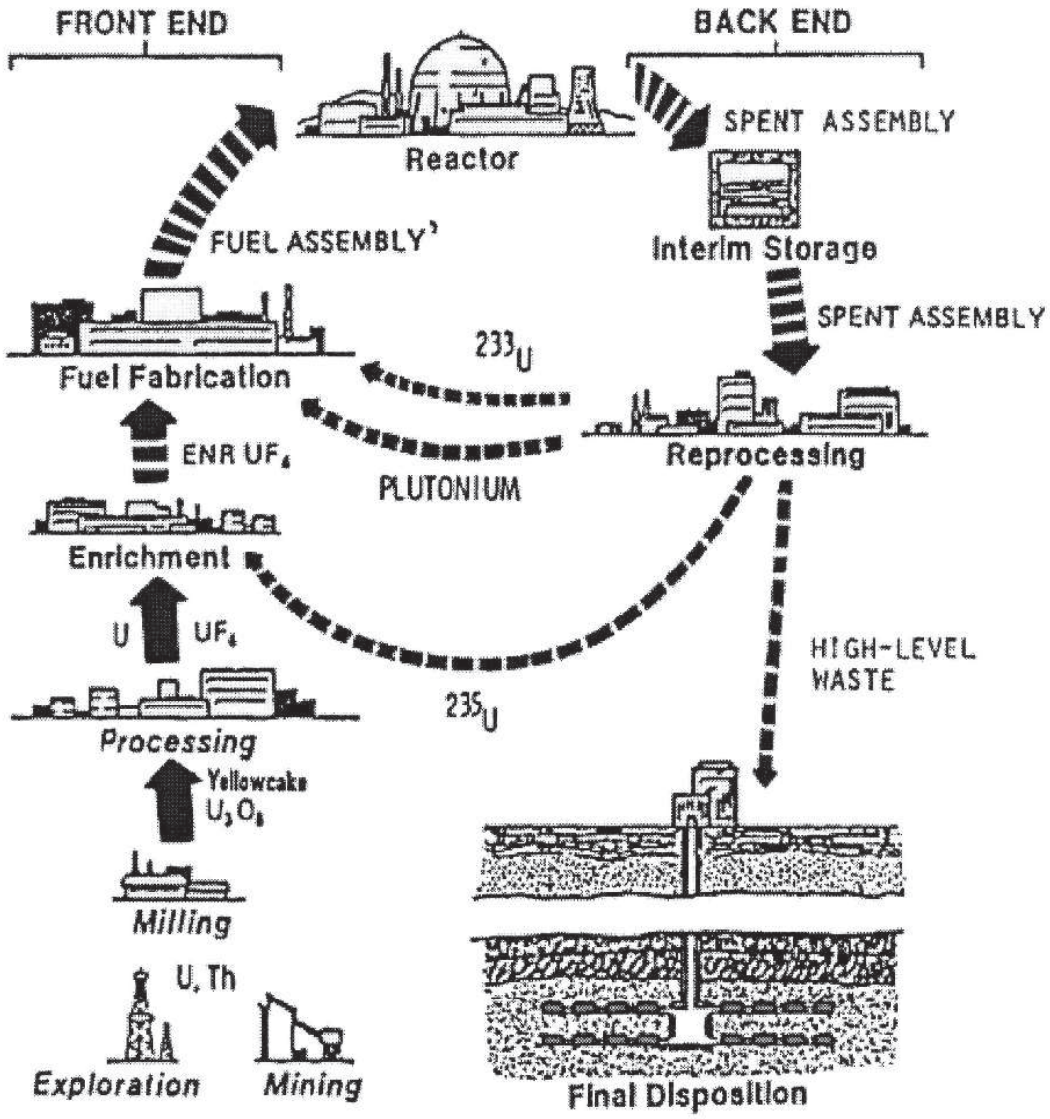


Fig. 9.25. Nuclear fuel cycle (Adapted from [8]).

9.7.1 Uranium Mining and Extraction

Uranium can be found in nature in several ores. Uranium has been found as a significant constituent in about 150 different minerals and in small concentration in another 50 minerals. Primary uranium minerals include uraninite and pitchblende. Other ores from which uranium may be extracted economically include autunite, tobernite, coffinite, and carnotite.

It is estimated that about 90% of world's low-cost uranium reserves are in Canada, South Africa, the United States, Australia, Niger, Namibia, Brazil, Algeria, and France. Sandstone formations in the Colorado Plateau and Wyoming Basin of the western United States also contain significant reserves of uranium.

9.7.1.1 Mining and Milling

Detailed exploration is necessary prior to any major investment being placed in mining. There are several methods of exploration utilized today, including geological mapping, airborne and surface surveys, hydro-chemical sampling, well-logging and botanical methods [9].

Methods of Mining

Open-pit and underground mining remain the dominant extraction methods in most of the world, accounting for 68% of production. Additionally, approximately 11% of uranium is produced from in-situ leaching technology, which is the preferred method in several countries, such as Kazakhstan. The remainder of fresh uranium supply is produced as a by-product material from copper and gold mining and a small amount is produced from water treatment operations [6].

Open-pit and underground mining are essentially the same as that used for other mineral extraction, such as coal. In-situ leaching is a more complex process. Several holes are drilled, approximately 50 ft apart, and a leaching solution is injected into the center hole. This solution consists of water, an oxidant and an ionic complex agent, designed to mobilize and dissolve uranium. The solution is pumped out from the remaining holes and uranium is removed through ion exchange. The stripped solution is re-oxidized and treated for proper pH, and is recycled back into the process. This method is not without problems, as it can contaminate groundwater, if not performed properly. To prevent this, more water is pumped out than is pumped in, creating a flow away from the water table. Once the mine has been fully utilized, it is flushed with clean water and all holes are plugged. If performed properly, in-situ mining has less environmental impact on mining sites than open-pit or underground mining and is highly effective for extracting uranium.

Following uranium mining from open-pit or underground methods, the ore is crushed, grinded, pulverized into powder and roasted to remove organics. The uranium is removed via ion exchange or solvent extraction. The resulting U_3O_8 product is then dried and packaged as yellowcake into 55 gallon drums for shipment to an enrichment facility.

Mining Wastes

Open-pit and underground mining produce significant amounts of tailings from the milling process. All of the material left over after the yellowcake is produced, remains as waste. These tailings typically contain some uranium as well as radon and other decay products. Additionally, the tailings contain acids from the milling process, which can leach out various toxic metal from soil and cause groundwater contamination. Strict enforcement of discharge limits are enforced by the Nuclear Regulatory Commission (NRC) and Environmental Protection Agency (EPA) in the United States.

Mill tailings must be sufficiently contained within tailing ponds, so that acids and radioactive materials do not escape into the environment. Several advances have been made in recent years, considering tailings were not regulated during the early years of uranium production.

In-situ mining also has several disadvantages. Although this type of mining does not produce tailings, the groundwater aquifer near these mines must be restored to similar conditions as they were before in-situ extraction began. This can be a rather challenging undertaking and has been the source of criticism for this method of mining.

Another source of waste generated by all of these mining techniques revolves around the milling treatment and the chemicals involved. Acids are employed for separation of uranium from the ore as well as stripping of resin columns. Additionally, solvent extraction utilizes various chemicals, which must also be disposed of properly.

9.7.2 Conversion and Enrichment

Although the processes involved in conversion and enrichment have not changed much over the past two decades, there are some notable exceptions and future trends.

Prior to fuel enrichment, the U_3O_8 produced in the mill must be purified and converted into uranium hexafluoride (UF_6). UF_6 was chosen as the enrichment compound due to its unique physical and chemical properties. It is a solid at room temperature, and its melting point is 147°F. This is the only compound of uranium which behaves in this fashion and has made it the ideal candidate for enrichment. Properties of various uranium compounds are given in Table 9.6. As can be seen from the table, among all the uranium compounds, only uranium hexafluoride has the melting point that is low enough to use in the separation process.

Table 9.6. Melting points of uranium and uranium compounds.

Uranium compounds	Melting point (°F)
Uranium metal	2071
Diboride	4289
Tetrabromide	960
Tribromide	1346
Dicarbide	4262
Tetrachloride	1094
Trichloride	1548
Hexafluoride	147
Tetrafluoride	1760
Tetraiodide	943
Mononitride	4766
Dioxide	4532
Disulfide	>2012

Source: Adapted from [10].

Various physical forms of UF_6 at various temperatures and pressures are expressed by its phase diagram. The boundaries between solid, liquid, gaseous forms detect the operating conditions. The phase diagram of UF_6 is shown in Fig. 9.26.

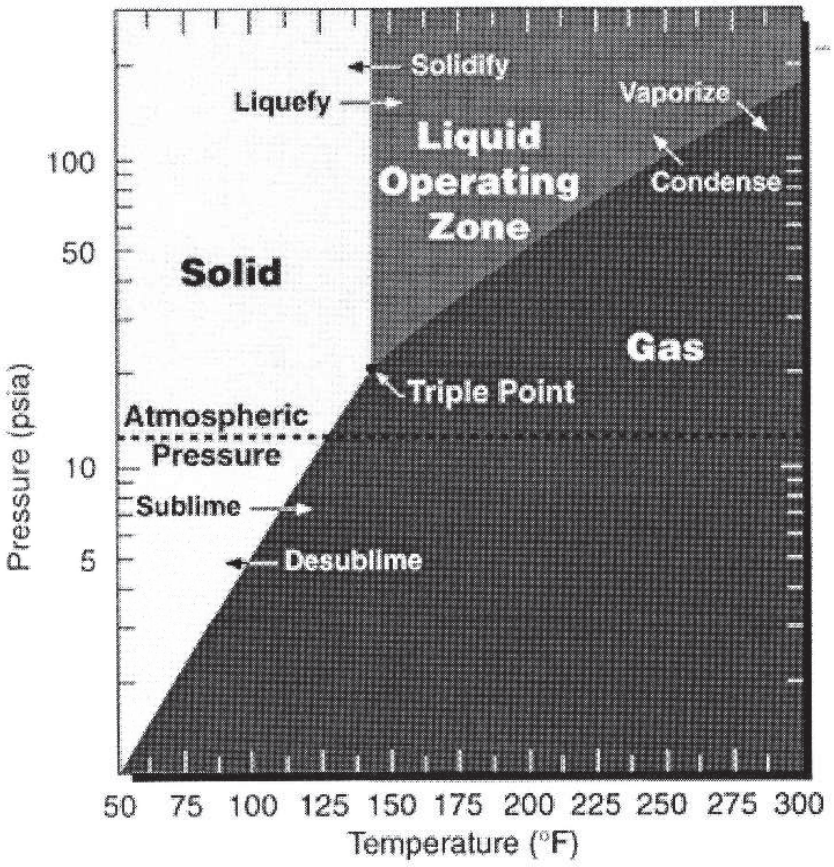


Fig. 9.26. The phase diagram of UF_6 (Adapted from. [10]).

A solvent extraction process, called Plutonium Uranium Extraction (PUREX), is used for extraction and purification of uranium. The PUREX process takes advantage of the preference of uranium to form coordinated compounds within aqueous solutions that can be selectively extracted from the solution by an organic solvents, such as tributyl phosphate (TBP). Uranium is next precipitated out from the solution using ammonium hydroxide. The second method involves using hydrogen peroxide to precipitate out uranium peroxide from weak acid solutions. Steps involved in producing UF_6 from yellowcake are shown in Fig. 9.27, and a simplified diagram of the solvent extraction unit is shown in Figure 9.28. Once these methods have been successfully completed, conversion to UF_6 follows [9].

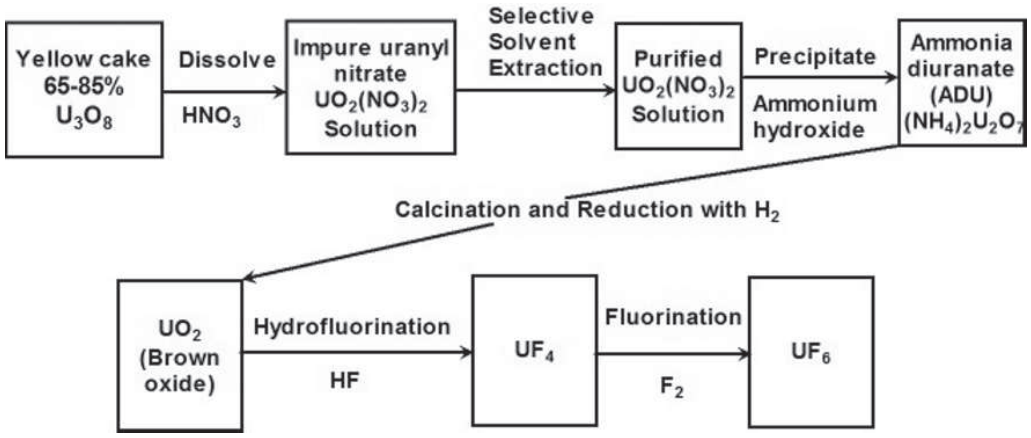


Fig. 9.27. Reactions pathways for UF_6 production (Adapted from [10]).

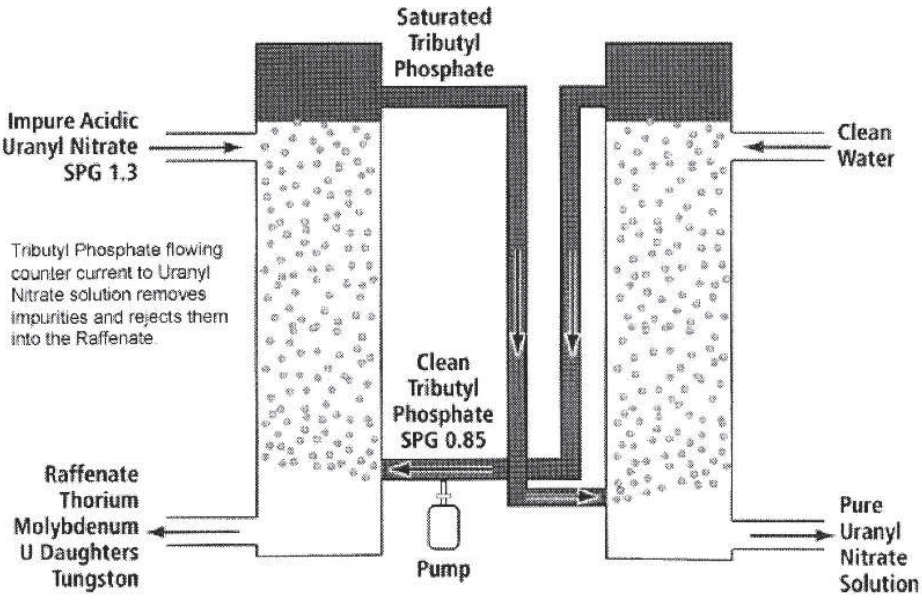


Fig. 9.28. A simplified schematic diagram of the solvent extraction unit (Adapted from [10]).

Uranium diuranate (ADU) that is precipitated out from the solution is converted to UO_2 by calcining it in hydrogen. UO_2 is next converted into UF_6 in two steps. Both steps utilize a fluidized bed reactor, which heats the UO_2 to approximately 1,000°F in the presence of HF first and then with F_2. The chemical reaction that takes place is:

$$UO_2 + 4HF \rightarrow 2H_2O + UF_4$$

This UF_4 is next reacted with fluorine gas to form the UF_6 as follows:

$$UF_4 + F_2 \rightarrow UF_6$$

The solvent extraction step removes impurities prior to the reduction step. This results in a very pure form of UF_6 at the end of the fluorination process, leaving a product ready for enrichment [9].

9.7.2.1 Enrichment

The enrichment of uranium refers to the increase in concentration of U-235 isotopes. Depending on the degree of enrichment, its grade varies. Also a number of enrichment processes have been suggested, but only few are used commercially.

Grades
- Highly enriched uranium (HEU)
- Low-enriched uranium (LEU)
- Slightly enriched uranium (SEU)

Methods
- Thermal diffusion
- Gaseous diffusion
- The gas centrifuge
- The Zippe centrifuge
- Aerodynamic processes
- Electromagnetic isotope separation
- Laser processes
- Chemical methods
- Plasma separation

Enrichment Process

Among the methods mentioned above, gaseous diffusion and gas centrifuge methods are most widely used commercially for enrichment of uranium. Laser separation is still in developmental stage, but is likely to compete with gaseous diffusion and

gas centrifuge methods in the near future. The United States has relied primarily on the gaseous diffusion method due to its more simplistic design and successful operational history. However, currently, most of the countries preferring gas centrifuge method.

Gaseous Diffusion

The first step in the diffusion process is conversion of solid UF_6 to the gaseous form by heating it above 135°F. It is then forced through a series of porous membranes, which separates U-235 from the U-238. Since U-235 is lighter than U-238, it diffuses at a relatively faster rate than U-238 through the membranes. Figure 9.29 illustrates the diffusion process during uranium enrichment. To achieve a U-235 concentration from its natural 0.7% to 3% to 5%, several thousand stages are connected in series.

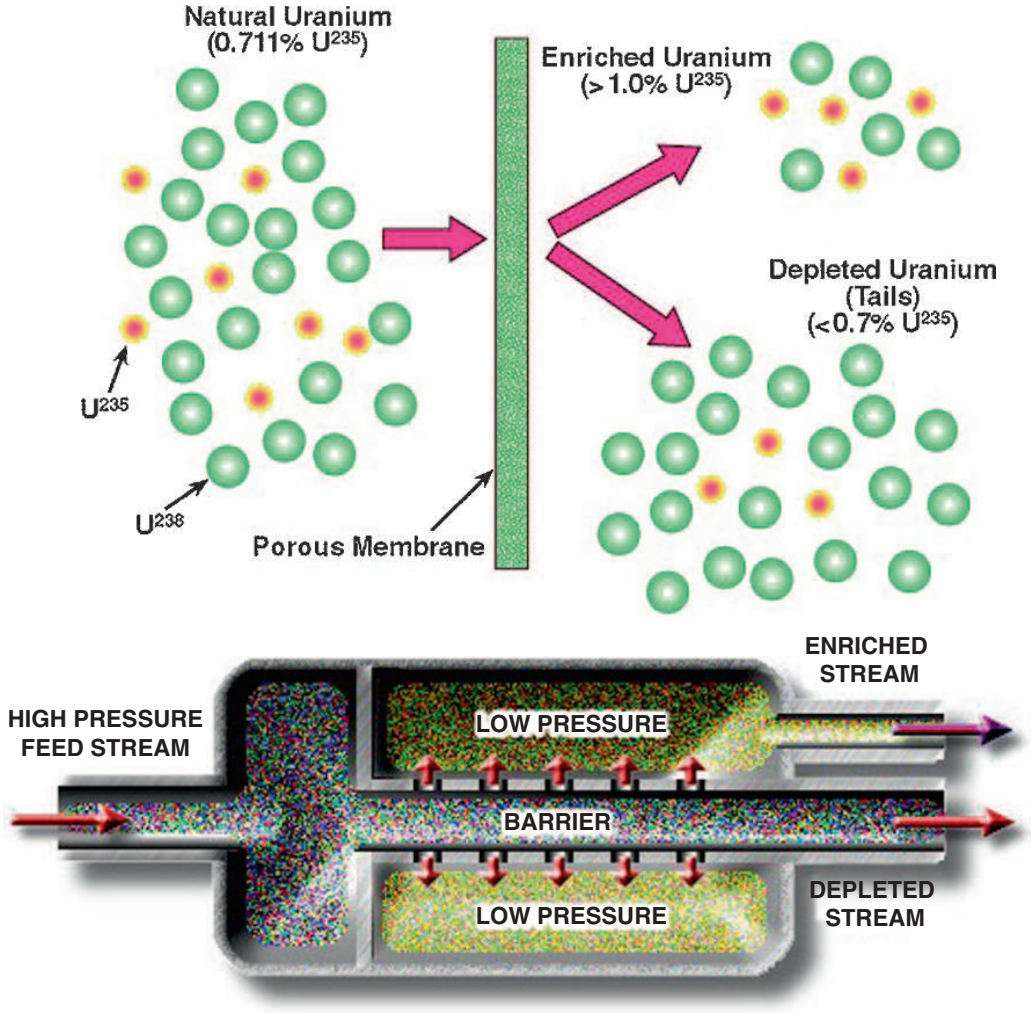


Fig. 9.29. An illustration of gaseous diffusion process for uranium enrichment and a diffusion cell (Adapted from [11]).

Gas Centrifuge

In this method, solid UF_6 is also heated first to form gaseous UF_6, which is then fed into the center of a rotor, spinning at very high velocity, inside a casing held at vacuum. Due to the mass differences, the heavier U-238 tends to separate from the lighter U-235 and is forced outward. The enriched product is then collected from the center and fed into the next centrifuge. The depleted UF_6 is fed back into the system for further separation. In order to achieve U-235 percentages adequate for reactor fuel, several thousand stages must be connected in series. Figure 9.30 illustrates a typical centrifuge unit used for fuel enrichment.

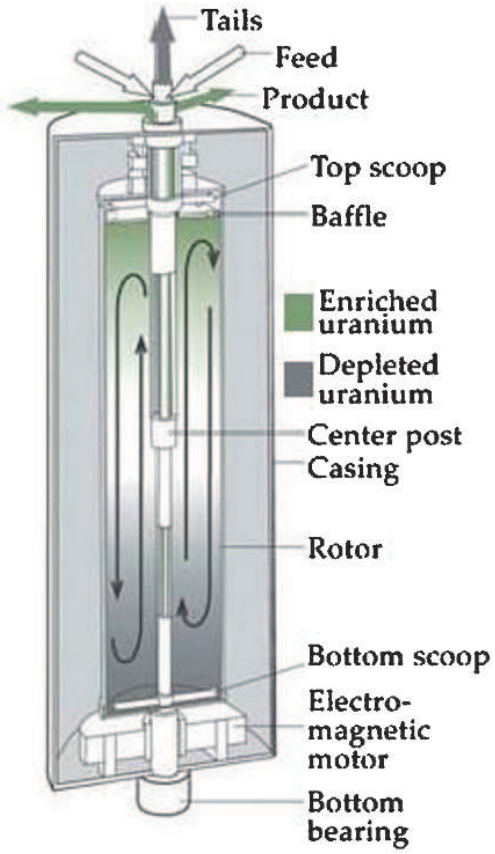


Fig. 9.30. Gas centrifuge unit for uranium enrichment (Adapted from [12]).

The centrifuge technology has been used worldwide with great success. The US Energy Corp (USEC) and AREVA are planning to build a new facility in New Mexico, USA, based on the centrifuge process, and another one near Idaho Falls, ID, USA. One of the big advantages of centrifuge technology is that it has a 5% energy savings over gaseous diffusion. Enrichment plants are extremely energy intensive, and savings from the centrifuge process are very significant. The US project will be the most advanced fuel enrichment plant in the world, employing the new AC100 centrifuge machines. Details of these units are classified, but the USEC expects the plant to produce 3.8 million separative work units (SWU) per year by 2012, using 11,500 machines, once the plant is fully on-line [13].

About 100,000–120,000 SWU is required to enrich the annual fuel loading for a typical 1,000 MWe light water reactor. Enrichment costs are substantially related to electrical energy used. The gaseous diffusion process consumes about 2,500kWh (9,000 MJ) per SWU, while modern gas centrifuge plants require only about 50 kWh (180 MJ) per SWU. Also, the current trend is to use the centrifuge technology over the diffusion technology (Table 9.7).

Table 9.7. Trends in the use of technology for uranium enrichment.

Supply source	2007	2017
Diffusion	25%	0
Centrifuge	65%	96%
HEU ex weapons	10%	4%

Source: Reference [5].

(The Separative Work Unit (*SWU*) is a measure of the work expended during an enrichment process. Uranium enrichment is sold as *SWU*. Higher levels of U-235 require more *SWU*).

The *SWU* may be calculated from the following equation:

$$SWU = P{\cdot}f(N_p) + W{\cdot}f(N_w) - F{\cdot}f(N_f) \tag{9.51}$$

where P is the amount of product, N_p is the product concentration, W is the amount of waste, N_w is the waste concentration, F is the amount of feed, and N_f is the feed concentration, and $f(x)$ is a value function and is defined as:

$$f(x) = (2x - 1)\ell n \frac{x}{(1-x)} \tag{9.52}$$

where x is a given concentration.

(The function $f(x)$ is dimensionless. The unit of *SWU* depends on the units of P, W, and F, which are generally expressed in units of kilograms (kg) or tonnes. The *SWU* is expressed as a kg-*SWU* or metric tones-*SWU*. The performance of a centrifuge can be expressed in terms of rate of enrichment, that is, *SWUs* per year or month. Individual centrifuges might be described in terms of kg-SWUs per year. A *SWU* per unit time is referred to, not as separative work, but as separative power).

Laser Separation

This technology differs substantially from the diffusion and centrifuge methods. Although the different isotopes of uranium have identical chemical behavior, their electronic energies are very different, causing them to absorb energy at different

wavelength. This is the fundamental basis behind laser technology, enabling isotopic separation. The laser is focused through the target, with the wavelength tuned specifically into that for U-235. The targeted U-235 atoms will lose an electron (ionize), with no affect on the rest of the feed material, mainly U-238. The U-235 is collected in a negatively charged surface in the liquid form, where it runs down to a die cast. The resulting pellets can then be shipped off to fuel fabrication. The tailings, which amount to 30% less than with other methods, are collected and can be disposed of or used in the fast reactor. Advantages of this technology include reduced tailings and laser separation uses just 5% of the energy required by gaseous diffusion [14]. The uranium feed for laser separation is in the metal form, eliminating the hazards associated with UF_6. Figure 9.31 illustrates the laser separation method for uranium enrichment.

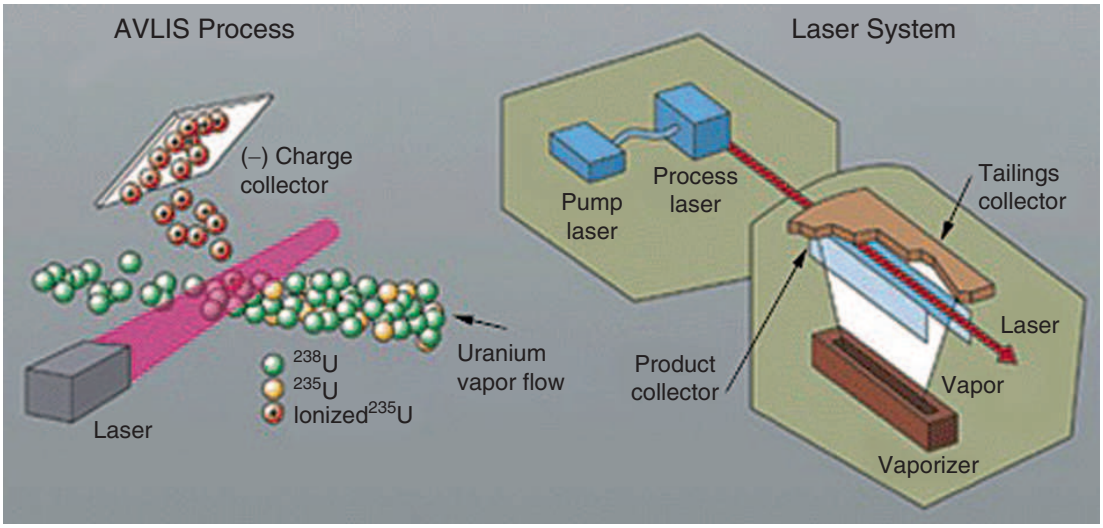


Fig. 9.31. Laser isotope separation method for uranium enrichment [15].

Scientists at Lawrence Livermore National Laboratory had demonstrated production scale capability of this enrichment process. However, the technology was transferred to USEC in the early 1990s, and they discontinued further development of the project in 1999 primarily due to declining uranium prices. Currently, Silex System Ltd, New South Wales, Australia, is further developing a laser based technology based on CRISLA (condensation repression by isotope selective laser activation) for full scale commercial application. The CRISLA process was invented by Dr. Jeff Eerkens, University of Missouri, Columbia, Missouri, USA. GE-Hitachi recently purchased the exclusive rights to the technology and moved the effort to their Wilmington, North Carolina location. Exelon and Entergy, which are the two largest nuclear utility owners in the United States, have signed letters of intent to buy laser-enriched uranium from GE [5].

9.7.2.2 Waste Generated by Enrichment Processes

The tailings of uranium enrichment are a key factor that goes into the price of SWU. Typically, tailings contain 0.2–0.3% by weight of U-235. There is approximately 700 million kilograms of depleted UF_6, containing 475 million kilograms of uranium in the United States. This is stored in 60,000 steel cylinders at the various laboratories and enrichment facilities. Recycling of this depleted UF_6 could be an economical and environmental success, which has been looked into by scientists at Lawrence Livermore National Laboratory [13].

9.7.3 Fuel Fabrication

Fuel fabrication varies greatly from one reactor to another, in terms of fuel and cladding type, fuel enrichment and geometry. The types of commercial reactors currently used for electricity generation are listed below.

- Research reactors
- Pressurized water reactor, (PWR)
- Boiling water reactor (BWR)
- Fast breeder reactor (FBR)
- Pressurized Heavy Water Reactor (PHWR) or CANDU

Fuels are produced in either the metal or ceramic form. Metal fuel has the benefits of higher thermal conductivity and ease of fabrication. Among the drawbacks of metal fuel are lower melting temperatures and its tendency to undergo growth upon irradiation. Ceramic fuels have the benefit of providing good retention of fission products during irradiation; low fabrication costs, and are chemically and structurally stable. Drawbacks include a brittle structure, susceptible to cracking and low thermal conductivity. There has been substantial research and development regarding advanced fuels and will be discussed further later in this chapter.

9.7.3.1 Ceramic Pellet Fuel

Commercial light water and CANDU reactors typically use UO_2 pellet fuel (MO_X fuel can also be used in these reactors). There are wet and dry processes available to convert the enriched UF_6 to ceramic grade UO_2. The three methods that are most widely used are the dry conversion, ammonium diuranate and the ammonium uranyl carbonate routes.

The dry conversion route begins with UF_6 decomposition using superheated steam to form UO_2F_2 particles. The particles then undergo pyrohydrolysis through interaction with steam and hydrogen. All these steps occur within a rotary kiln, which is the most common method of conversion in the U.S. and France [16]. The reactions that take place within the kiln are:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$
$$UO_2F_2 + H_2O \rightarrow UO_3 + 2HF$$
$$UO_3 + H_2 \rightarrow UO_2 + H_2O$$

The ammonium diuranate route involves hydrolysis of UF_6, filtration of the resulting precipitate and heating to form UO_2. Ammonium hydroxide is mixed in with the feed UF_6 to form the ammonium diuranate, which can readily be filtered, washed to remove as much entrained fluoride as possible and dried to remove moisture and excess ammonia. The product is then passed on to a pyrohydrolysis and reduction furnace, which removes the remaining fluorine and completes conversion to UO_2. The reactions that take place during this process are:

$$UF_6 + 2\ H_2O \rightarrow UO_2F_2 + 4HF$$
$$2\ UO_2F_2 + 8\ HF + 14\ NH_4OH \rightarrow (NH_4)_2U_2O_7 + 12\ NH_4F + 11\ H_2O$$

The ammonium uranyl carbonate route is used by Germany and Sweden. Advantages of this technology include production of a free-flowing, granular UO_2 with uniform particle size. The afore-mentioned method requires compaction and granulation prior to pressing into pellets. The UF_6 feed is heated and directed to a precipitation vessel, which contains a mixture of ammonium hydroxide and ammonium carbonate. The uranium is precipitated out as ammonium uranyl carbonate. The solid is then filtered out of solution, washed with clean ammonium carbonate to remove most of the fluoride, then dried and transferred to a furnace, supplied with hydrogen and steam at 650°C. The remaining fluoride is removed as HF and decomposition of the ammonium uranyl carbonate produces UO_2. The reactions that take place during this process are:

$$UF_6 + 5\ H_2O + 10\ NH_3 + 3\ CO_2 \rightarrow (NH_4)_4UO_2(CO_3)_3 + 6\ NH_4F$$
$$(NH_4)_4UO_2(CO_3)_3 + H_2 \rightarrow UO_2 + 3\ CO_2 + 4\ NH_3 + 3\ H_2O$$

Most of the light water fuel used today is in the form of small cylindrical ceramic pellets. Enriched UO_2, from the previously mentioned processes, is mixed with an organic pore-forming agent, which will later decompose during the sintering process. The pores that are formed provide the space for collection of fission products produced during irradiation, preventing fuel growth beyond tolerances. A binding agent is also added to the powder prior to pressing, which results in a

"green" pellet prior to sintering. The selection of binding agent is crucial since this determines the amount of handling damage that will occur to the pellet prior to sintering, thus lowering rejection rates. Pressure in the range of 3 to 4 tons/cm^2 is applied to the powder-mixture within a die until the resulting pellet density is approximately 5.5–6.0 g/cm^3. The final process of pellet production is sintering, which is performed in ovens at temperatures of approximately 1750°C [16]. The sintering converts the green pellet into the ceramic fuel with a crystalline grain structure while retaining the original shape. The pellets are then grounded to strict tolerances and the removed material is recycled back into the process.

9.7.3.2 Research Reactor Fuel Assembly

The Training, Research, Isotopes, General Atomics (TRIGA) reactor is the most widely used research reactor throughout the world. It can be built with thermal power levels ranging from 10 kW to 10 MW. A photograph of the fuel pellet and a fuel pin of the TRIGA reactor is shown in Fig. 9.32. Depending on the type of the research reactor, the fuel assembly is also different and is shown in Fig. 9.33.

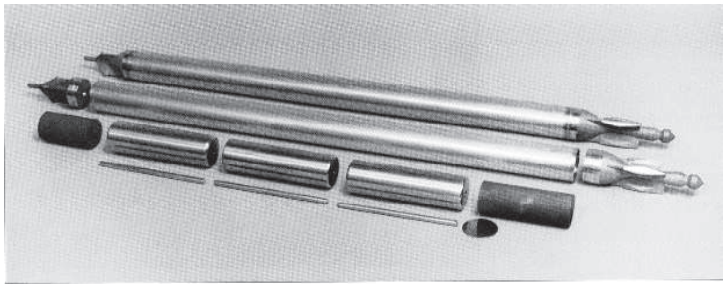


Fig. 9.32. Fuel pellets along with fuel pins used in TRIGA reactor (Printed with permission from [17]).

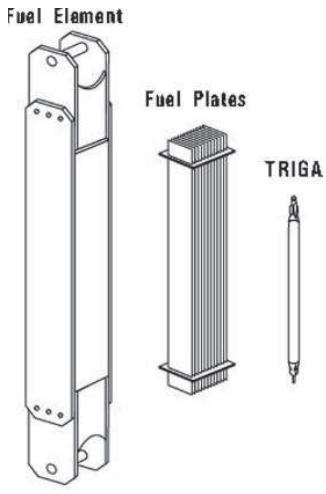


Fig. 9.33. Typical research reactor fuel assembly (Adapted from [18]).

As shown in Fig. 9.34, the assembly of the TRIGA fuel assembly contains a number of other components.

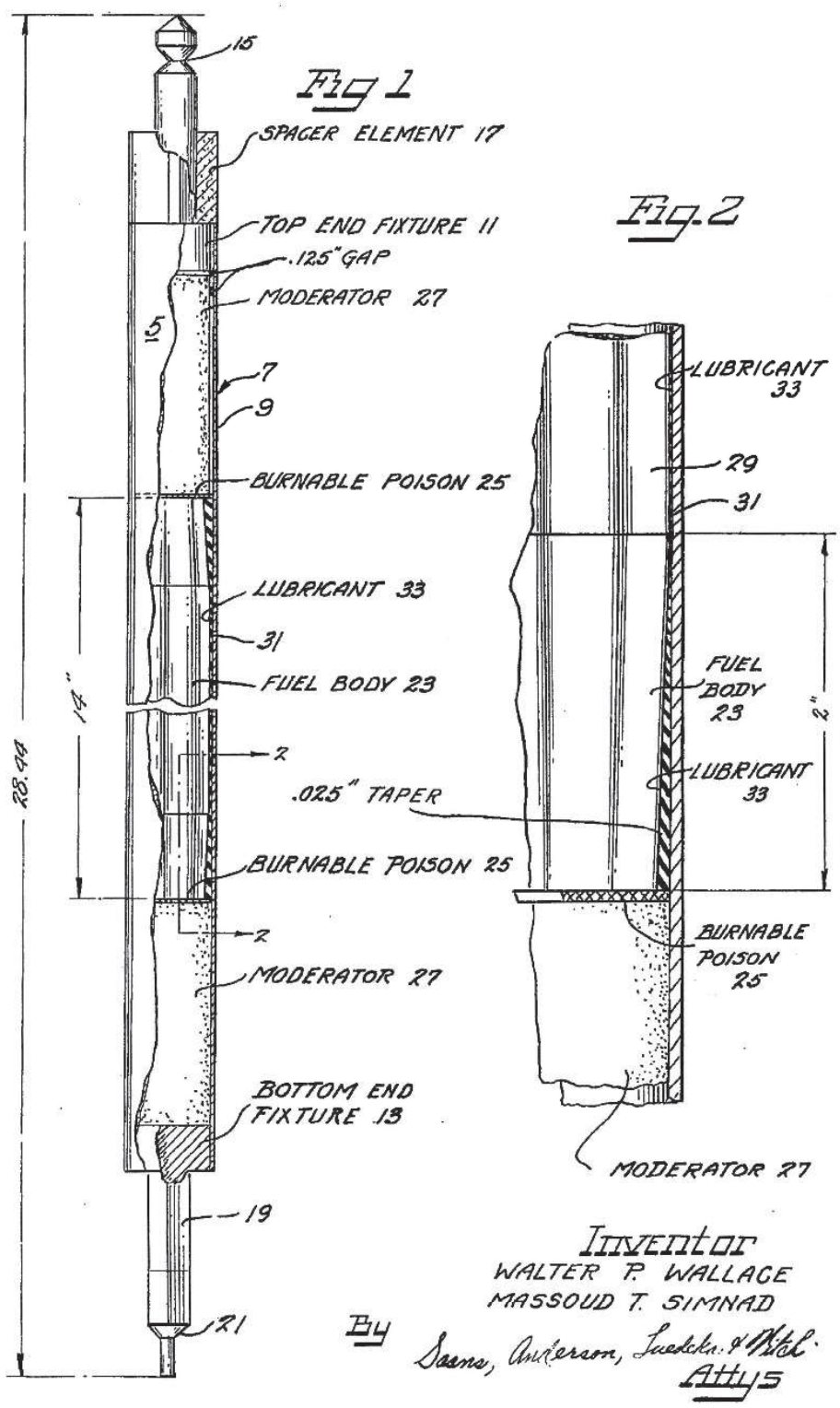


Fig. 9.34. The details of a TRIGA fuel pin design (Adapted from [19]).

9.7.3.3 Light Water Reactor Fuel Assemblies

The fuel assembly for light water reactors starts with the loading of pellets into fuel pins. Several materials have been utilized for these pins, including various alloys of stainless steel and zircoloy. However, zircoloy has become the material of choice due to its high heat transfer capabilities, low absorption cross section and chemical stability when exposed to the coolant and fuel. One end of each pin is welded, and fuel pellets are stacked and a spring holds them in place. This is shown in Fig. 9.35. The pin is evacuated, pressurized with helium and the open end is sealed and welded. PWR fuel pins are pressurized to approximately 2,000 psi and BWR fuel pins are pressurized to approximately half of that. This is due to different operating pressures within the pressure vessels of these reactors.

The loaded fuel pins are arranged into a proper geometry for the respective application. PWR and BWR cores utilize a variety of square fuel pin arrays. A distinct difference lies in the control rod accommodations of the two core designs. PWR assemblies have guide tubes, which allow control rods to slide in and out. BWR control rods are a crucifix shape and are placed in the gaps between assemblies. BWR assemblies are also surrounded by a zircoloy shroud, to prevent coolant mixing between assemblies.

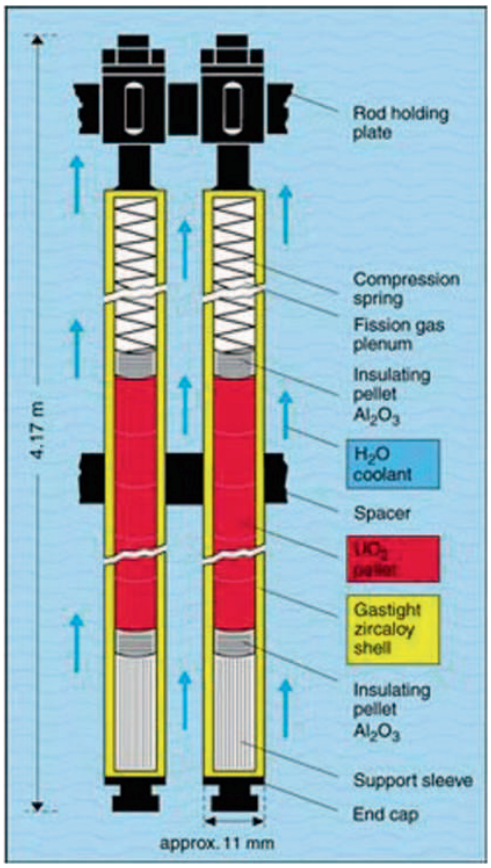


Fig. 9.35. Basic steps in fuel rod and fuel assembly construction (Printed with permission from [20]).

9.7.3.4 CANDU Reactor Fuel Assemblies

CANDU reactors utilize the pressurized water design. The fuel pin design is essentially the same, but the assembly arrangement is different from PWR. A distinct advantage of CANDU reactor design is that it does not require enriched fuel. This drastically reduces the front-end costs associated with fuel production. Natural UO_2 pellets are again stacked into zircoloy pins, with a spring holding them in place. A graphite layer is added to CANDU fuel pins between the pellets and zircoloy pin. The pins are pressurized with helium, welded and leak checked.

Fig. 9.36 shows a CANDU fuel assembly. As can be seen from the figure, it has significantly different arrangement than PWR or BWR fuel assemblies. The circular assemblies are 0.5 m long and are arranged horizontally within the core.

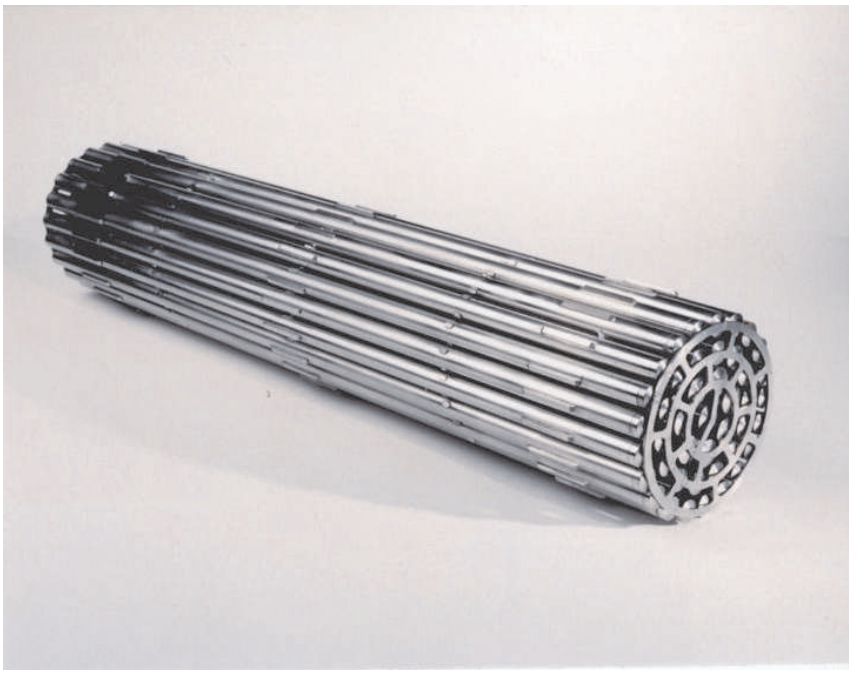


Fig. 9.36. A CANDU fuel assembly (Courtesy of Atomic Energy of Canada Limited, Mississauga, Ontario, Canada).

9.7.3.5 Mixed Oxide (MOX) Fuel Assemblies

The objective of reprocessing is to use separated plutonium as the fuel in the PWR or BWR reactors. Pu-239 is generated in light water reactors, which is a fissile isotope and responsible for approximately 30% of power production in PWR and BWR cores. Spent fuel from these reactors still contains a substantial amount of fissile isotopes, including the Pu-239. During reprocessing of light water spent fuel, Pu-239 is collected and mixed with natural or enriched uranium to produce mixed oxide fuel (MOX). This fuel may contain up to 8% PuO_2 and run in existing PWR cores with up to 30–40% of the MOX fuel. No significant core modifications is

necessary [16]. The use of MOX fuel greatly enhances the utilization of reactor fuel beyond the once-through cycle and has been utilized by Japan and France for many years. The United States is currently in the process of licensing a facility for MOX production as a partial solution to the vast supply of spent reactor fuel awaiting disposal.

9.7.3.6 Metal Fuel

Metallic fuel has not been heavily utilized in the past, but may be ready to become more prominent with the development of fast breeder reactors and new pyro-metallurgical procedures, which combine reprocessing and fuel fabrication [21]. Advantages of metal fuel include relative ease of fabrication and its extremely high heat transfer capabilities, as compared with oxide fuels. The main disadvantage of metallic fuels is the low melting temperature, which has been a chief concern of regulatory agencies. However, the very high heat transfer capability of these fuels overcomes the low melting points and ensures fuel stability.

9.7.3.7 Thorium Fuel

Thorium is far more plentiful in nature than uranium. This is especially true in India, where the ratio of thorium to uranium reserves is approximately 6 to 1. India's growth in energy consumption is dramatic, and they have invested substantial resources into finding ways to utilize their thorium reserves. Table 9.8 provides the estimated thorium reserves worldwide at an extraction cost of $80/kg.

Table 9.8. Thorium reserves of various countries [22].

Country	Tonnes	% of world
Australia	452,000	18
USA	400,000	16
Turkey	344,000	14
India	319,000	13
Venezuela	300,000	12
Brazil	221,000	9
Norway	132,000	5
Egypt	100,000	4
Russia	75,000	3
Greenland	54,000	2
Canada	44,000	2
South Africa	18,000	1
Other countries	33,000	1
World total	2,492,000	

RAR + Inferred to USD 80/kg Th

"Source" Data for Australia compiled by [22]

Thorium (Th-232) can be fabricated as an oxide fuel for pellets, micro-particle (for TRISO fuel) or metal fuel. As oxide fuel, thorium can be placed around enriched uranium providing a blanket for the fissile material. The concept is essentially the same as that of a light water breeder reactor. Th-232 is a fertile material and absorbs neutrons to become U-233, which fissions and keeps the chain reaction going. However, Pu-233 is also generated during this process. Also, there are substantial concerns about reprocessing, as U-233 has some intense gamma-emitting daughters with short half-lives.

9.7.3.8 Fuel Cladding

Fuel cladding serves as a protection device for the fuel, provides a barrier for fission products escaping the fuel and acts as a heat transfer surface from the fuel to the coolant. Ideal cladding material must have a small cross section for neutron absorption and interact well with the fuel and coolant. Materials that have been used as cladding include stainless steel and zircoloy, the latter being most heavily utilized for most applications.

Future reactors will need advanced cladding materials, as many of these reactors will subject the fuel to much higher operating temperatures. TRISO fuel, which will be discussed later in this chapter, is well suited for higher operating temperatures. The pyrolytic-carbon layers protect the fuel from the high temperatures and the silicon carbide layer provides the extra fission product barrier. Unlike prior fuel designs, the TRISO fuel cladding is incorporated into the fuel particle itself. The disadvantage of this unique design is the reprocessing challenges this represents, making once-through of TRISO a distinct possibility. Other future reactors concepts do not have cladding at all. Molten fuel concepts have no specific fuel structure and hence, no cladding.

9.8 Uranium Downblending

Uranium used in nuclear weapons is enriched to approximately 93% of U-235, while uranium used in commercial nuclear power plants typically is enriched to 3–5% U-235. Uranium enriched to more than 20% U-235 is called Highly Enriched Uranium (HEU) and can only be used in nuclear weapons and in research reactors. Surplus HEU can, however, be downblended with Low Enriched Uranium (LEU) to make it suitable for use in commercial nuclear fuel.

The downblending only involves uranium. In contrast, plutonium is used for the production of mixed oxide fuel (MOX).

In 1993, the U.S. and Russia signed the US-Russia HEU Agreement, under which Russia was to supply the downblended uranium derived from 500 t of HEU to the USA over a period of about 20 years. While the deliveries under this agreement are still ongoing, the U.S. now have begun downblending some of their own surplus HEU.

9.8.1 Blending Process

In a first step, the HEU and the blendstock have to be converted to the chemical form required for the selected blending process, if not already in the appropriate form. For the downblending process, there exist the following methods:

Mixing of liquids

Uranium in the form of uranyl nitrate hexahydrate (UNH), $UO_2(NO_3)_2 \cdot 6H_2O$, or molten uranium metal are mixed together to form the final mixture.

Mixing of gases

Uranium is converted to uranium hexafluoride (UF_6) and mixed together in the gaseous phase.

The existing commercial downblending facilities in the U.S. (BWXT in Lynchburg, Virginia, and NFS in Erwin, Tennessee) are using the UNH process, while the Russian facilities (in Novouralsk, Seversk, and Zelenogorsk) are using the UF_6 process. Historically, downblending has also been performed at the following DOE nuclear weapons facilities in the U.S.: the Y-12 Plant in Oak Ridge, Tennessee (UNH and molten metal processes), and the Savannah River Site (SRS) in Aiken, South Carolina (UNH process).

After the blending, the material has to be converted to UO_2, before it can be used in the production of commercial nuclear fuel. The blending process has been further described by Arbital and Snider [23].

9.9 Power Production/Burn Up

A nuclear reactor core is composed of a several hundred “assemblies”, arranged in a regular array of cells, each cell being formed by a fuel or control rod surrounded, in most designs, by a moderator and coolant, which is water in most reactors.

As the fuel is consumed (under go fission), the old fuel rods must be replaced periodically with fresh ones (this period is called a cycle). The fuel is generally replaced every 18–24 months. Only a part of the assemblies (typically one third) are removed since the fuel depletion is not spatially uniform, and the rest of the fuels are rearranged in the core. The new assemblies are not placed exactly at the same location of the removed ones. Even bundles of the same age may have different burn-up levels, which depends on their previous positions in the core. Thus the available bundles must be arranged in such a way that the yield is maximized, while safety limitations and operational constraints are satisfied. Consequently reactor operators are faced with the so-called optimal fuel reloading problem, which consists in optimizing the rearrangement of all the assemblies, the old and fresh ones, while still maximizing the reactivity of the reactor core so as to maximize fuel burn-up and minimize fuel-cycle costs.

9.10 Types of Nuclear Reactors

9.10.1 Research Reactors

Most research and training reactors are pool type and are not designed for power production. In pool type reactor, the core containing fuel rods and control rods is immersed in an open pool of water. The water under normal pressure acts as neutron moderator, cooling agent and radiation shield. In this type of reactors, operators may work above the reactor safely since water can provide adequate shielding. Pool reactors are used as a source of neutrons and for training, and in rare instances for process heat but not for electricity generation.

Research reactors typically range from 10 kW to 10 MW. Research reactors are simpler than power reactors and operate at lower temperatures. These reactors typically need far less fuel, but more highly enriched uranium, typically up to 20% U-235; although some use 93% U-235. The core is cooled typically by natural or forced convection with water, and a moderator is required to slow down the neutrons and enhance fission. To reduce neutron loss from the core generally a reflector is used. The most common type of research reactor is called TRIGA. A typical TRIGA reactor is shown in Fig. 9.37.

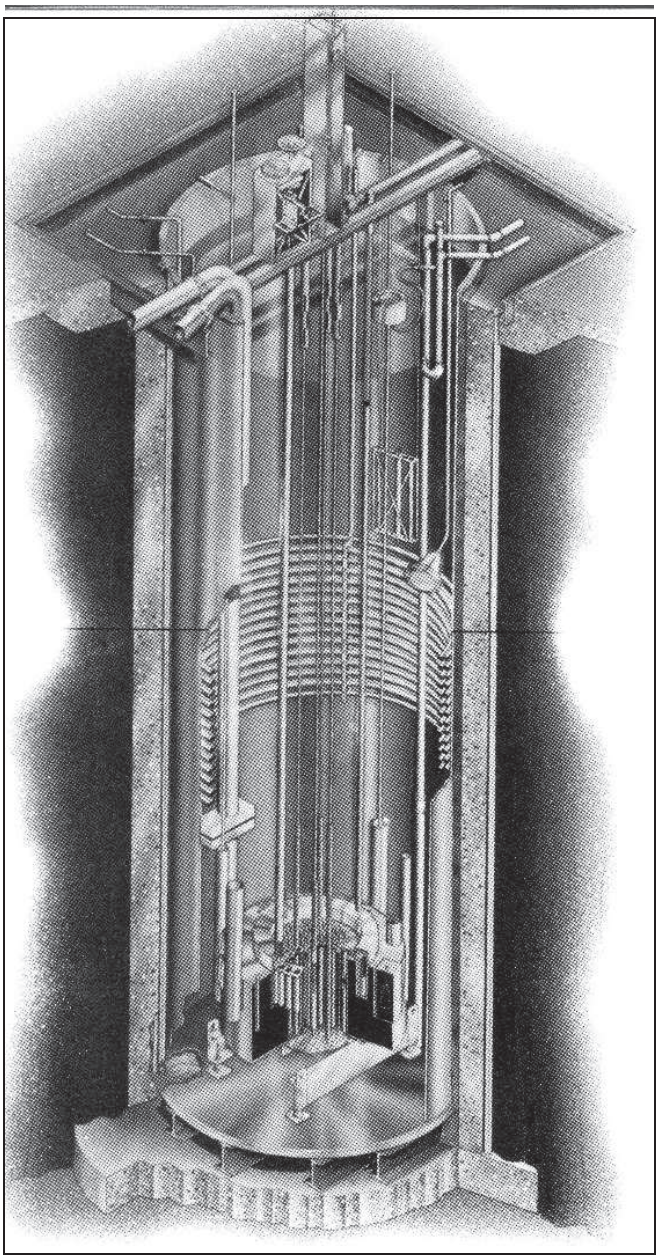


Fig. 9.37. The cutaway view of a 10 kW TRIGA Mark I reactor at San Diego, CA, USA (Printed with permission from [17]).

9.10.2 Commercial Reactors for Electricity Generation

As can be seen from Fig. 9.38, nuclear reactors for electricity generation are continuously undergoing design changes. Several types of the nuclear reactors are in use either commercially or for research purpose. The power reactor deployment timeline is shown in this figure, from past reactors to anticipated future reactors, although it has shifted to the right multiple times.

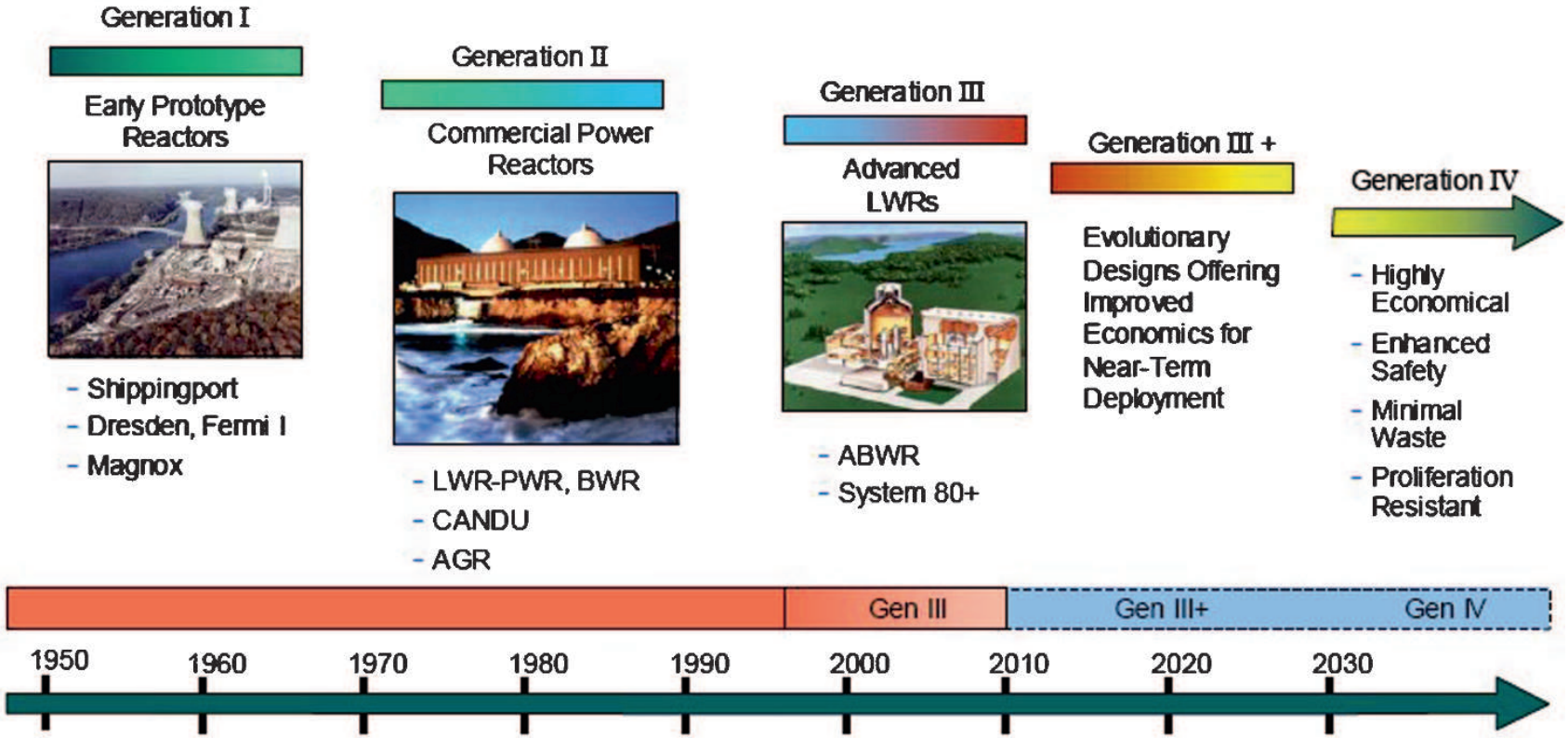


Fig. 9.38. Timeline of deployment of various generations of nuclear power reactor. Adapted from U.S. Department of Energy (2002) A technology roadmap for Generation IV nuclear energy systems. U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, Decemfber 2002, GIF-002-00. http://www.ne.doe.gov/GenIV/documents/gen_iv_roadmap.pdf [24].

9.11 Generation II Light Water Reactors

Current nuclear reactors used for power production are Generation II reactors, and 439 reactors are operating worldwide at the present time, with 104 in the U.S. PWRs and BWRs belong to Generation II reactor designs. Several advances have been made over the years in order to increase the efficiency and improve safety. However, the basic design of the reactors has not changed substantially. Although both of these reactor designs utilize water as a moderator and coolant, there are distinct differences between these two reactors.

The light water design necessitates the use of enriched uranium fuel, typically 3–5% U-235. After a typical three cycles of use, the fuel reaches its maximum burn-up and is no longer suitable as fuel for these reactors. The fuel is allowed to decay in spent fuel pools, which provide radiation shielding and decay heat removal capabilities. Once sufficient decay time has been reached, the fuel can be placed in dry storage, sent to a final repository site, or re-processed to retrieve useful fuel isotopes and minimize the high level waste requiring ultimate disposal. Descriptions of Generation II reactors can be found in a number of nuclear engineering text books [25–31] and NRC reactor concept manuals [32–34].

9.11.1 Pressurized Water Reactor (PWR)

Pressurized water reactors (PWR) account for approximately 60% of U.S. operating reactors. The primary coolant system consists of large pumps, which circulate the water through the reactor, where the energy is transferred from the fuel elements to the coolant, then on to a secondary coolant circuit to produce steam. The cooler water is then circulated back to the reactor and the process repeats. Another essential component within the primary coolant system is a high pressure pump, which maintains primary coolant pressure at approximately 2,000 psi. This prevents the coolant from flashing to steam at its normal operating temperature of 600°F .

The steam is used to run both high and low pressure turbines to generate the electricity. The exhausted steam is directed to a condenser, where it is cooled through interactions with a third coolant (usually river or sea water). The condensed secondary water is then pumped back to the steam generator and the cycle repeats.

A schematic flow diagram of a PWR nuclear power plant is shown in Fig. 9.39. The PWR utilizes UO_2 pellets within zircoloy pins. These pins are arranged into a square assembly, typically 14 × 14 to 18 × 18 arrays as shown in Fig. 9.40. The assembled core is shown in Fig. 9.41. There are approximately 120–190 of these assemblies in each reactor core. A PWR vessel with the fuel assembly and other accessories is shown in Fig. 9.42.

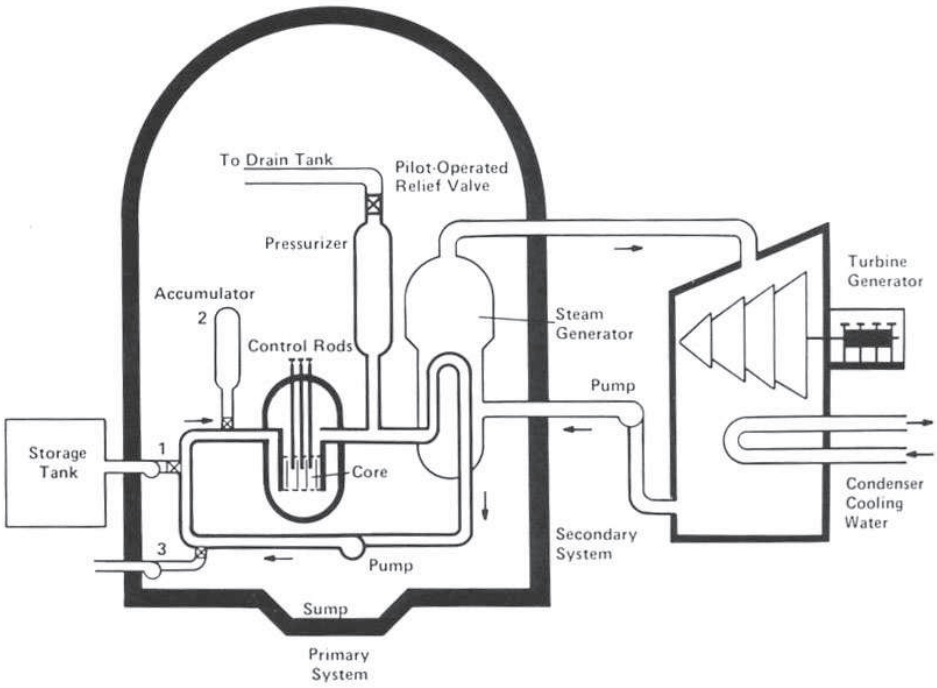


Fig. 9.39. Schematic of a PWR system (Printed with permissionfrom [35]).

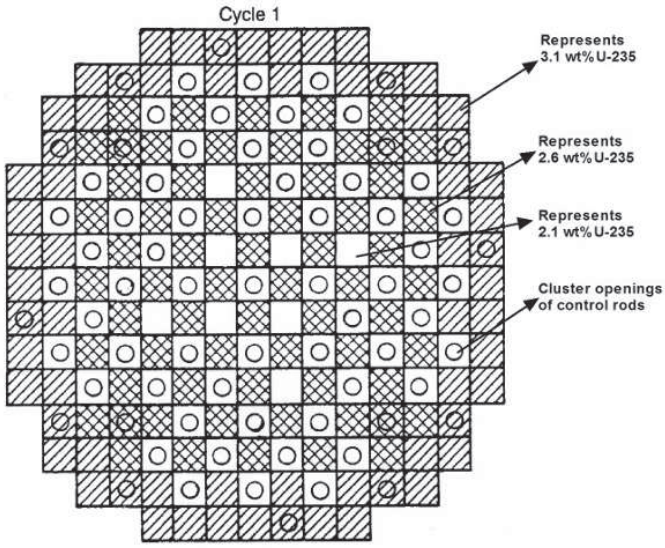


Fig. 9.40. Initial fuel loading of PWR and their arrangements in the fuel assembly (Courtesy of Westinghouse Electric Corp. Printed with permission from [25]).

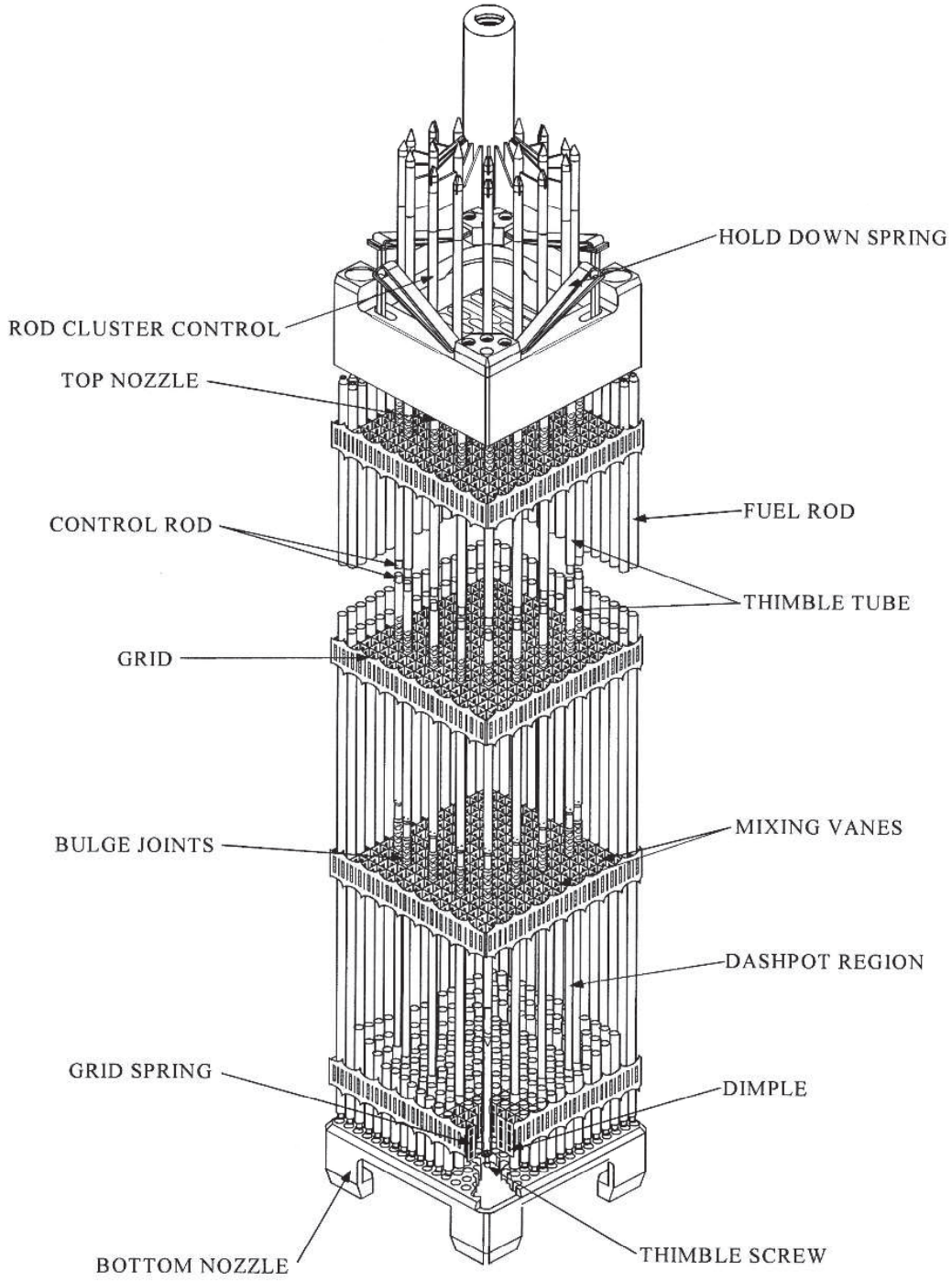


Fig. 9.41. A PWR fuel assembly for 17 × 17 array of rods (Courtesy of [33]).

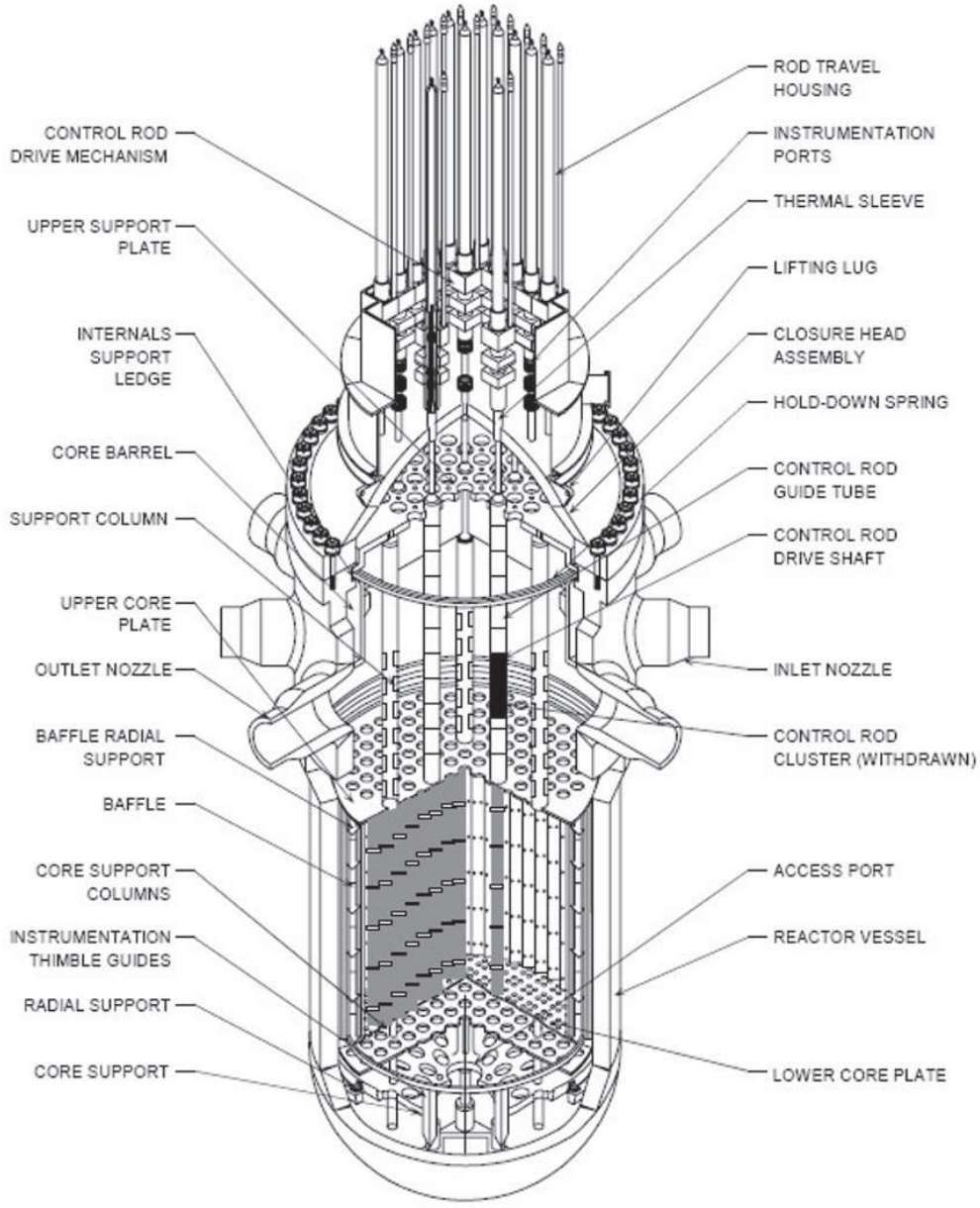


Fig. 9.42. A PWR pressure vessel with fuel assembly and various reactor components (Courtesy of [34]).

9.11.2 Boiling Water Reactor (BWR)

Boiling water reactors (BWR) encompass approximately 40% of operating commercial reactors in the U.S. A schematic diagram of a BWR nuclear power plant is shown in Fig. 9.43. Although water is still utilized as both coolant and moderator, primary coolant is circulated into the pressure vessel and directed to the bottom of the reactor core, similar to PWR designs. However, BWR coolant is at a much

lower pressure than their PWR counterparts and mass boiling is allowed in the fuel region. The steam is collected at the top of the pressure vessel, where the quality is increased prior to direct feed into the main turbine for electricity generation. Similarly to a PWR, the exhausted steam is fed through a condenser where secondary water (lake, river, sea water, etc.) cools the primary coolant and condenses it back to a liquid. This liquid is then returned to the pressure vessel and the process repeats.

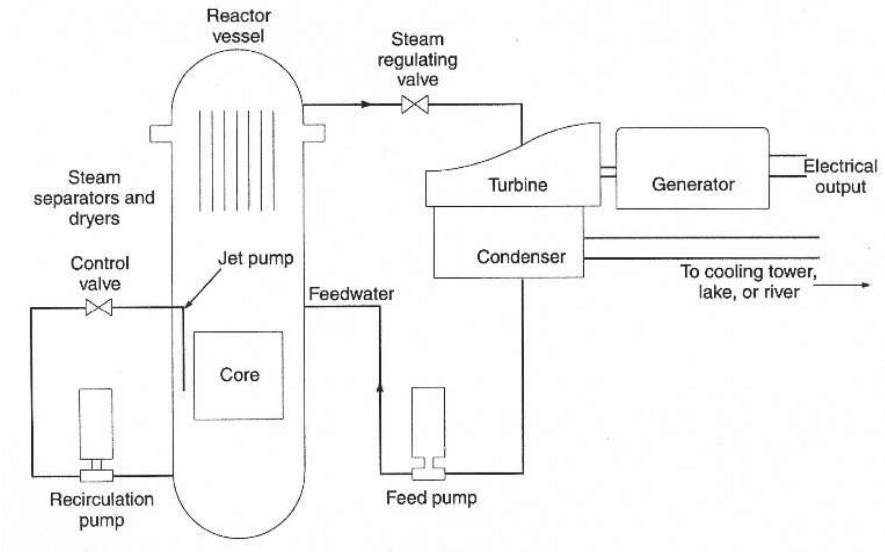


Fig. 9.43. Flow diagram of a BWR direct cycle system (Courtesy of [36]).

BWR fuel consists of oxide pellets in zircoloy pins, arranged in typically 9 × 9 or 10 × 10 arrays, as shown in Fig. 9.44. These assemblies are much smaller and lighter than PWR assemblies, but a far greater number (up to 800) is required to make up the larger BWR core.

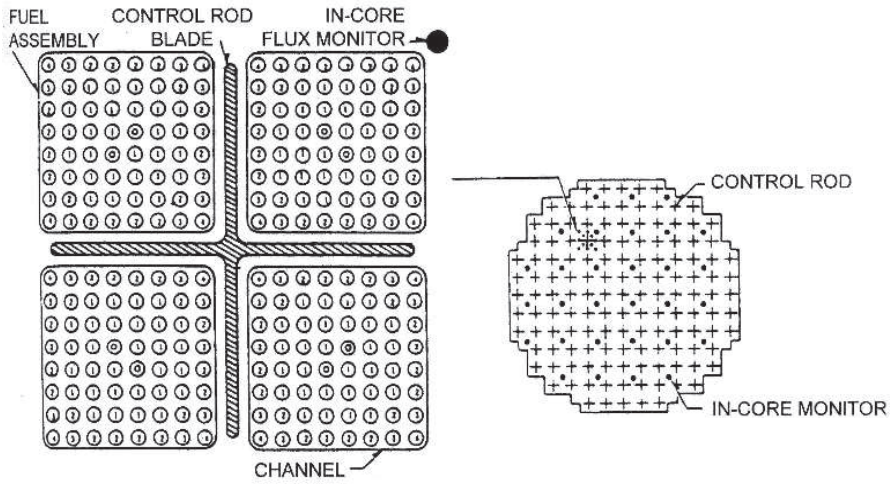


Fig. 9.44. Fuel rod arrangement in a BWR core (Printed with permission from [25]).

The BWR fuel assembly and the cutaway view of the pressure vessel are shown in Figs. 9.45 and 9.46, respectively.

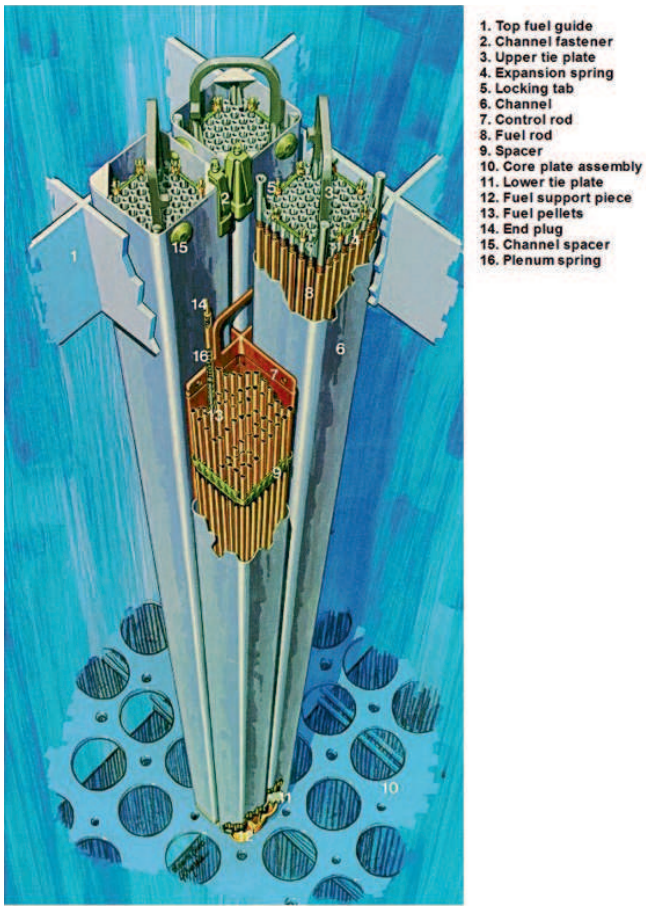


Fig. 9.45. A BWR fuel assembly (Adapted from [37]).

A set of pumps re-circulates water within the pressure vessel to help pre-heat the incoming coolant water. Start-ups and shut-downs are primarily accomplished by manipulating the control rods into and out of the reactor core, which are inserted through the bottom of the core. In contrast, in a PWR control rods are inserted from the top of the core.

The transfer of heat from one coolant medium to another is a source of efficiency loss in any reactor. Since the BWR design incorporates core cooling and steam generation into one loop, this provides an advantage over the PWR design. BWR pressure vessels are not subjected to higher pressures and do not require the thicker vessel walls, drastically reducing the cost and difficulty of forging. However, these economical advantages are countered by the radiological issues associated with BWR designs. Since the steam is in direct contact with the turbine, it becomes contaminated and requires massive and costly shielding during operation. Accident analysis becomes more challenging as well, since a portion of the primary loop must exit containment to drive the turbine.

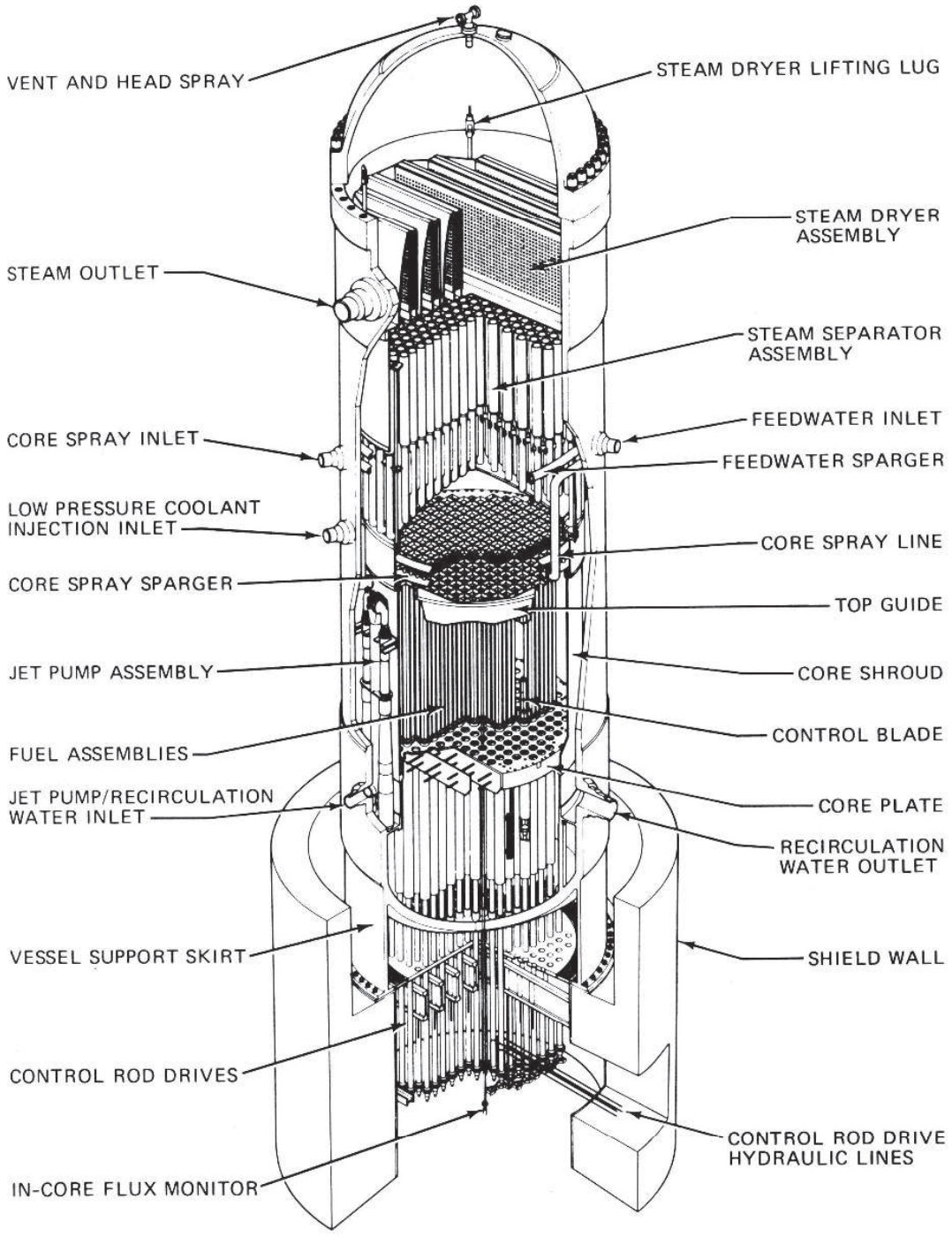


Fig. 9.46. Cutaway view of the model BWR pressure vessel (Courtesy of GE Nuclear Energy).

A major difference between the PWR and the BWR design is that in the PWR the primary and secondary coolant systems have boundaries between one another, which are not crossed under normal conditions. This allows containment of the PWR primary system within one building and prevents radioactive contamination

of the steam turbine. Isolation of the primary coolant within a single containment structure offers an additional benefit with regards to accident analysis, such as a loss of coolant accident.

To reiterate, the PWR design is not without its disadvantages. The pressure vessel of the PWR is thicker and costlier than that of the BWR due to the high pressure of the primary coolant. Additionally, the use of three heat transfer mediums in the PWR lowers the overall plant efficiency.

9.11.3 CANDU Reactors

Heavy water (D_2O) is an extremely good moderator; it slows down neutrons without absorbing them. This efficient use of neutron inventory allows the use of natural uranium to continue the chain reaction, where it otherwise would not be possible with the use of other moderating materials.

As mentioned previously, Canada has a vast supply of uranium deposits and is one of the world's leading suppliers. Canada has never pursued a nuclear weapons program and thus did not feel the need to build a fuel enrichment facility. However, during WWII, Canada assisted the U.S. by developing the first heavy-water moderated reactor in Montreal, which could be used to produce plutonium. After the war ended, Canada continued to develop the heavy water moderated reactor, and it has now evolved into a pressurized heavy water moderated reactor, called the CANada Deuterium Uranium (CANDU) reactor. The reactor system is shown in Fig. 9.47.

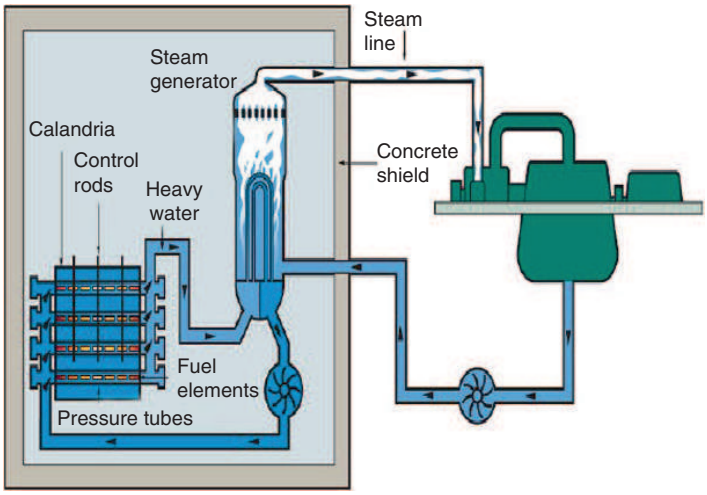


Fig. 9.47. Layout of a CANDU reactor system (Printed with permissionfrom [38]).

The CANDU design is similar to the PWR design in many respects. It is also pressurized to prevent boiling of the coolant and steam generators convert secondary coolant into steam for the turbine. In a CANDU reactor the fuel assembly is

horizontally mounted, allowing for fuel bundles to be replaced during operation without any need to shut down the reactor completely. These bundles consist of 37 zircoloy fuel rods, containing uranium oxide pellets. Twelve bundles lie end to end in each channel, and are simply forced out of one end by new bundles being inserted into the opposite end. The inside view of a CANDU is shown in Fig. 9.48.

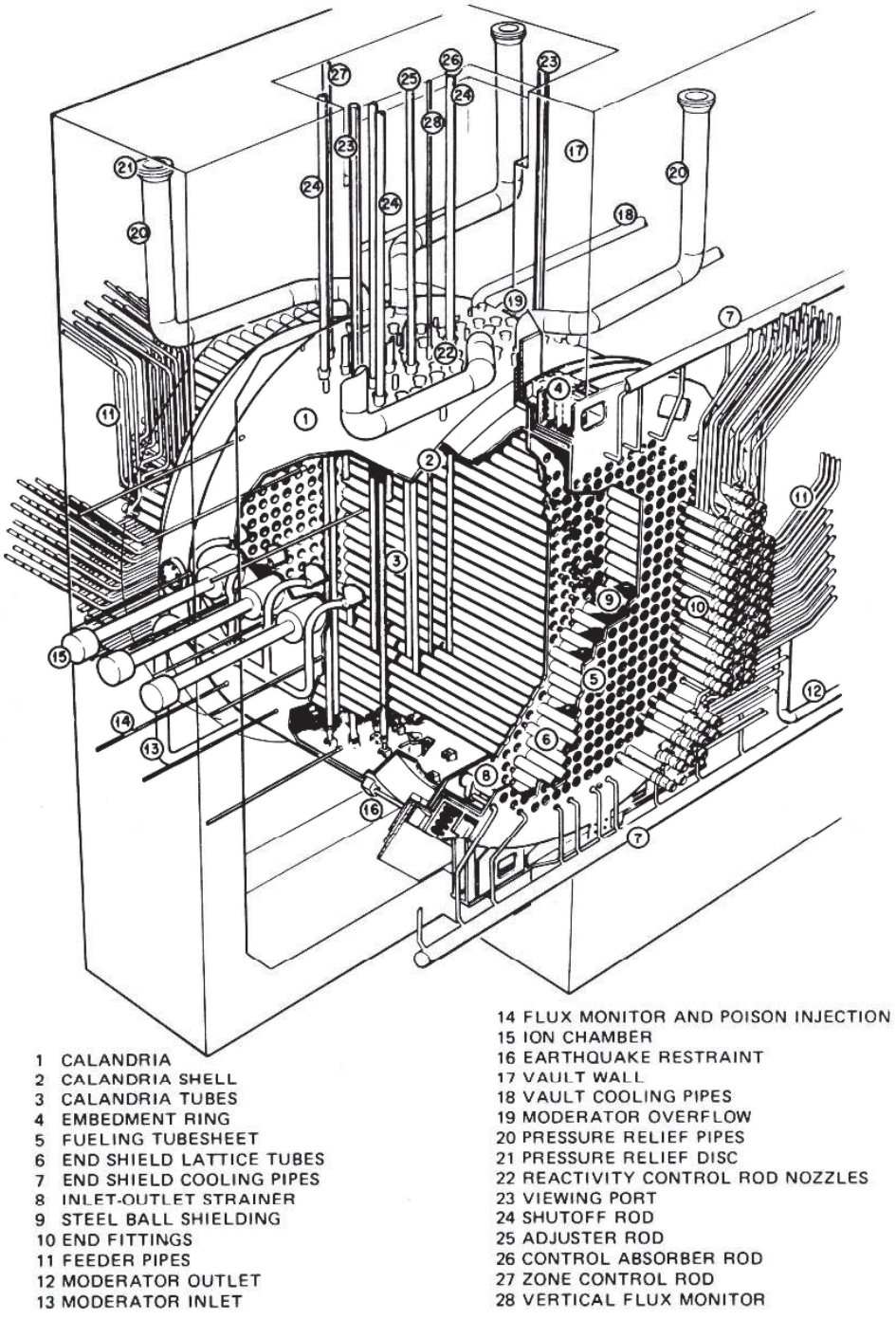


Fig. 9.48. The cutaway view of a CANDU reactor assembly (Adapted from [39]).

CANDU technology remains popular within Canada and has been partially adopted in India due to its ability to utilize thorium fuels. Flexible fuel capabilities are one of the chief advantages of this design. Additionally, the ability to re-fuel during operation pushes capacity factors much higher than is usually seen in PWR and BWR designs. Utilizing natural uranium eliminates enrichment costs, drastically reducing the price of fuel. The biggest drawback of the CANDU design is the high cost of heavy water. The CANDU system is shown in Fig. 9.49.

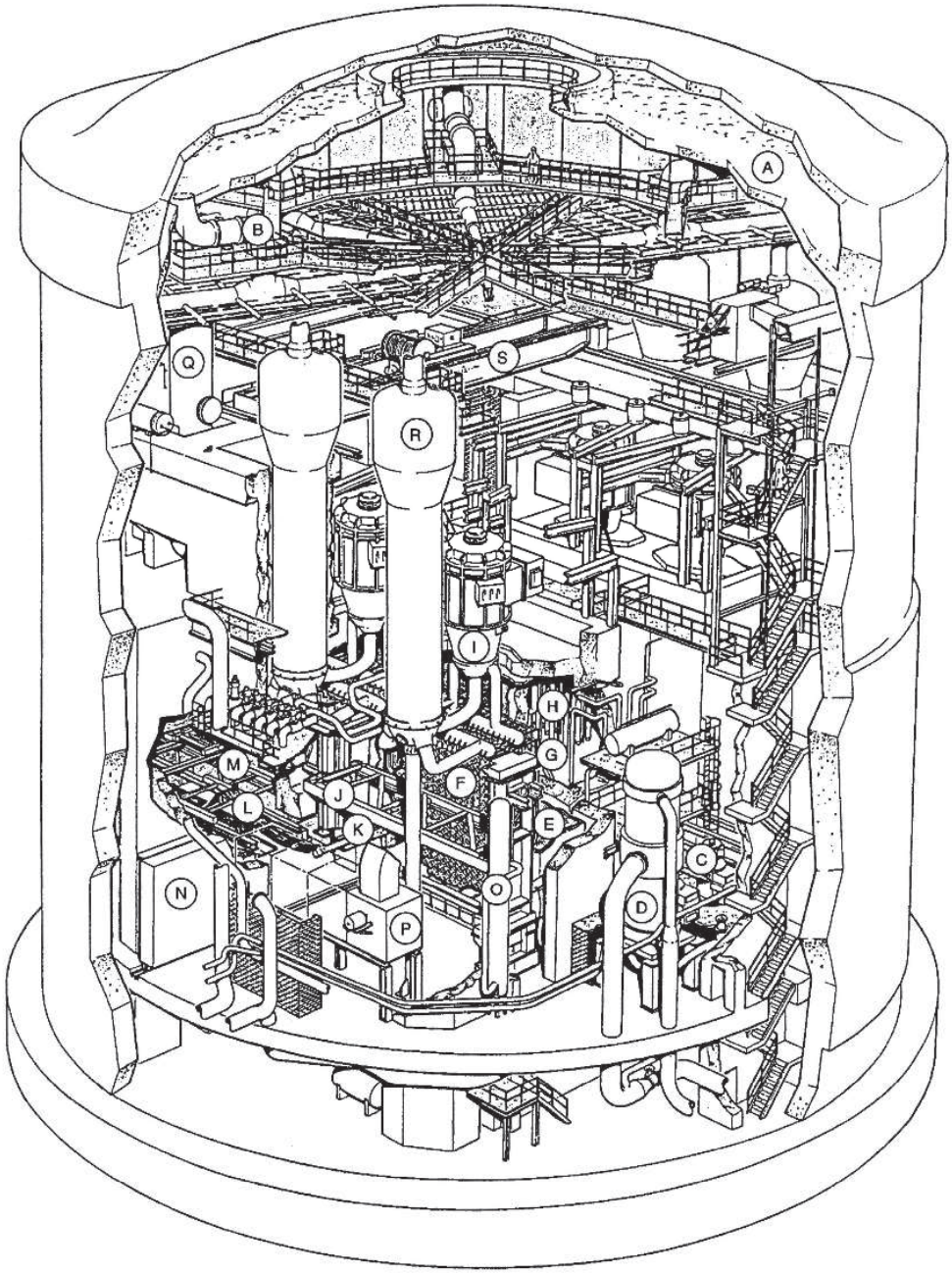


Fig. 9.49. A CANDU reactor (Printed with permission from [35]).

9.11.4 RBMK Reactors

The Russian water-cooled, graphite moderated reactors were developed in the former Soviet Union and 27 have been built and operated (Fig. 9.50). RBMK cores have some similarities to Advanced Gas Reactor (AGR) and BWR designs. The overall layout is similar to an AGR, but each fuel assembly is housed in a pressurized tube, containing light water as a coolant. Despite the pressure, the coolant is allowed to boil and directly feeds the turbine. The fuel is made up of 18 zircoloy rods, containing enriched uranium oxide, to make up a 3.5 m long bundle [16]. The fuel bundles are stacked in the core.

Negative void temperature coefficients are inherent in the design of most nuclear reactors and add a large degree of safety. A detailed analysis of the reactor safety system has been given by IAEA [40]. As temperature rises within the core, the reactor essentially shuts itself down as the void co-efficient takes on a greater influence on core reactivity. This is not the case with the RBMK design, as graphite has a positive void co-efficient and any increase in temperature adds positive reactivity to the core. This design instability became infamous in 1986, when Chernobyl reactor number 4 catastrophically failed, during planned testing, in which operators had disregarded critical operational procedures and safety systems were over-ridden. Since then, the RBMK design has been modified to prevent a similar occurrence. The most significant change has been utilization of higher enriched fuels to help absorb thermal neutrons and create a negative void feedback.

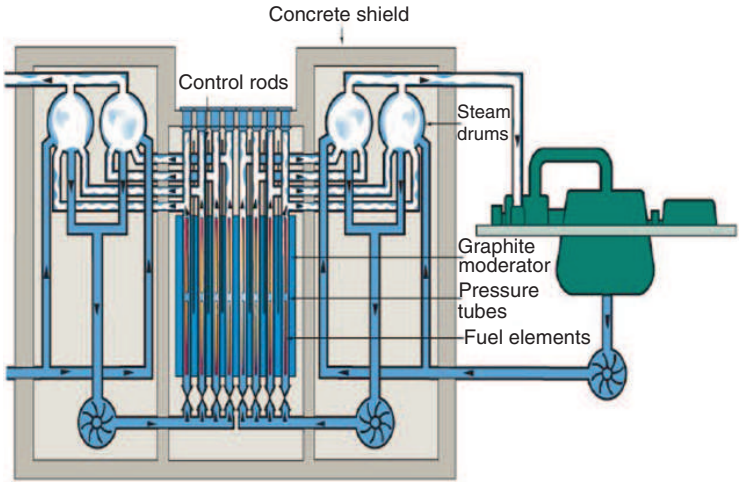


Fig. 9.50. A RBMK reactor system (Printed with permission from [38]).

9.12 Generation III & III+ Reactors

Generation III and III+ reactors are the next generation of reactors, some of which have already been built and are operating worldwide. Industry experience has prompted drastic alterations in the design, construction, licensing and operation of nuclear power plants. In the past, reactors were built to suit individual companies and even individual sites. This will no longer be the case, as each of the next generation designs will be built more modularly, requiring the customers to prepare their sites to suit the reactor design. The shift in design philosophy is intended to reduce construction time and costs, stream-line licensing procedures for regulators and standardize operational procedures.

Many of these new designs have incorporated passive or inherent safety features, some requiring no active controls or operator action to mitigate design accidents. This important characteristic will allow the new reactors to operate more efficiently and produce more power than their predecessors. Additionally, these reactors have been designed for longer operational lifetimes (60 years) and higher fuel burnup. The Generation III and III+ reactors are listed in Table 9.9.

Table 9.9. Generation III and III+ reactors.

Reactor Type	Abbreviation Used	Design life	Vendor
Advanced Boiling Water Reactor	ABWR		General Electric, USA
Advanced Passive-1000	AP1000	60	Westinghouse Electric Co., USA
US Evolutionary Power Reactor	USEPR	60	Areva, France
US Advanced Pressurized Water Reactor	USAPWR	60	Mitsubishi Heavy Industries, Japan
Economic Simplified Boiling Water Reactor	ESBWR		General Electric, USA
Advanced CANDUS Reactor	CANDUACR		Atomic Energy Canada Ltd., Canada
International Reactor Innovative and Secure	IRIS		International Team
System 80+	System 80+		Westinghouse Electric Co., USA

9.12.1 Advanced Light Water Reactors

Several different designs of advanced light water reactor are in the process of being deployed worldwide. A focus will be made on the most likely designs to be deployed and the regions of the world in which they will be located.

Utilities within the U.S. have several designs under consideration, although none have yet been built. NRG Energy and South Texas Project are set to begin construction on the first two new generation reactors in the U.S, and already submitted the combined license application to the NRC. These will be of the advanced boiling water reactor (ABWR) designed by General Electric. This design was selected mainly due to operational experience of the four similar units that have been operating in Japan since 1996, with four more under construction. This is a proven design, which will dramatically reduce the construction time and enhance operational confidence. The ABWR (Figs. 9.51 and 9.52) features internal circulating pumps for improved safety, fully digital control systems and demonstrated reduction in construction and operational costs [41].

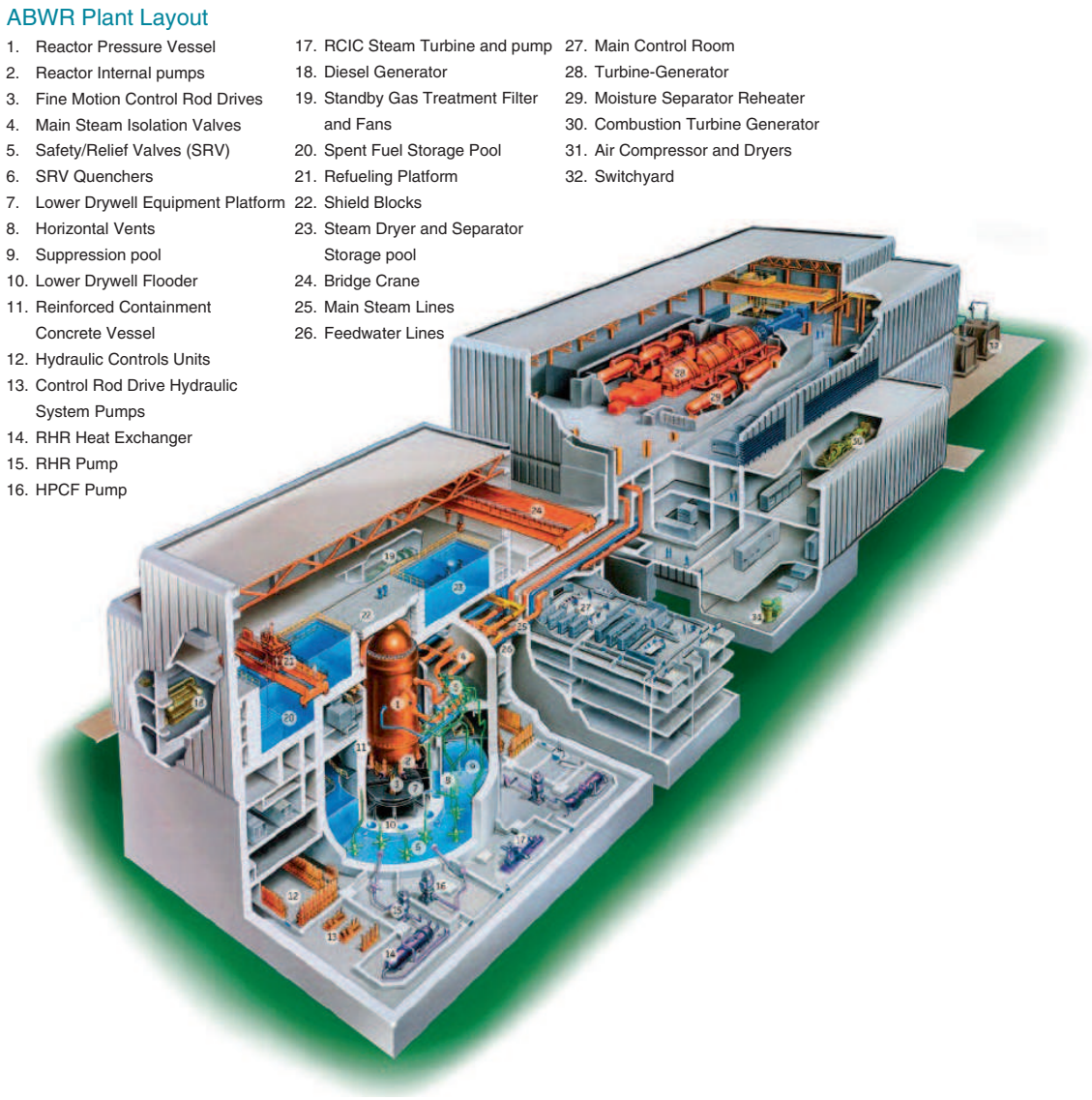


Fig. 9.51. The layout of an ABWR power plant (Courtesy of GE Hitachi Nuclear Energy. www.ge-energy.com/nuclear).

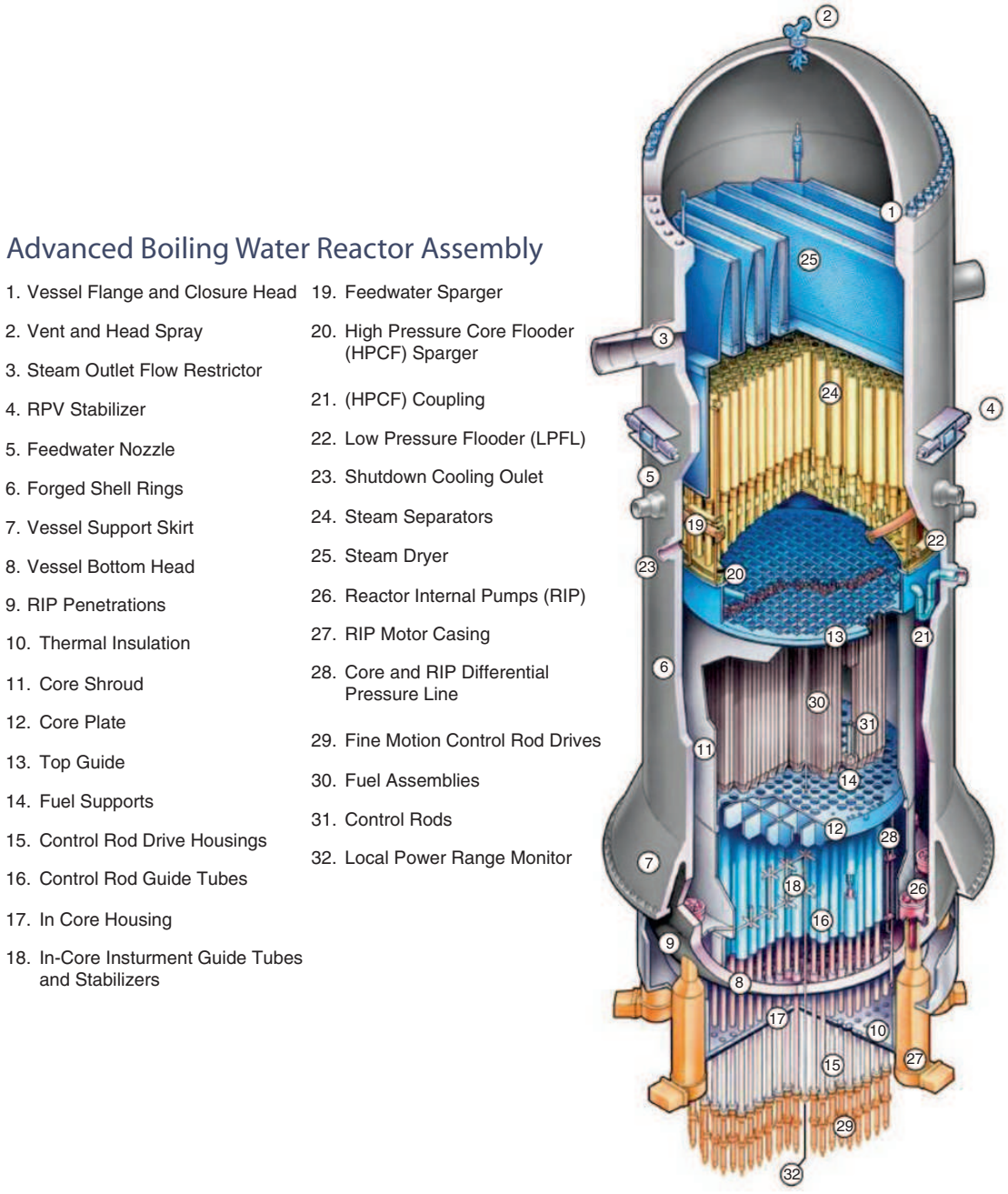


Fig. 9.52. The cutaway view of the ABWR reactor vessel (Courtesy of GE Hitachi Nuclear Energy. www.gepower.com/prod_serv/products/nuclear_energy/en/downloads/abwr_callouts.pdf).

9.12.2 Advanced Passive-1000 Reactor (AP-1000)

Westinghouse has received final design certification from the NRC for its Advanced Passive-1000 (AP-1000) reactor, the first Generation III+ design to do so. The AP-1000 design has improved upon proven PWR technology to include

enhanced safety features, as the most serious accidents require no operator action or AC power to shut down the plant or remove decay heat. This AP-1000 (Fig. 9.53) is an 1,100 MWe, scaled-up version of the AP-600 originally designed [42–47]. The steam generators (Fig. 9.54) are located within the containment building. The additional safety features shown in Fig. 9.55 makes it very attractive among other reactor designs.

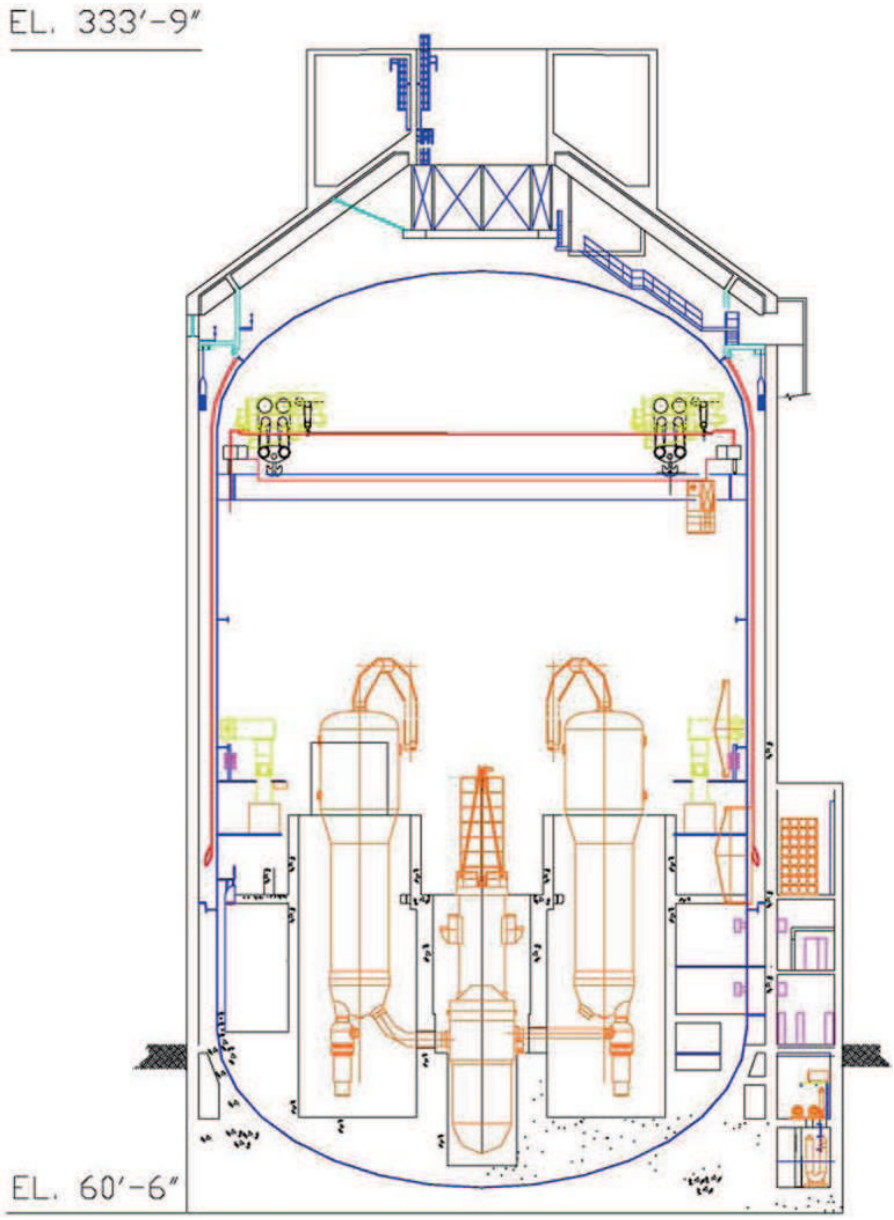


Fig. 9.53. AP 1000 reactor pressure vessel assembly (Printed with permission from [47]).

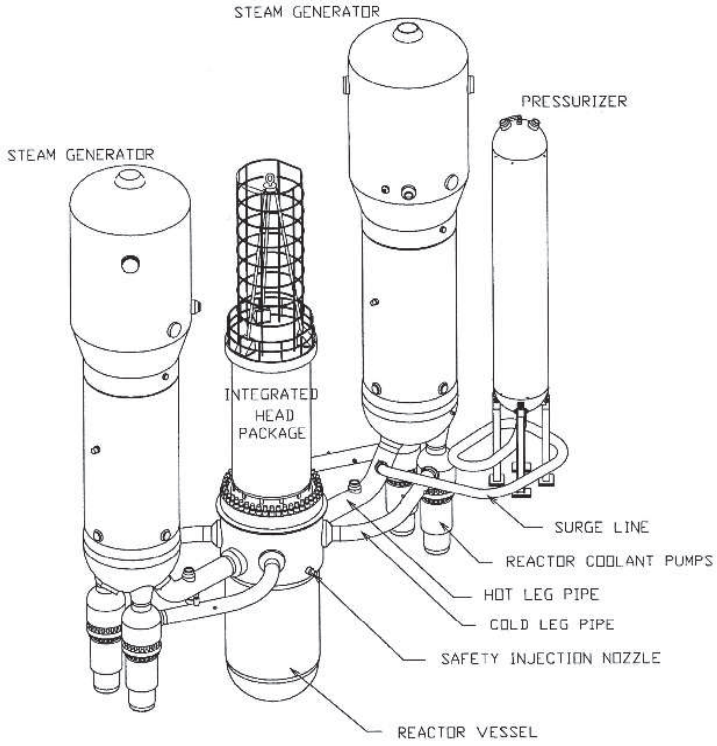


Fig. 9.54. Steam generator assembly of AP1000 (Printed with permission from [47]).

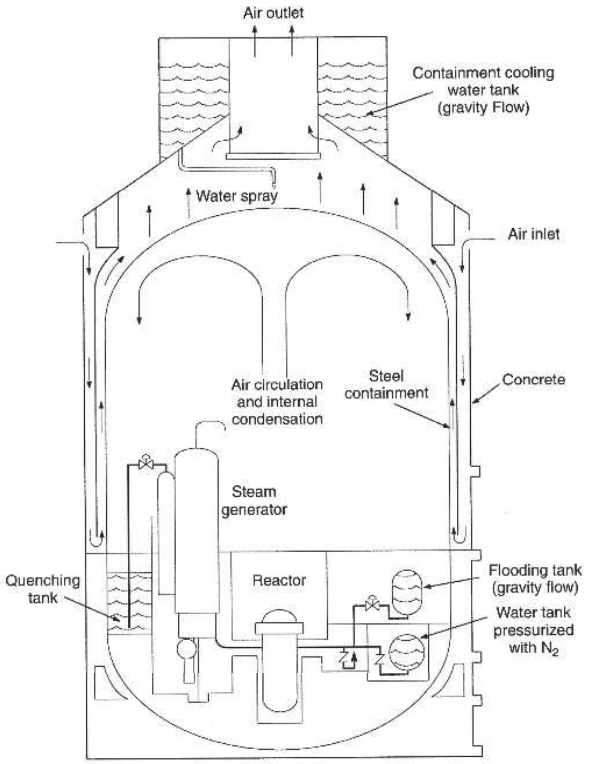


Fig. 9.55. Safety features of AP1000 (Printed with permission from [35]).

9.12.3 Advanced Pressurized Water Reactor (APWR)

Mitsubishi has designed a large 1,700 MWe Advanced PWR (APWR), which boasts one of the largest outputs of any commercial reactor designs to date and is shown in Fig. 9.56. This plant is a larger and more efficient version of the PWR designs currently in operation throughout the U.S. An application for standardized design certification was submitted to the NRC in December, 2007 and is awaiting approval. The fuel assembly of APWR is shown in Fig. 9.57, and the pressure vessel is shown in Fig. 9.58.

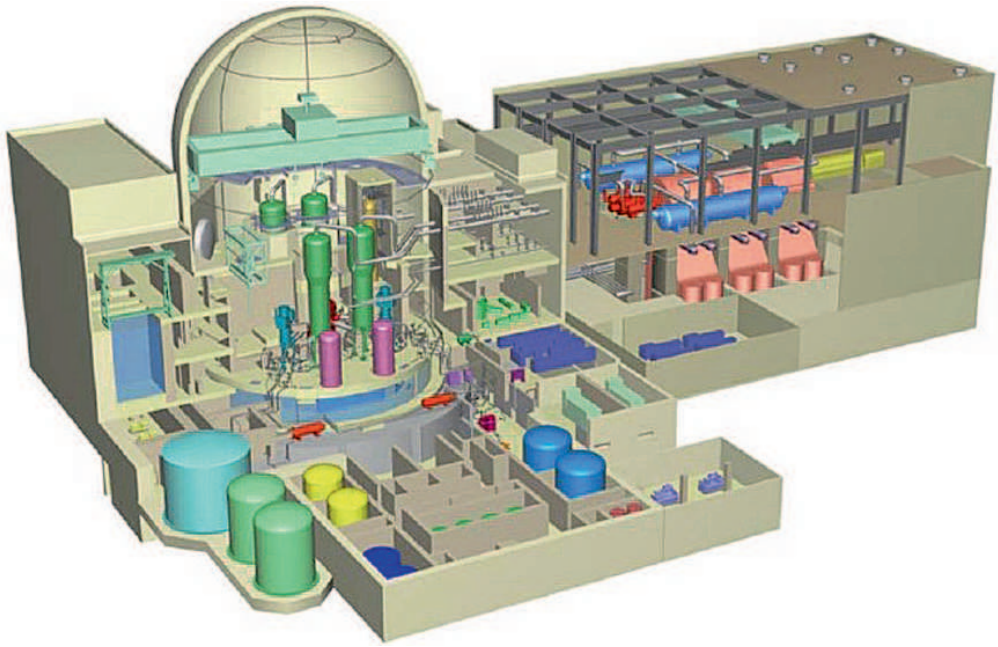


Fig. 9.56. Layout of a APWR systems (Printed with permission from [48]).

9.12.4 Evolutionary Power Reactor (EPR)

France has become the nuclear industry leader over the past two decades. The European Pressurized Reactor (EPR), shown in Fig. 9.59, was designed by Areva, which is a 1,700 MWe unit and uses four redundant safety loops and passive systems. Two of these plants are currently under construction in Europe and more are scheduled for construction worldwide. AREVA has submitted for regulator

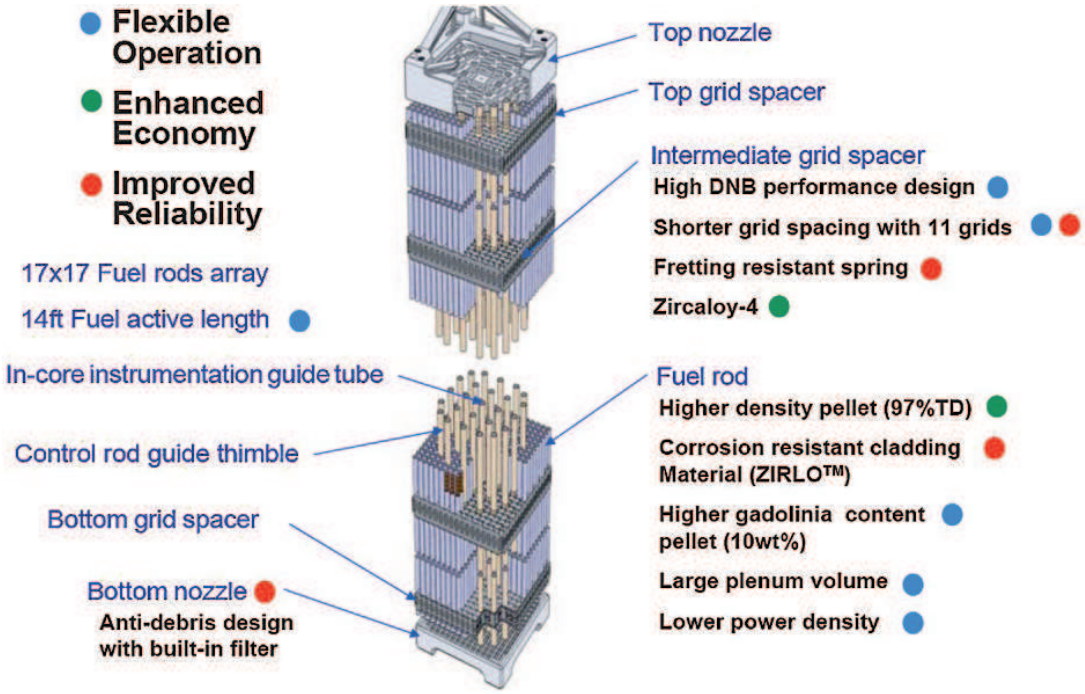


Fig. 9.57. Fuel assembly of USAPWR (Printed with permission from [48]).

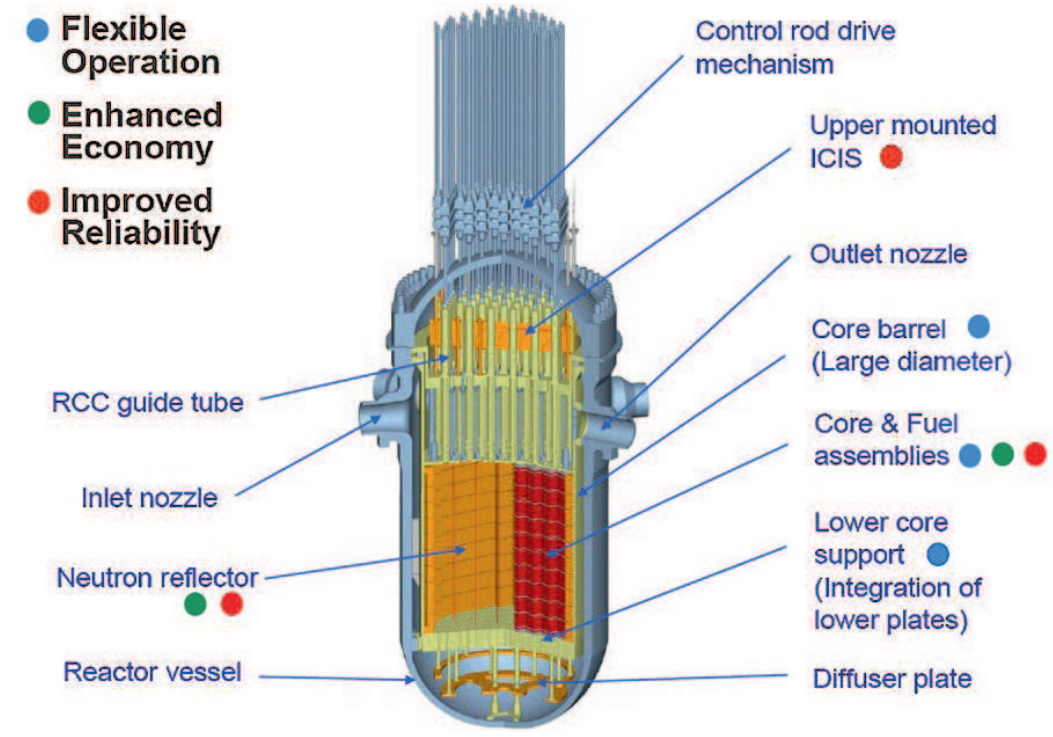


Fig. 9.58. The cutout view of US APWR designed by Mitsubishi Heavy Industries for US market (Printed with permission from [48]).

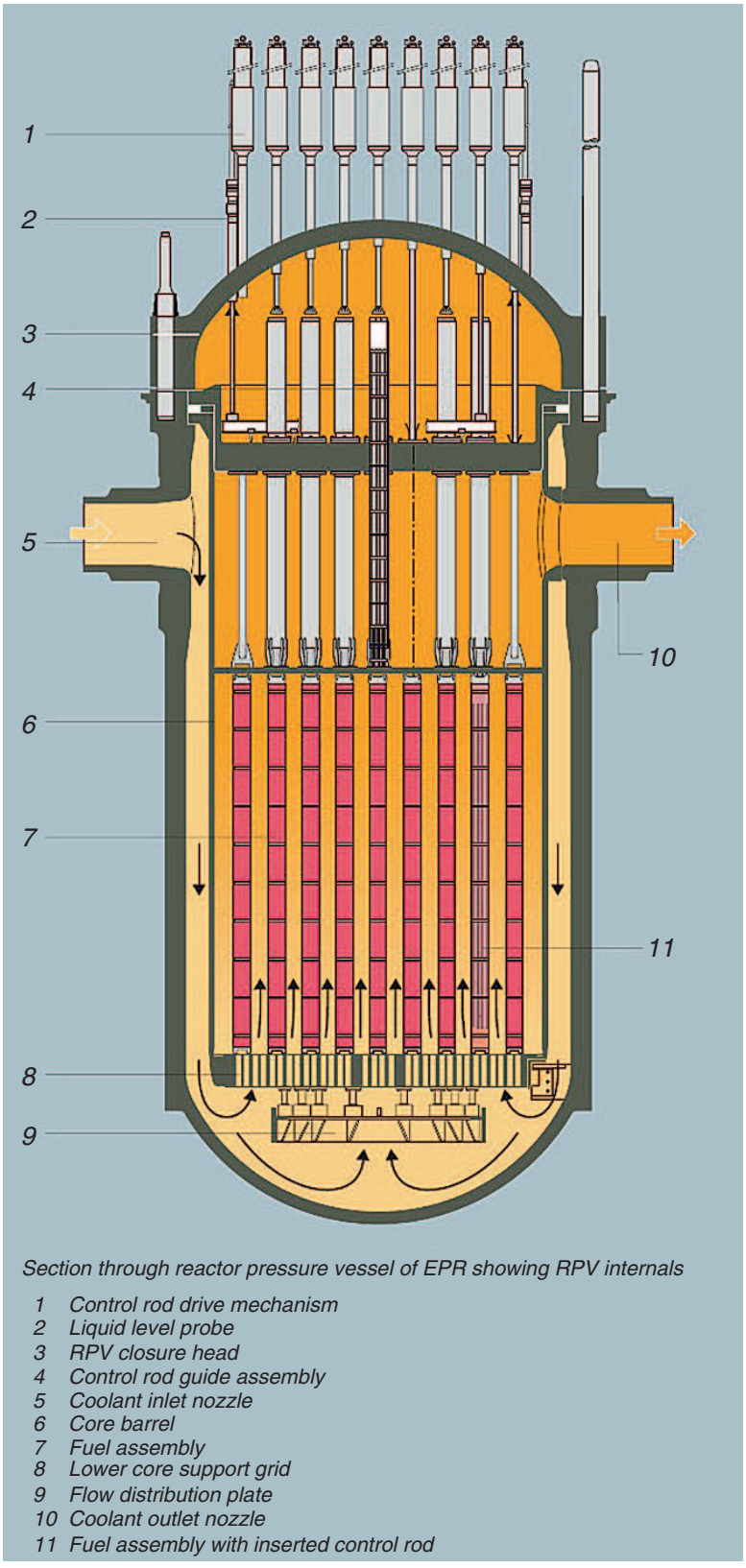


Fig. 9.59. Cut out of the EPR pressure vessel. US EPR (Adapted from [49]).

approval in several countries including the USA. This modified design for the USA has been renamed as the Evolutionary Power Reactor (EPR). The layout for an USEPR is shown in Fig. 9.60 and the cutaway of the reactor to be used in the USEPR is shown in Fig. 9.61.

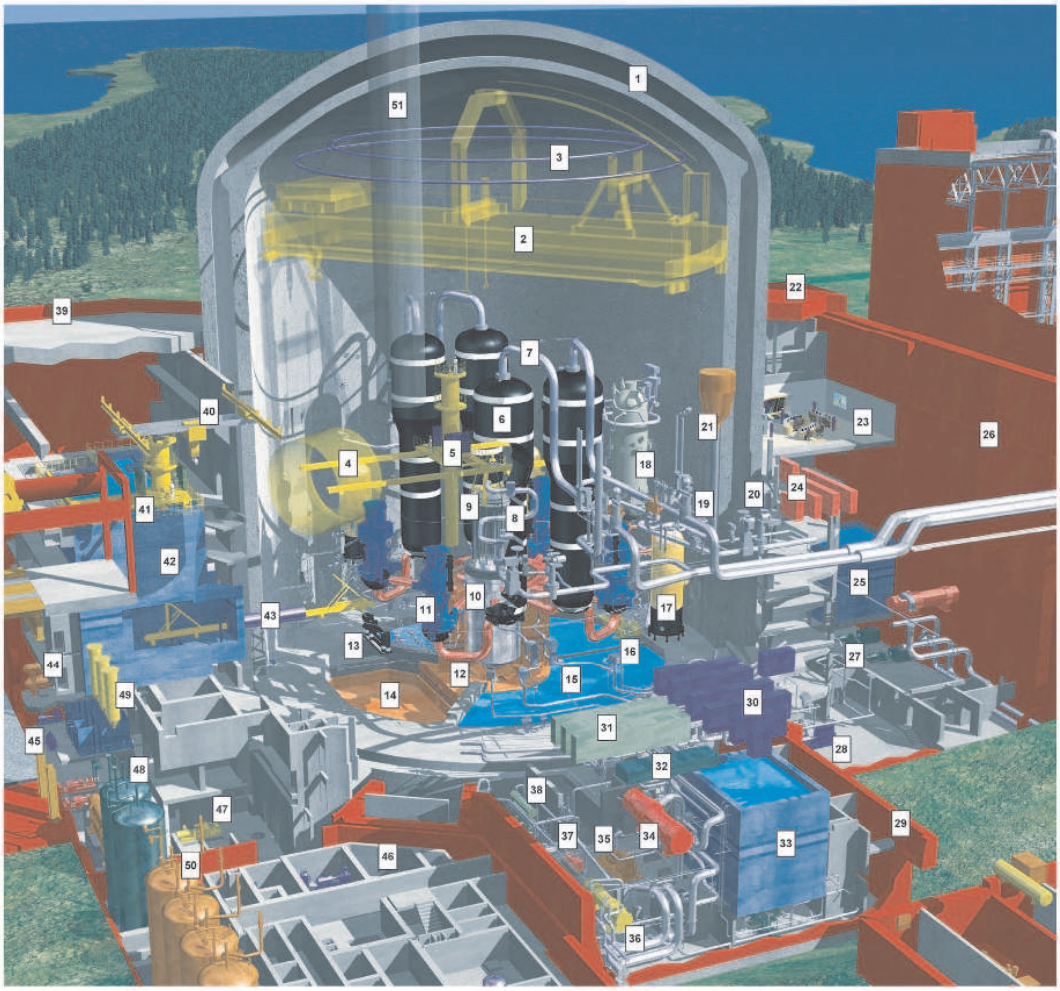


Fig. 9.60. The layout of the US-EPR power plant (Adapted from [49]).

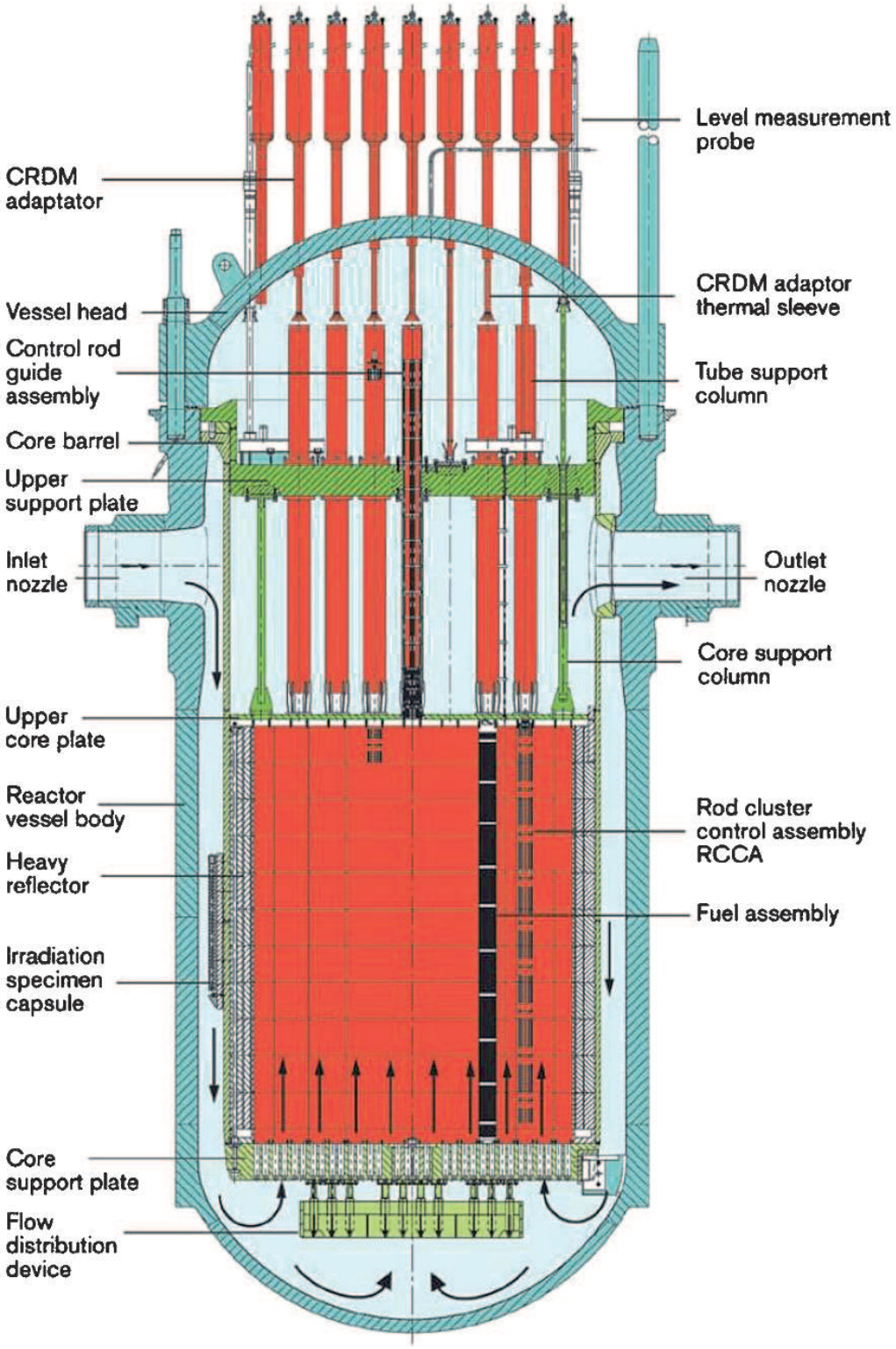


Fig. 9.61. The reactor pressure vessel for US-EPR (Adapted from [49]).

9.12.5 Economic & Simplified Boiling Water Reactor (ESBWR)

General Electric has designed a 1,390 MWe reactor, called the Economic & Simplified Boiling Water Reactor (ESBWR) to compete with other Generation III+ reactors [50–53]. The technical data of an ESBWR system is given in Table 9.10. The ESBWR is designed with natural circulation cooling and will have better safety features then the current BWR (Figs. 9.62–9.64). A comparison of various safety features between BWR and ESBWR is given in Table 9.11. This reactor is behind the ABWR and AP-1000 in terms of projected deployment, but orders have been already placed for the ESBWR.

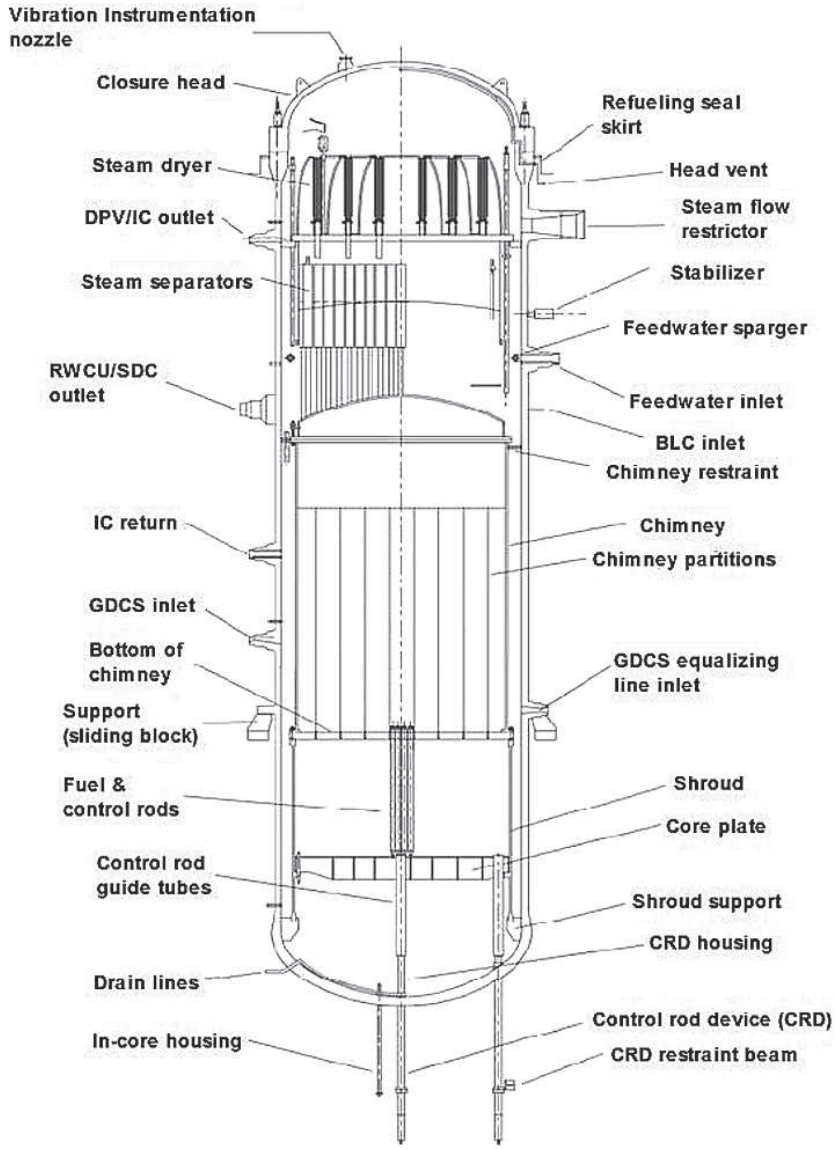


Fig. 9.62. A schematic of the reactor pressure vessel of Economic Simplified Boiling Water Reactor (ESBWR) (Printed with permission from [51]).

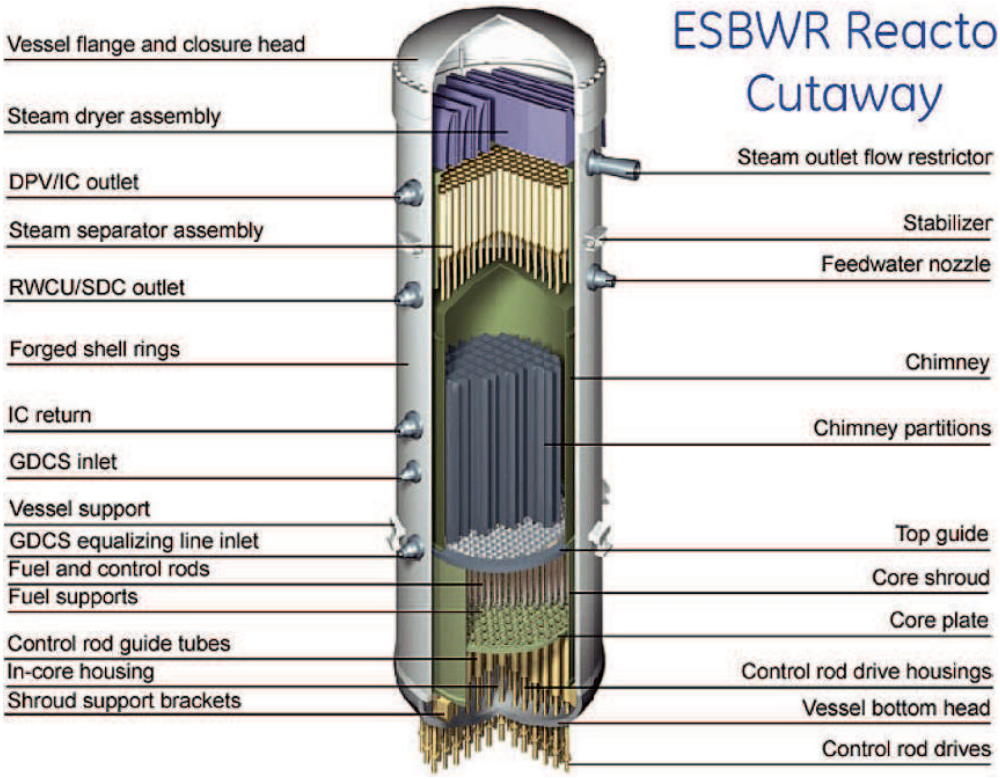


Fig. 9.63. Cutaway of ESBWR reactor (Courtesy of GE Hitachi Nuclear Energy).

Table 9.10. ESBWR, Technical fact sheet

Parameters	**Values**
Plant life (years)	60
Thermal power	4,500 MW
Electrical power	1,560 MW
Plant efficiency	34.7 %
Reactor type	Boiling water reactor
Core	
Fuel type	Enriched UO_2
Fuel enrichment	4.2% [c]
No. of fuel bundles	1,132
Coolant	Light water
Moderator	Light water
Operating cycle length[a]	12–24 months
Outage duration[b]	~14 days
Percent fuel replaced at refueling	See footnote [d]
Average fuel burnup at discharge	~50,000 MWd/MT
Number of Steam Lines	4
Number of feedwater trains	2
Containment Parameters	
Design temperature	340°F
Design pressure	45 psig

(Continued)

Table 9.10. (Continued)

Parameters	Values
Reactor Parameters	
Design temperature	575°F
Operating temperature	550°F
Design pressure	1,250 psig
Nominal operating pressure	1,040 psia
Feedwater & Turbine Parameters	
Turbine Inlet/Outlet temperature	543/93°F
Turbine Inlet/Outlet pressure	985/0.8 psia
Feedwater temperature	420°F
Feedwater pressure	1,050 psia
Feedwater flow	4.55 x 104 gpm
Steam mass flow rate	19.31 x 10^6 lbs/h
Yearly waste generated	
High level (spent fuel)	50 t
Intermediate level (spent resins, filters, etc.) and Low Level (compactables/non-compactables) Waste	1,765 ft^3

[a]Days of operation between outages; [b]For refueling only
[c] For a 24 month cycle; [d]20% for a 12 month cycle, 42% for a 24 month cycle
Source: Reference [53].

Table 9.11. A comparison of safety features between BWR and ESBWR.

Function	Current BWR reactors safety systems	ESBWR	
		Safety systems	Nonsafety
High-pressure inventory control	Motor and/or steam driven pumps with some vessel inventory loss and containment heat up	Isolation condensers conserve coolant inventory and avoid containment heat up	Multiple motor-driven pumps
Depressurization and low pressure inventory control	Automatic depressurization system with complex cooling water systems	Diverse/redundant automatic depressurization system using pool with gravity flow for inventory control	Diesel generator–driven pumps
Containment decay Heat removal	Diesel generator–driven pumped systems with complex cooling water systems and ultimate heat sink	Completely passive condensers with simple transfer of heat to pools that can boil off to the atmosphere	DG-driven pumps and cooling water
Fission product control and off-site doses	Double containment barriers and motor-driven filter and purge systems	Numerous in-containment natural removal mechanisms	HVAC systems
Severe accident features	Inserting or igniters for hydrogen control and other features to limit corium impact. Containment vent added as backup in ABWR. Lower drywell flooder. External reactor building connection to RPV.	Inert containment	Core catcher and passive lower drywell flooder to limit corium impact and the ability to easily connect portable systems

Source: Reference [52].

9.12.6 International Reactor Innovative & Secure (IRIS) Reactor

International Reactor Innovative & Secure (IRIS) reactor is designed by Westinghouse as a modular 335 MWe PWR. The IRIS pressure vessel contains the steam generators and coolant circulating pumps, adding significant accident mitigation features to the design (Fig. 9.64). The design certification for the IRIS is expected in 2010, with a first of a kind plant built by 2015 [54–57].

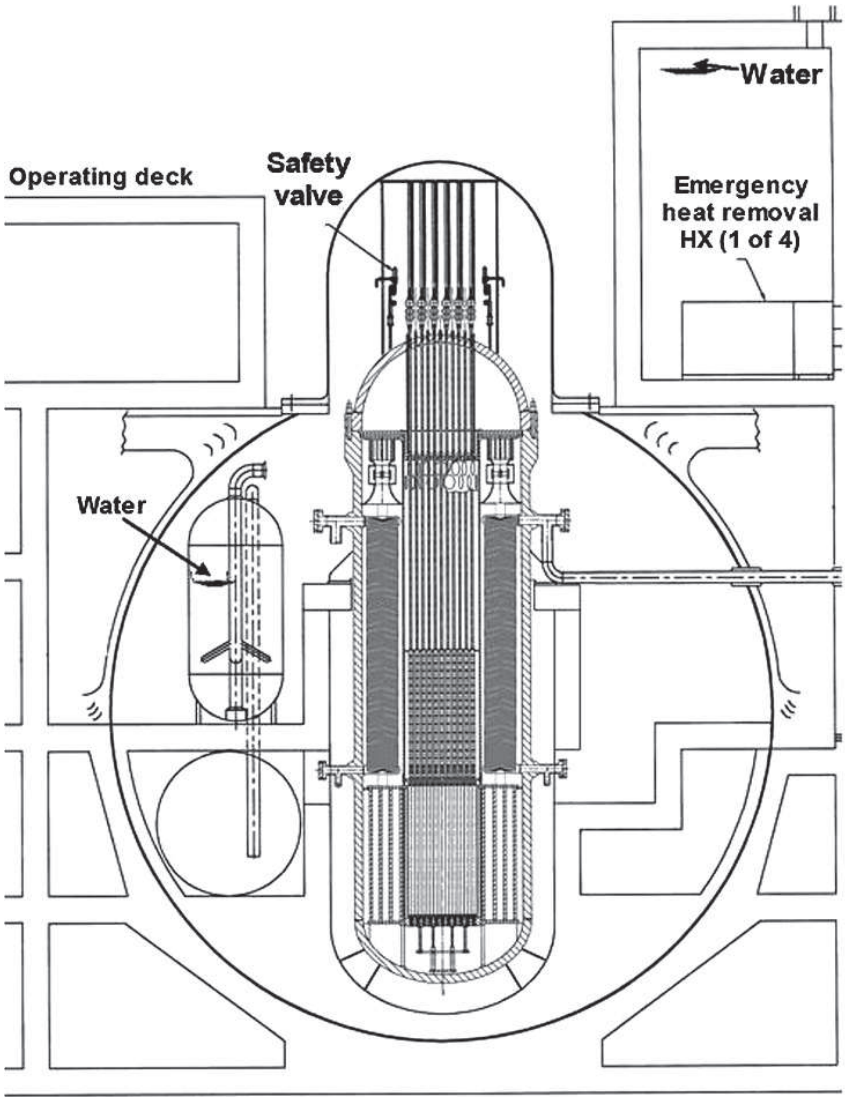


Fig. 9.64. The design of the steel containment building for IRIS system (Printed with permission from [56]).

The layout of the pressure vessel and the power plant are shown in Fig. 9.65 and Fig. 9.66, respectively.

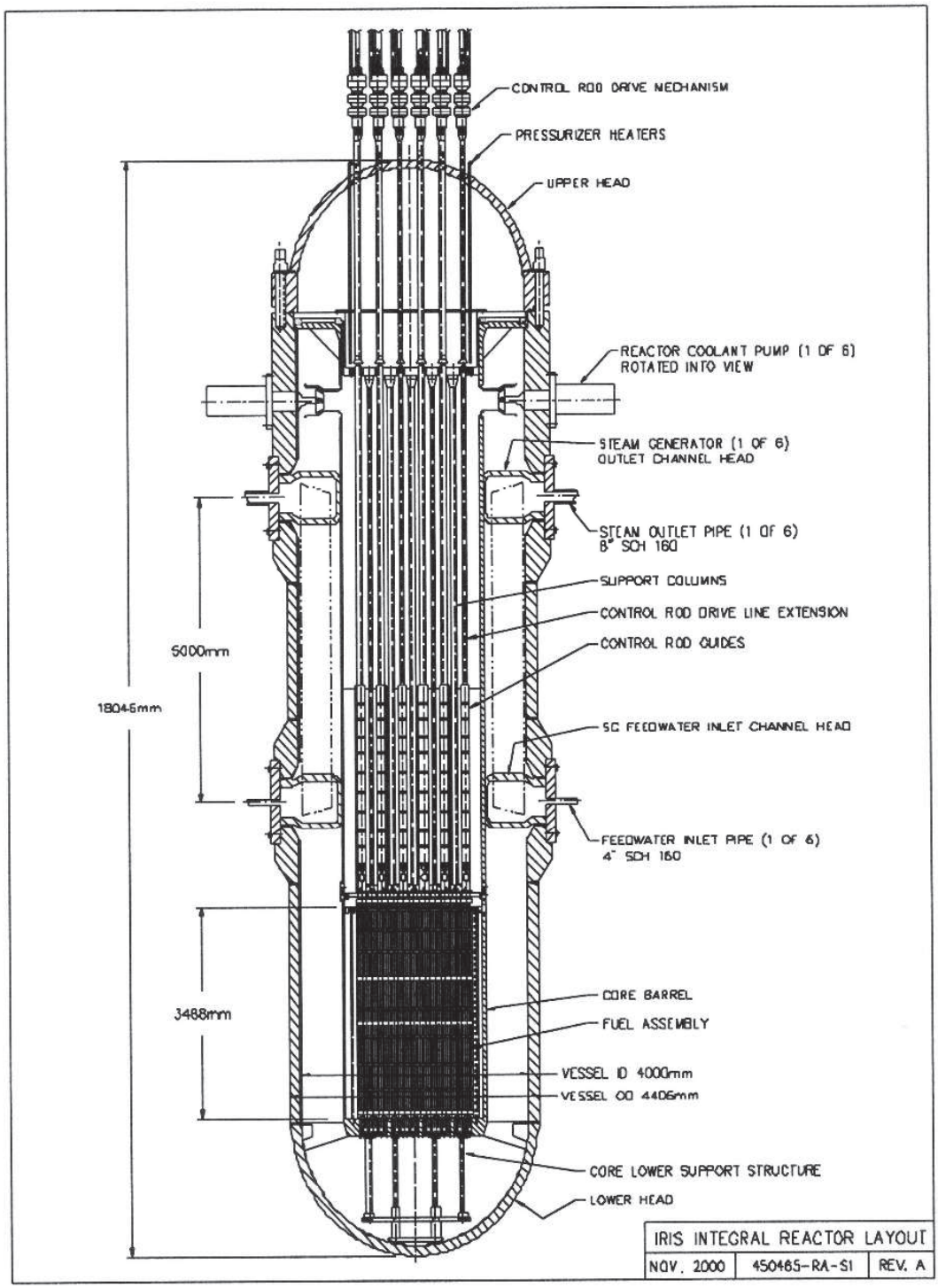


Fig. 9.65. Layout of reactor pressure vessel of IRIS reactor (Adapted from [57]).

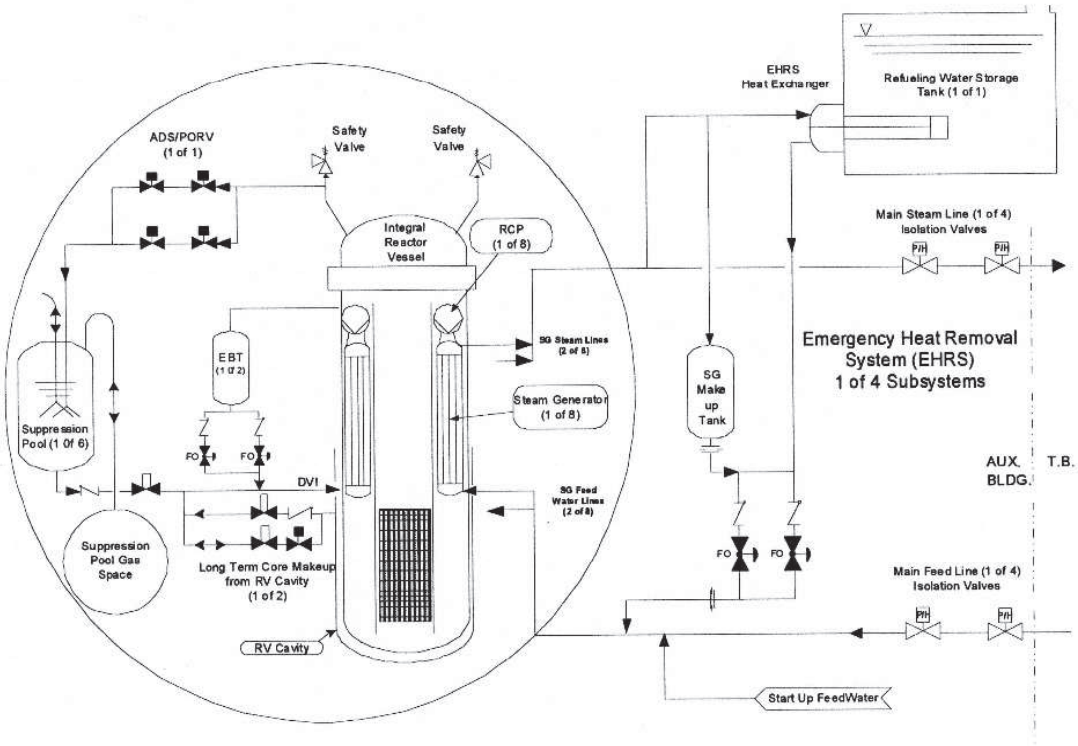


Fig. 9.66. Lyaout of a IRIS plant (Printed with permission from [56]).

9.12.7 APR-1400

South Korea's regulatory agency awarded design certification to the APR-1400 in 2003. This 1,450 MWe design evolved from a U.S. design developed in the 1900's, known as the US System 80+. This design incorporates the common attributes of generation III and III+ reactors in that it utilizes burnable poisons within the fuel to increase burn-up to 60 GWD/t, longer plant lifetime and increased safety features.

9.12.8 VVER-1200

Russia signed an agreement with Natsionalna Elektricheska Kompania and Atomstroyexport in January 2008 to build two new AES-92 plants in Bulgaria, utilizing VVER-1000 reactors, with a combined output of 2,000 MWe. These reactors are also being built in China and India. Russia has improved upon this design to create the VVER-1200, which has an output of 1150–1200 MWe with higher efficiency and longer anticipated lifetime. Russia is projecting to begin operating these reactors in 2012 and is hoping to connect an additional 20–25 of these reactors to the grid by 2020 [58].