Chapter 10

Nuclear Power as a Basis for Future Electricity Production in the World: Generation III and IV Reactors

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Additional information is available at the end of the chapter

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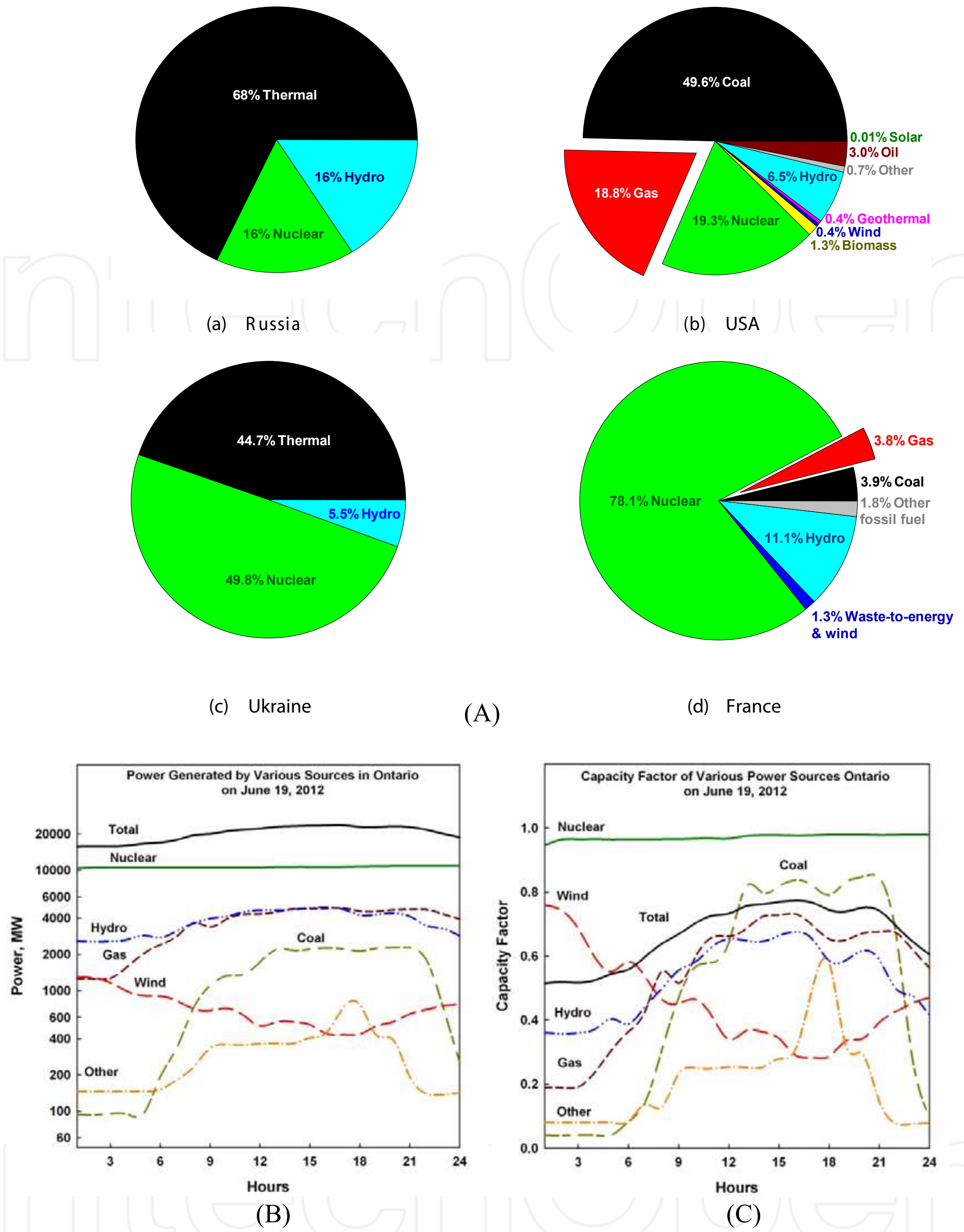
1. Introduction

It is well known that the electrical-power generation is the key factor for advances in any other industries, agriculture and level of living (see Table 1). In general, electrical energy can be produced by: 1) non-renewable sources such as coal, natural gas, oil, and nuclear; and 2) renewable sources such as hydro, wind, solar, biomass, geothermal and marine. However, the

No.	Country	Watts per person	Year	HDI* (2010)
1	Norway	2812	2005	1
2	Finland	1918	2005	16
3	Canada	1910	2005	8
4	USA	1460	2011	4
5	Japan	868	2005	11
6	France	851	2005	14
7	Germany	822	2009	10
8	Russia	785	2010	65
9	European Union	700	2005	
10	Ukraine	446	2005	69
11	China	364	2009	89
12	India	51	2005	119

* HDI – Human Development Index by United Nations; The HDI is a comparative measure of life expectancy, literacy, education and standards of living for countries worldwide. It is used to distinguish whether the country is a developed, a developing or an under-developed country, and also to measure the impact of economic policies on quality of life. Countries fall into four broad human-development categories, each of which comprises ~42 countries: 1) Very high – 42 countries; 2) high – 43; 3) medium – 42; and 4) low – 42.

Table 1. Electrical-energy consumption per capita in selected countries (Wikipedia, 2012).



*The net capacity factor of a power plant (Wikipedia, 2012) is the ratio of the actual output of a power plant over a period of time and its potential output if it had operated at full nameplate capacity the entire time. To calculate the capacity factor, take the total amount of energy the plant produced during a period of time and divide by the amount of energy the plant would have produced at full capacity. Capacity factors vary significantly depending on the type of fuel that is used and the design of the plant. Typical capacity factors for modern NPPs can be within 90%, thermal and hydro-electric power plants can be on average within 45% (can vary within 10 – 99% depending on local conditions), wind power plants – 20 – 40% and photovoltaic solar power plants – 15 – 20%.

Figure 1. (A). Electricity production by source in selected countries (data from 2005 – 2010 presented here just for reference purposes) (Wikipedia, 2012). (B). Power generated by various sources in the Province of Ontario (Canada) on June 19, 2012 (based on data from http://ieso.ca/imoweb/marketdata/genEnergy.asp). (C). Capacity factors* of various power sources in the Province of Ontario (Canada) on June 19, 2012 (based on data from http://ieso.ca/imoweb/marketdata/genEnergy.asp).

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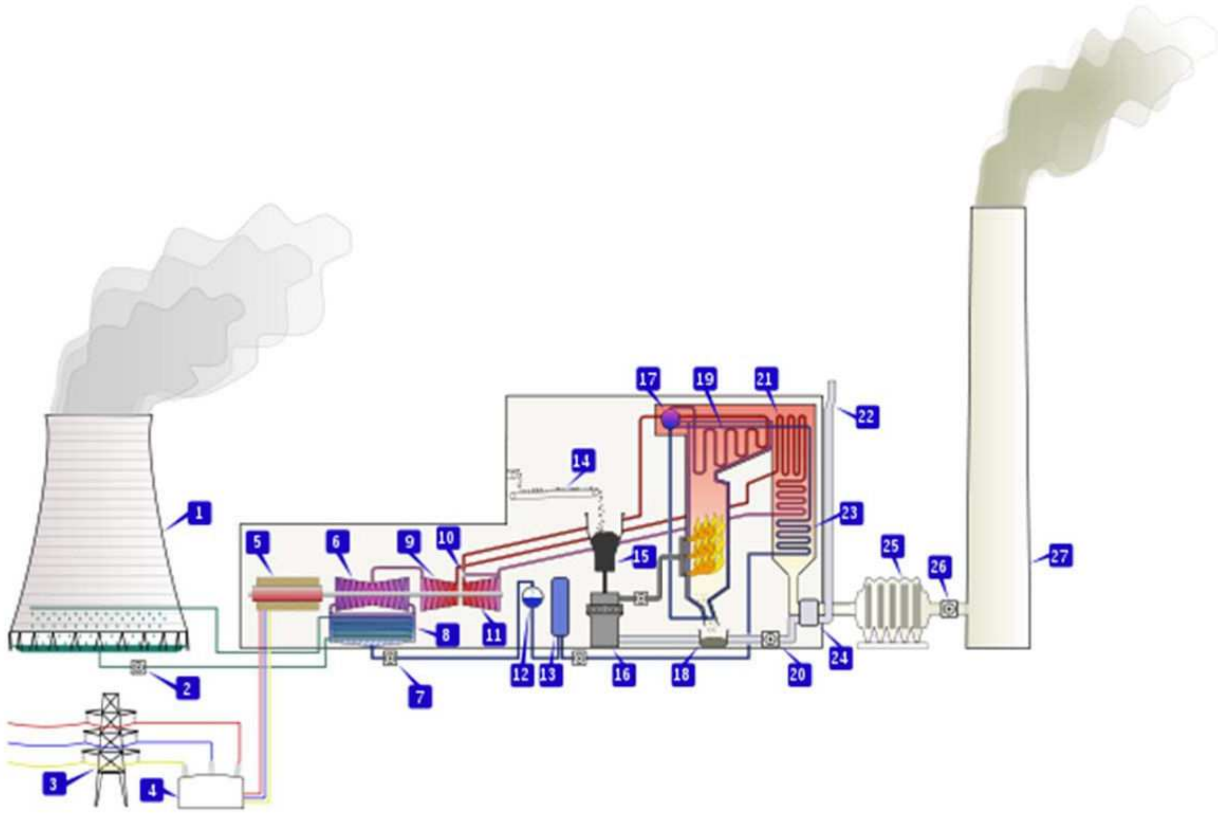
main sources for electrical-energy production are: 1) thermal - primary coal and secondary natural gas; 2) nuclear and 3) hydro. The rest of the sources might have visible impact just in some countries (see Figure 1). In addition, the renewable sources such as wind (see Figure 1b,c) and solar are not really reliable sources for industrial power generation, because they depend on Mother nature and relative costs of electrical energy generated by these and some other renewable sources with exception of large hydro-electric power plants can be significantly higher than those generated by non-renewable sources. Therefore, thermal and nuclear electrical-energy production will be considered further.

2. Thermal power plants

In general, the major driving force for all advances in thermal and Nuclear Power Plants (NPPs) is thermal efficiency. Ranges of thermal efficiencies of modern power plants are listed in Table 2 for references purposes.



1) Cooling tower; 2) Cooling-water pump; 3) Transmission line (3-phase); 4) Step-up transformer (3-phase); 5) Electrical generator (3-phase); 6) Low-pressure steam turbine; 7) Condensate pump; 8) Surface condenser; 9) Intermediate-pressure steam turbine; 10) Steam control valve; 11) High-pressure steam turbine; 12) Deaerator; 13) Feedwater heater; 14) Coal conveyor; 15) Coal hopper; 16) Coal pulverizer; 17) Boiler steam drum; 18) Bottom-ash hopper; 19) Superheater; 20) Forced-draught (draft) fan; 21) Reheater; 22) Combustion-air intake; 23) Economiser; 24) Air preheater; 25) Precipitator; 26) Induced-draught fan; and 27) Flue-gas stack.

Figure 2. Typical scheme of coal-fired thermal power plant (Wikipedia, 2012):

No	Power Plant	Gross Efficiency %
1	Combined-cycle power plant (combination of Brayton gas-turbine cycle (fuel natural or Liquefied Natural Gas (LNG); combustion-products parameters at the gas-turbine inlet: $T_{in} \approx 1650°C$) and Rankine steam-turbine cycle (steam parameters at the turbine inlet: $T_{in} \approx 620°C$ ($T_{cr} = 374°C$)) (see Figure 8).	Up to 62
2	Supercritical-pressure coal-fired thermal power plant (new plants) (Rankine-cycle steam inlet turbine parameters: $P_{in} \approx$ 25-38 MPa (P_{cr} = 22.064 MPa), $T_{in} \approx$ 540-625°C (T_{cr} = 374°C) and $T_{reheat} \approx$ 540-625°C) (see Figures 2 and 3).	Up to 55
3	Subcritical-pressure coal-fired thermal power plant (older plants) (Rankine-cycle steam: $P_{in} \approx$ 17 MPa, $T_{in} \approx$ 540°C (T_{cr} = 374°C) and $T_{reheat} \approx$ 540°C) (see Figure 2).	Up to 40
4	Carbon-dioxide-cooled reactor (Advanced Gas-cooled Reactor (AGR) (see Figure 12)) NPP (Generation III, current fleet) (reactor coolant – carbon dioxide: $P \approx$ 4 MPa and T_{in} / $T_{out} \approx$ 290 / 650°C; secondary Rankine-cycle steam: $P_{in} \approx$ 17 MPa ($T_{sat} \approx$ 352°C) and $T_{in} \approx$ 560°C (T_{cr} = 374°C))	Up to 42
5	Sodium-cooled Fast Reactor (SFR) NPP (see Figure 15) (Generation III and IV, currently just one reactor – BN-600 operates in Russia) (reactor coolant – liquid sodium: $P \approx$ 0.1 MPa and $T_{max} \approx$ 500-550°C; secondary Rankine-cycle steam: $P_{in} \approx$ 14 MPa ($T_{sat} \approx$ 337°C) and $T_{in} \approx$ 505°C (T_{cr} = 374°C)).	Up to 40
6	Pressurized Water Reactor (PWR) NPP (Generation III+, to be implemented within next 1–10 years) (reactor coolant – light water: $P \approx$ 16 MPa (T_{sat} = 347°C) and $T_{out} \approx$ 327°C; secondary Rankine-cycle steam: $P_{in} \approx$ 7.8 MPa and $T_{in} = T_{sat} \approx$ 293°C).	Up to 36-38
7	PWR NPP (see Figure 9) (Generation III, current fleet) (reactor coolant – light water: $P \approx$ 16 MPa (T_{sat} = 347°C) and T_{in} / $T_{out} \approx$ 290 / 325°C; secondary Rankine-cycle steam: $P_{in} \approx$ 7.2 MPa and $T_{in} = T_{sat} \approx$ 288°C).	32-36
8	Boiling Water Reactor (BWR) NPP (see Figure 10) (Generation III, current fleet) (reactor coolant light water; direct cycle; steam parameters at the turbine inlet: $P_{in} \approx$ 7.2 MPa and $T_{in} = T_{sat} \approx$ 288°C). Advanced BWR (ABWR) NPP (Generation III+) has approximately the same thermal efficiency.	~34
9	RBMK reactor (boiling reactor, pressure-channel design) NPP (see Figure 14) (Generation II and III, current fleet) (reactor coolant light water; direct cycle; steam parameters at the turbine inlet: $P_{in} \approx$ 6.6 MPa and $T_{in} = T_{sat} \approx$ 282°C).	~32
10	Pressurized Heavy Water Reactor (PHWR) NPP (see Figure 11) (Generation III, current fleet) (reactor coolant – heavy water: $P_{in} \approx$ 11 MPa; $P_{out} \approx$ 10 MPa (T_{sat} = 311°C) and T_{in} / $T_{out} \approx$ 265 / 310°C; secondary Rankine-cycle steam (light water): $P_{in} \approx$ 4.6 MPa and $T_{in} = T_{sat} \approx$ 259°C).	~32

[1]Gross thermal efficiency of a unit during a given period of time is the ratio of the gross electrical energy generated by a unit to the thermal energy of a fuel consumed during the same period by the same unit. The difference between gross and net thermal efficiencies includes internal needs for electrical energy of a power plant, which might be not so small (5% or even more).

Table 2. Typical ranges of thermal efficiencies (gross[1]) of modern thermal and nuclear power plants (shown just for reference purposes).

3. Coal-fired thermal power plants

For thousands years, mankind used and still is using wood and coal for heating purposes. For about 100 years, coal is used for generating electrical energy at coal-fired thermal power plants worldwide. All coal-fired power plants (see Figure 2) operate based on, so-called, steam Rankine cycle, which can be organized at two different levels of pressures: 1) older or smaller capacity power plants operate at steam pressures no higher than 16 – 17 MPa and 2) modern large capacity power plants operate at supercritical pressures from 23.5 MPa and up to 38 MPa (see Figure 3). Supercritical pressures[1] mean pressures above the critical pressure of water, which is 22.064 MPa (see Figure 4). From thermodynamics it is well known that higher thermal efficiencies correspond to higher temperatures and pressures (see Table 2). Therefore, usually subcritical-pressure plants have thermal efficiencies of about 34 – 40% and modern supercritical-pressure plants – 45 – 55%. Steam-generators outlet temperatures or steam-turbine inlet temperatures have reached level of about 625°C (and even higher) at pressures of 25 – 30 (35 – 38) MPa. However, a common level is about 535 – 585°C at pressures of 23.5 – 25 MPa (see Figure 3).

In spite of advances in coal-fired power-plants design and operation worldwide they are still considered as not environmental friendly due to producing a lot of carbon-dioxide emissions as a result of combustion process plus ash, slag and even acid rains (Pioro et al., 2010). However, it should be admitted that known resources of coal worldwide are the largest compared to that of other fossil fuels (natural gas and oil).

For better understanding specifics of supercritical water compared to water at subcritical pressures it is important to define special terms and expressions used at these conditions. For better understanding of these terms and expressions Figures 4 – 7 are shown below.

4. Definitions of selected terms and expressions related to critical and supercritical regions (Pioro and Mokry, 2011a)

Compressed fluid is a fluid at a pressure above the critical pressure, but at a temperature below the critical temperature.

Critical point (also called a *critical state*) is a point in which the distinction between the liquid and gas (or vapour) phases disappears, i.e., both phases have the same temperature, pressure and specific volume or density. The *critical point* is characterized by the phase-state parameters T_{cr}, P_{cr} and V_{cr} (or ρ_{cr}), which have unique values for each pure substance.

Near-critical point is actually a narrow region around the critical point, where all thermophysical properties of a pure fluid exhibit rapid variations.

Pseudocritical line is a line, which consists of pseudocritical points.

Pseudocritical point (characterized with P_{pc} and T_{pc}) is a point at a pressure above the critical pressure and at a temperature ($T_{pc} > T_{cr}$) corresponding to the maximum value of the specific heat at this particular pressure.

Supercritical fluid is a fluid at pressures and temperatures that are higher than the critical pressure and critical temperature. However, in the present chapter, a term *supercritical fluid* includes both terms – a *supercritical fluid* and *compressed fluid*.

1 See some explanations on supercritical-pressures specifics at the end of this section.

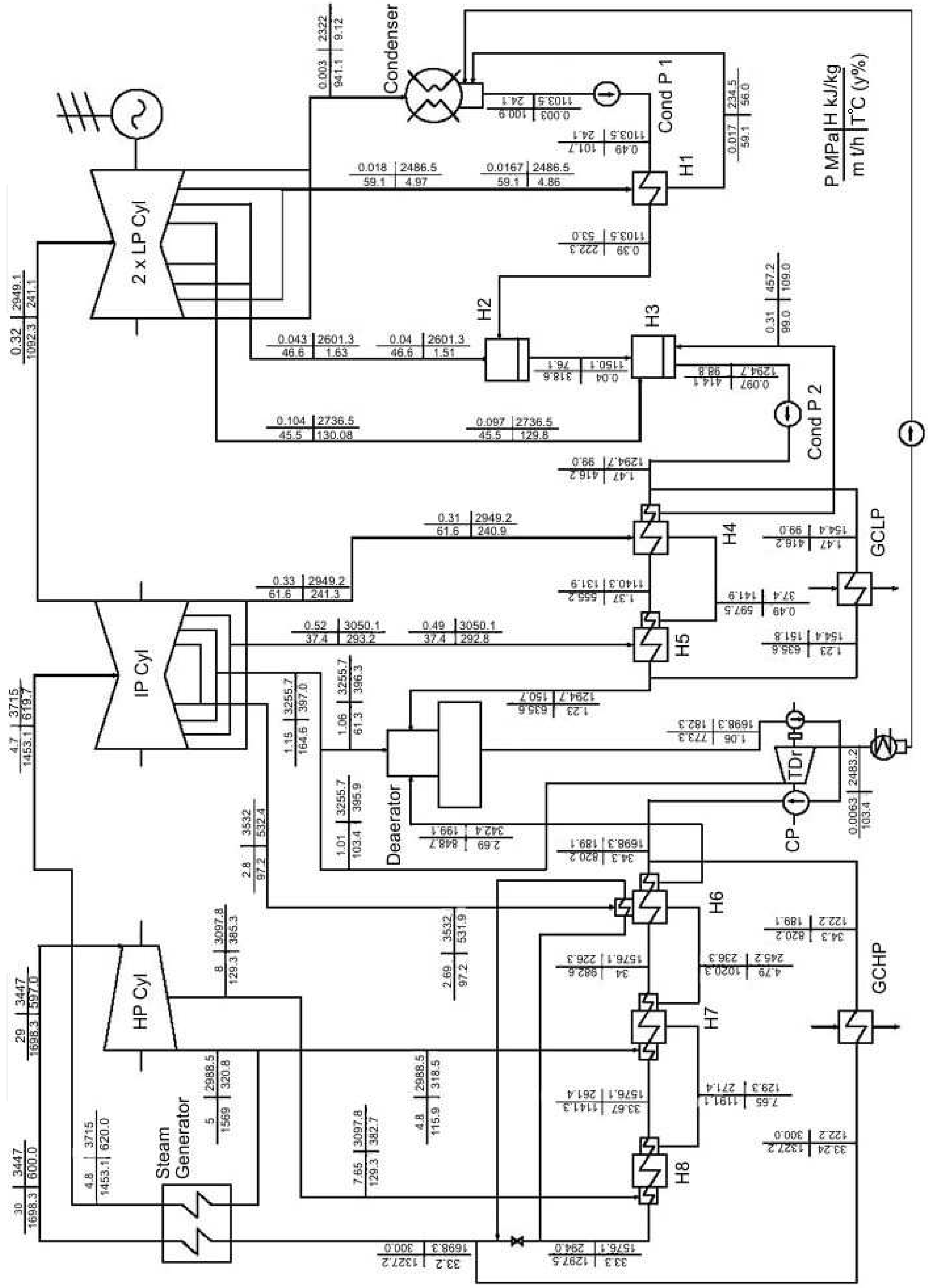


Figure 3. Supercritical-pressure single-reheat regenerative cycle 600-MW$_{el}$ Tom'-Usinsk thermal power plant (Russia) layout (Kruglikov et al., 2009): Cond P – Condensate Pump; CP – Circulation Pump; Cyl – Cylinder; GCHP – Gas Cooler of High Pressure; GCLP – Gas Cooler of Low Pressure; H – Heat exchanger (feedwater heater); HP – High Pressure; IP – Intermediate Pressure; LP – Low Pressure; and TDr – Turbine Drive.

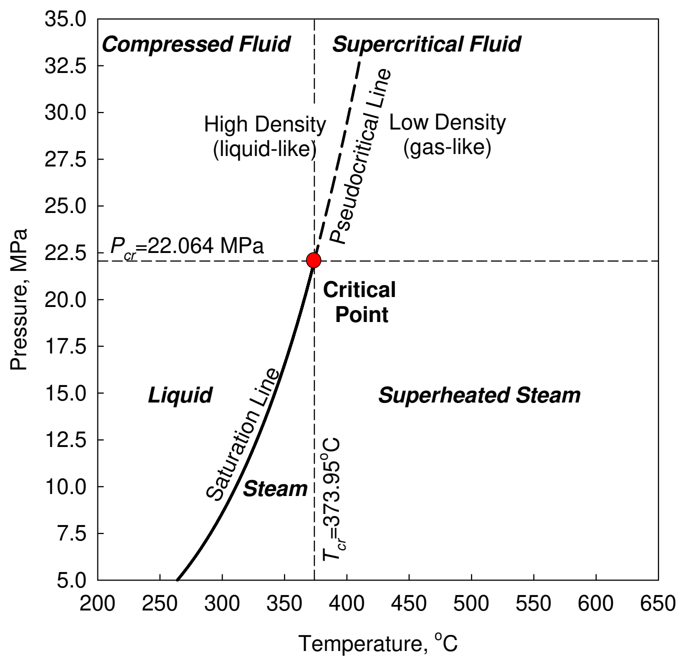


Figure 4. Pressure-Temperature diagram for water.

Supercritical "steam" is actually supercritical water, because at supercritical pressures fluid is considered as a single-phase substance. However, this term is widely (and incorrectly) used in the literature in relation to supercritical "steam" generators and turbines.

Superheated steam is a steam at pressures below the critical pressure, but at temperatures above the critical temperature.

General trends of various properties near the critical and pseudocritical points (Pioro et al., 2011; Pioro and Mokry, 2011a; Pioro and Duffey, 2007) can be illustrated on a basis of those of water. Figure 5 shows variations in basic thermophysical properties of water at a supercritical pressure of 25 MPa (also, in addition, see Figure 6). Thermophysical properties of 105 pure fluids including water, carbon dioxide, helium, refrigerants, etc., 5 pseudo-pure fluids (such as air) and mixtures with up to 20 components at different pressures and temperatures, including critical and supercritical regions, can be calculated using the NIST REFPROP software (2010).

At critical and supercritical pressures a fluid is considered as a single-phase substance in spite of the fact that all thermophysical properties undergo significant changes within critical and pseudocritical regions (see Figure 5). Near the critical point, these changes are dramatic. In the vicinity of pseudocritical points, with an increase in pressure, these changes become less pronounced (see Figure 6).

At supercritical pressures properties such as density (see Figure 5) and dynamic viscosity undergo a significant drop (near the critical point this drop is almost vertical) within a very narrow temperature range, while the kinematic viscosity and specific enthalpy (see Figure 5)

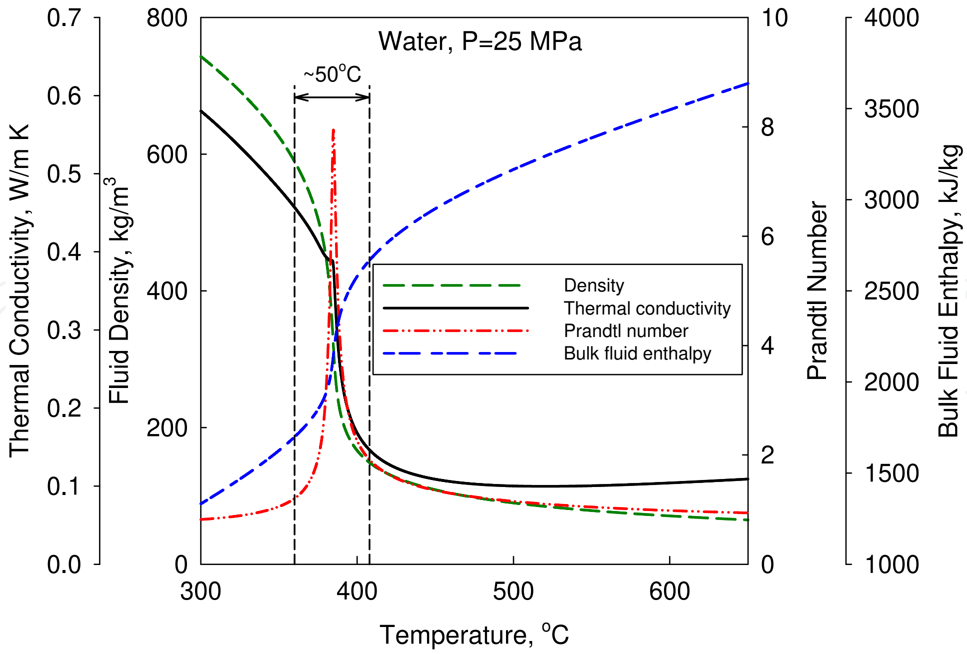


Figure 5. Variations of selected thermophysical properties of water near pseudocritical point: Pseudocritical region at 25 MPa is about ~50°C.

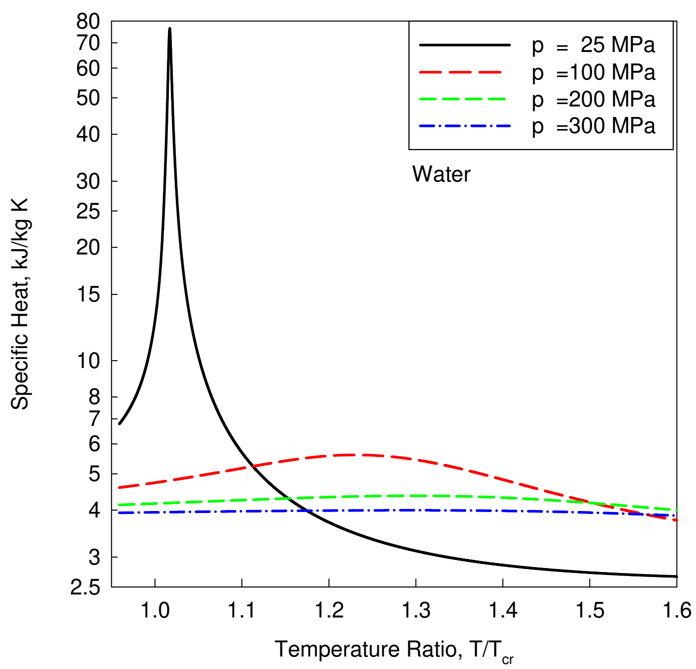


Figure 6. Specific heat variations at various supercritical pressures: Water.

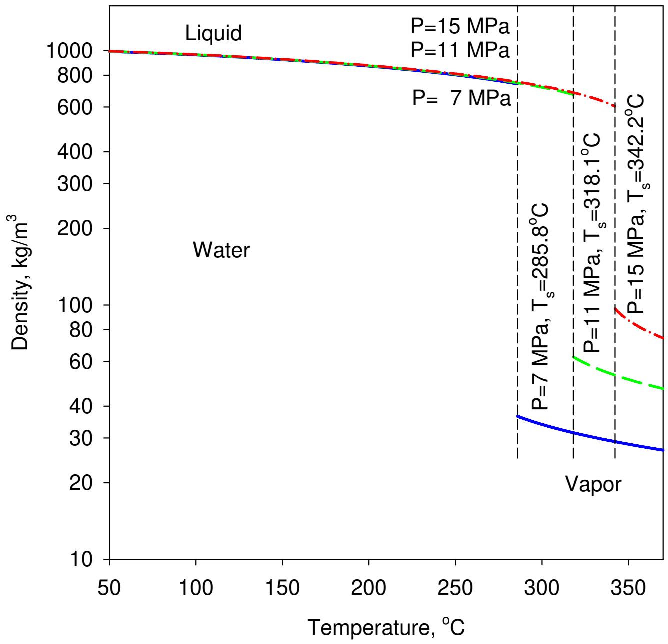


Figure 7. Density variations at various subcritical pressures for water: Liquid and vapour.

undergo a sharp increase. The volume expansivity, specific heat, thermal conductivity and Prandtl number have peaks near the critical and pseudocritical points (see Figures 5 and 6). Magnitudes of these peaks decrease very quickly with an increase in pressure (see Figure 6). Also, "peaks" transform into "humps" profiles at pressures beyond the critical pressure. It should be noted that the dynamic viscosity, kinematic viscosity and thermal conductivity (see Figure 5) undergo through the minimum right after critical and pseudocritical points.

The specific heat of water (as well as of other fluids) has a maximum value in the critical point. The exact temperature that corresponds to the specific-heat peak above the critical pressure is known as a pseudocritical temperature (see Figure 4). At pressures approximately above 300 MPa (see Figure 6) a peak (here it is better to say "a hump") in specific heat almost disappears, therefore, such term as a *pseudocritical point* does not exist anymore. The same applies to the *pseudocritical line*. It should be noted that peaks in the thermal conductivity and volume expansivity may not correspond to the pseudocritical temperature (Pioro et al., 2011; Pioro and Mokry, 2011a; Pioro and Duffey, 2007).

In general, crossing the pseudocritical line from left to right (see Figure 4) is quite similar as crossing the saturation line from liquid into vapour. The major difference in crossing these two lines is that all changes (even drastic variations) in thermophysical properties at supercritical pressures are gradual and continuous, which take place within a certain temperature range (see Figure 5). On the contrary, at subcritical pressures there is properties discontinuation on the saturation line: one value for liquid and another for vapour (see Figure 7). Therefore, supercritical fluids behave as single-phase substances (Gupta et al., 2012). Also, when dealing with supercritical fluids we usually apply the term "*pseudo*" in front of a *critical point*, *boiling*, *film boiling*, etc. Specifics of heat transfer at supercritical pressures can be found in Pioro et al. (2011), Mokry et al. (2011), Pioro and Mokry (2011b), and Pioro and Duffey (2007).

5. Combined-cycle thermal power plants

Natural gas is considered as a relatively "clean" fossil fuel compared to coal and oil, but still emits a lot of carbon dioxide due to combustion process when it used for electrical generation. The most efficient modern thermal power plants with thermal efficiencies within a range of 50 – 62% are, so-called, combined-cycle power plants, which use natural gas as a fuel (see Figure 8).

In spite of advances in thermal power plants design and operation, they still emit carbon dioxide into atmosphere, which is currently considered as one of the major reasons for a climate change. In addition, all fossil-fuel resources are depleting quite fast. Therefore, a new reliable and environmental friendly source for the electrical-energy generation should be considered.

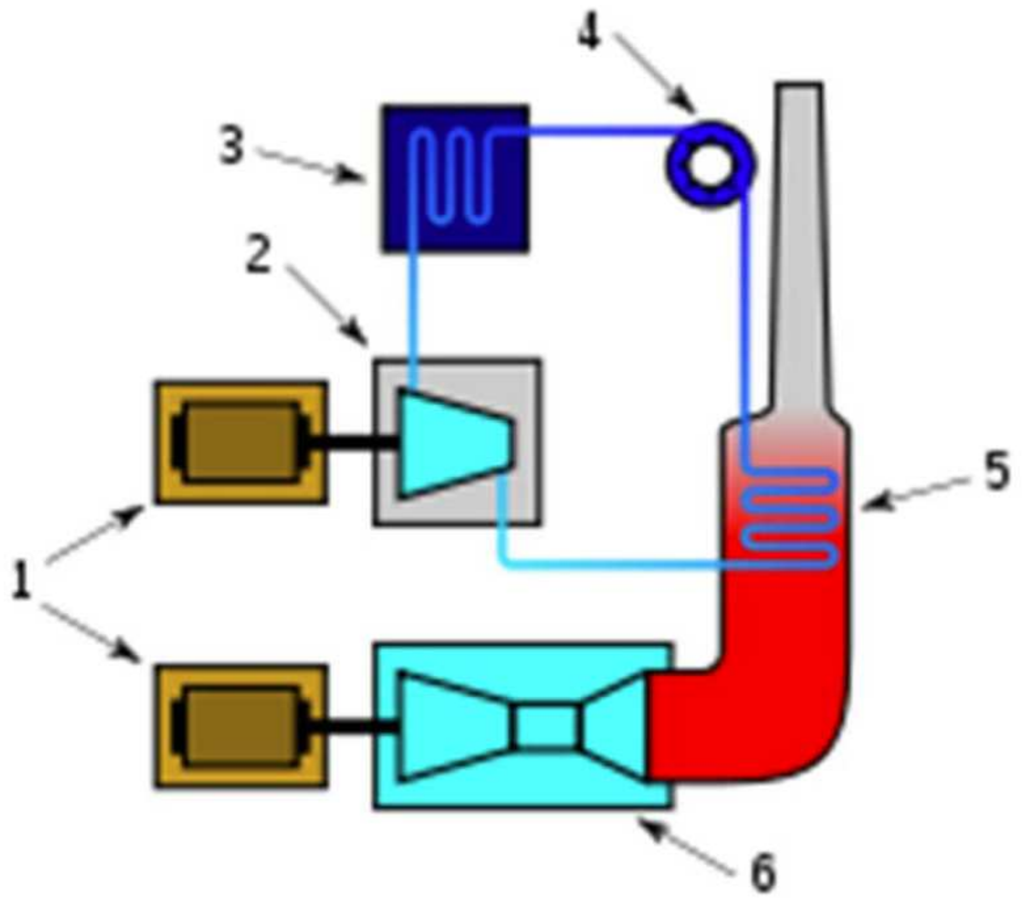


Figure 8. Working principle of combined-cycle thermal power plant (gas turbine (Brayton cycle) and steam turbine (Rankine cycle) plant) (Wikipedia, 2012): 1 electrical generators; 2 steam turbine; 3 condenser; 4 circulation pump; 5 steam generator / exhaust-gases heat exchanger; and 6 gas turbine.

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6. Nuclear power plants

6.1. Modern nuclear reactors

Nuclear power is also a non-renewable source as the fossil fuels, but nuclear resources can be used for significantly longer time than some fossil fuels plus nuclear power does not emit carbon dioxide into atmosphere. Currently, this source of energy is considered as the most viable one for electrical generation for the next 50 – 100 years.

For better understanding specifics of current and future nuclear-power reactors it is important to define their various classifications.

6.2. Classifications of nuclear-power reactors

1. By neutron spectrum: (a) thermal (the vast majority of current nuclear-power reactors), (b) fast (currently, only one nuclear-power reactor is in operation in Russia: SFR – BN-600), and (c) interim or mixed spectrum.

2. By reactor-core design:

i. Neutron-core design: (a) homogeneous, i.e., the fuel and reactor coolant are mixed together (one of the Generation IV nuclear-reactors concepts) and (b) heterogeneous, i.e., the fuel and reactor coolant are separated through a sheath or cladding (currently, all nuclear-power reactors);

ii. General core design: (a) Pressure-Vessel (PV) (the majority of current nuclear-power reactors including PWRs, BWRs, etc.) and (b) Pressure-Channel (PCh) or Pressure-Tube (PT) reactors (CANDU ((CANada Deuterium-Uranium) reactors, RBMKs), EGPs (Power Heterogeneous Loop reactor (in Russian abbreviations)), etc.).

3. By coolant:

i. Water-cooled reactors: (a) Light-Water (H_2O) Reactors (LWRs) - PWRs, BWRs, RBMKs, EGPs, and (b) heavy-water (D_2O) reactors – mainly CANDU-type reactors.

ii. Gas-cooled reactors: Carbon-dioxide-cooled reactors (Magnox[2] reactors (Gas-Cooled Reactors (GCRs)) and AGRs) and helium-cooled reactors (two Generation IV nuclear-reactor concepts); (c) liquid-metal-cooled reactors: SFR, lead-cooled and lead-bismuth-cooled reactors (Generation IV nuclear-reactor concepts); (d) molten-salt-cooled reactors (one of Generation IV nuclear-reactor concepts); and (e) organic-fluids-cooled reactors (existed only as experimental reactors some time ago).

4. By type of a moderator (Kirillov et al., 2007): (a) liquid moderator (H_2O and D_2O are currently used in nuclear-power reactors as moderators) and (b) solid moderator (graphite[3] (RBMKs, EGPs, Magnox reactors (GCRs), AGRs), zirconium hydride (ZrH_2), beryllium (Be) and beryllium oxide (BeO)).

2 In this reactor the fuel-rod sheath is made of magnesium alloy known by the trade name as "Magnox", which was used as the name of the reactor (Hewitt and Collier, 2000).

5. By application: (a) power reactors (PWRs, BWRs, CANDU reactors, GCRs, AGRs, RBMKs, EGPs, SFR from current fleet) (b) research reactors (for example, NRU (National Research Universal) (AECL, Canada, http://www.aecl.ca/Programs/NRU.htm), etc.), (c) transport or mobile reactors (submarines and ships (icebreakers, air-carriers, etc.), (d) industrial reactors for isotope production (for example, NRU), etc., and (e) multipurpose reactors (for example, NRU, etc.).

6. By number of flow circuits: (a) single-flow circuit (once-through or direct-cycle reactors) (BWRs, RBMKs, EGPs); (b) double-flow circuit (PWRs, PHWRs, GCRs, AGRs) and (c) triple-flow circuit (usually SFRs).

7. By fuel enrichment: (a) Natural-Uranium fuel (NU) (99.3%$_{wt}$ of non-fissile isotope uranium-238 (^{238}U) and 0.7% of fissile isotope uranium-235 (^{235}U)) (CANDU-type reactors, Magnox reactors), (b) Slightly-Enriched Uranium (SEU) (0.8 – 2%$_{wt}$ of ^{235}U), (c) Low-Enriched Uranium (LEU) (2 – 20% of ^{235}U) (the vast majority of current nuclear-power reactors: PWRs, BWRs, AGRs, RBMKs, EGPs), and (d) Highly-Enriched Uranium (HEU) (>20%$_{wt}$ of ^{235}U) (can be SFR).

8. By used fuel (Peiman et al., 2012): (a) Conventional nuclear fuels (low thermal conductivity): Uranium dioxide (UO_2, used in the vast majority of nuclear-power reactors), Mixed OXides (MOX) ($(U_{0.8}Pu_{0.2})O_2$, where 0.8 and 0.2 are the molar parts of UO_2 and PuO_2, used in some reactors) and thoria (ThO_2) (considered for a possible use instead of UO_2 in some countries, usually, with large resources of this type of fuel, for example, in India); and (b) alternative nuclear fuels (high thermal conductivity): Uranium dioxide plus silicon carbide (UO_2–SiC), uranium dioxide composed of graphite fibre (UO_2–C), uranium dioxide plus beryllium oxide (UO_2–BeO), uranium dicarbide (UC_2), uranium monocarbide (UC) and uranium mononitride (UN); the last three fuels are mainly intended for use in high-temperature Generation IV reactors.

First success of using nuclear power for electrical generation was achieved in several countries within 50-s, and currently, Generations II and III nuclear-power reactors are operating around the world (see Tables 3 and 4 and Figures 9-15). In general, definitions of nuclear-reactors generations are as the following: 1) Generation I (1950 – 1965) – early prototypes of nuclear reactors; 2) Generation II (1965 – 1995) – commercial power reactors; 3) Generation III (1995 – 2010) – modern reactors (water-cooled NPPs with thermal efficiencies within 30 – 36%; carbon-dioxide-cooled NPPs with the thermal efficiency up to 42% and liquid sodium-cooled NPPs with the thermal efficiency up to 40%) and Generation III+ (2010 – 2025) – reactors with improved parameters (evolutionary design improvements) (water-cooled NPPs with the thermal efficiency up to 38%) (see Table 5); and 4) Generation IV (2025 - …) – reactors in principle with new parameters (NPPs with the thermal efficiency of 43 – 50% and even higher for all types of reactors).

3 After the Chernobyl NPP severe nuclear accident in Ukraine in 1986 with the RBMK reactor, graphite is no longer considered as a possible moderator in any water-cooled reactors.

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1. PWRs (see Figure 9 and Tables 6 and 7) – 267 (*268*) (248 (*247*) GW_{el}); forthcoming – 89 (93 GW_{el}).
2. BWRs or ABWRs (see Figure 10 and Table 8) – 84 (*92*) (85 (*78*) GW_{el}); forthcoming – 6 (8 GW_{el}).
3. GCRs (see Figures 12 and 13) – 17 (*18*) (9 GW_{el}), UK (AGRs (see Figure 12) – 14 and Magnox (see Figure 13) – 3); forthcoming – 1 (0.2 GW_{el}).
4. PHWRs (see Figure 11) – 51 (*50*) (26 (*25*) GW_{el}), Argentina 2, Canada 22, China 2, India 18, Pakistan 1, Romania 2, S. Korea 4; forthcoming – 9 (5 GW_{el}).
5. Light-water, Graphite-moderated Reactors (LGRs) (see Figure 14 and Table 6) – 15 (10 GW_{el}), Russia, 11 RBMKs and 4 EGPs[1] (earlier prototype of RBMK).
6. Liquid-Metal Fast-Breeder Reactors (LMFBRs) (see Figure 15 and Table 6) – 1 (0.6 GW_{el}), SFR, Russia; forthcoming – 4 (1.5 GW_{el}).

[1]EGP –channel-type, graphite moderated, light water, boiling reactor with natural circulation.

Table 3. Operating and forthcoming nuclear-power reactors (in total - 435 (*444*) (net 370 (*378*) GW_{el}) (Nuclear News, 2012); (*in Italic mode*) - number of power reactors before the Japan earthquake and tsunami disaster in spring of 2011) (Nuclear News, 2011).

No.	**Nation**	**# Units**	**Net GW_{el}**
1.	USA	104	103
2.	France	58	63
3.	Japan[1]	50 (*54*)	44 (*47*)
4.	Russia	33	24
5.	S. Korea	21 (*20*)	19 (*18*)
6.	Canada[2]	22	15
7.	Ukraine	15	13
8.	Germany	9 (*17*)	12 (*20*)
9.	UK	18 (*19*)	10
10.	China 14 (*13*)	11 (*10*)	

[1]Currently, i.e., in October of 2012, only 2 reactors in operation. However, more reactors are planned to put into operation.

[2]Currently, i.e., October of 2012, 18 reactors in operation and 4 already shut-down.

Table 4. Current nuclear-power reactors by nation (10 first nations) (Nuclear News, 2012); (*in Italic mode*) - number of power reactors before the Japan earthquake and tsunami disaster in spring of 2011) (Nuclear News, 2011).

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ABWR – Toshiba, Mitsubishi Heavy Industries and Hitachi-GE (Japan-USA) (the only one Generation III+ reactor design already implemented in the power industry).	
Advanced CANDU Reactor (ACR-1000) AECL, Canada.	
Advanced Plant (AP-1000) – Toshiba-Westinghouse (Japan-USA) (6 under construction in China and 6 planned to be built in China and 6 – in USA).	
Advanced PWR (APR-1400) – South Korea (4 under construction in S. Korea and 4 planned to be built in United Arad Emirates).	
European Pressurized-water Reactor (EPR) AREVA, France (1 should be put into operation in Finland, 1 under construction in France and 2 in China and 2 planned to be built in USA).	
VVER[1] (design AES[2]-2006 or VVER-1200 with ~1200 MW_{el}) – GIDROPRESS, Russia (2 under construction in Russia and several more planned to be built in various countries). Reference parameters of Generation III+ VVER (Ryzhov et al., 2010) are listed below:	
Parameter	Value
Thermal power, MW_{th}	3200
Electric power, MW_{el}	1160
NPP thermal efficiency, %	36
Primary coolant pressure, MPa	16.2
Steam-generator pressure, MPa	7.0
Coolant temperature at reactor inlet, °C	298
Coolant temperature at reactor outlet, °C	329
NPP service life, years	50
Main equipment service life, years	60
Replaced equipment service life, years, not less than	30
Capacity factor, %	up to 90
Load factor, %	up to 92
Equipment availability factor	99
Length of fuel cycle, years	4-5
Frequency of refuellings, months	12-18
Fuel assembly maximum burn-up, MW day/kgU	up to 60-70
Inter-repair period length, years	4-8
Annual average length of scheduled shut-downs (for refuellings, scheduled maintenance work), days per year	16-40
Refueling length, days per year	≤16

Number of not scheduled reactor shutdowns per year	≤1
Frequency of severe core damage, 1/year	$<10^{6}$
Frequency of limiting emergency release, 1/year	$<10^{7}$
Efficient time of passive safety and emergency control system operation without operator's action and power supply, hour	≥24
OBE/SSE, magnitude of MSK-64 scale	6 and 7*
Compliance with EUR requirements, yes/no	Yes
*RP main stationary equipment is designed for SSE of magnitude 8.	

[1]VVER or WWER - Water Water Power Reactor (in Russian abbreviations).

[2]AES – Atomic Electrical Station (Nuclear Power Plant) (in Russian abbreviations).

Table 5. Selected Generation III+ reactors (deployment in 5–10 years).

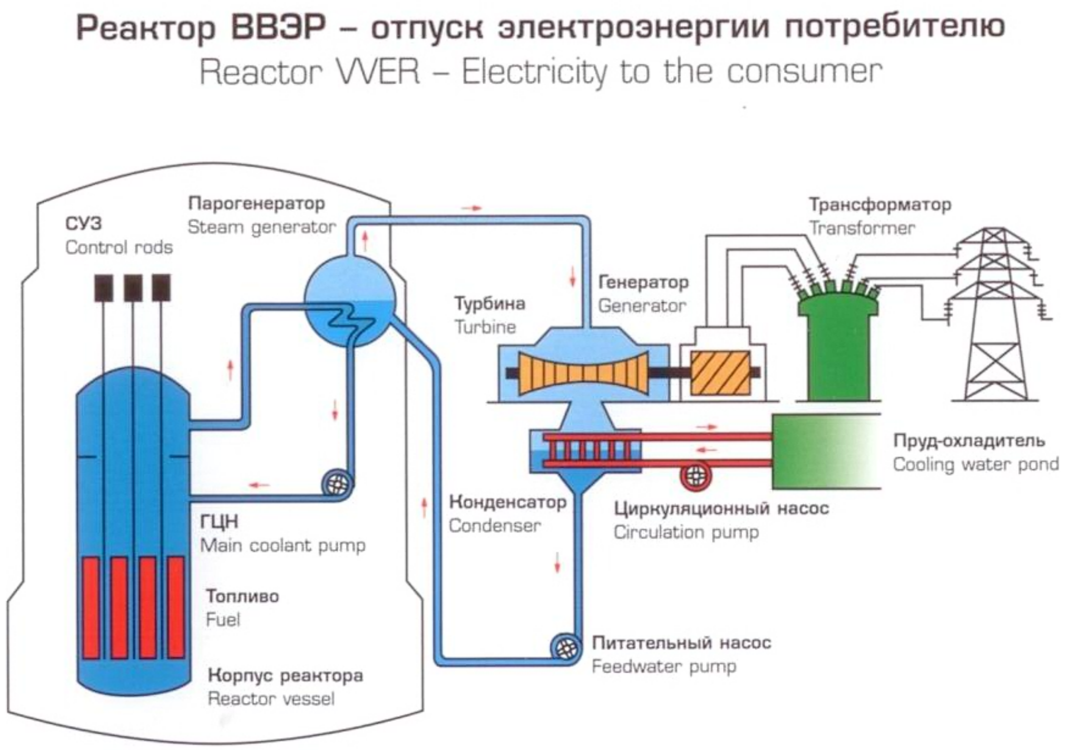

Figure 9. Scheme of typical Pressurized Water Reactor (PWR) (Russian VVER) NPP (ROSENERGOATOM, 2004) (courtesy of ROSENERGOATOM): General basic features – 1) thermal neutron spectrum; 2) uranium-dioxide (UO_2) fuel; 3) fuel enrichment about 4%; 4) indirect cycle with steam generator (also, a pressurizer required (not shown)), i.e., double flow circuit (double loop); 5) Reactor Pressure Vessel (RPV) with vertical fuel rods (elements) assembled in bundle strings cooled with upward flow of light water; 6) reactor coolant and moderator are the same fluid; 7) reactor coolant outlet parameters: Pressure 15 – 16 MPa (T_{sat} = 342 – 347°C) and temperatures inlet / outlet 290 – 325°C; and 8) power cycle - subcritical-pressure regenerative Rankine steam-turbine cycle with steam reheat[4] (working fluid - light water, turbine steam inlet parameters: Saturation pressure of 6 – 7 MPa and saturation temperature of 276 – 286°C).

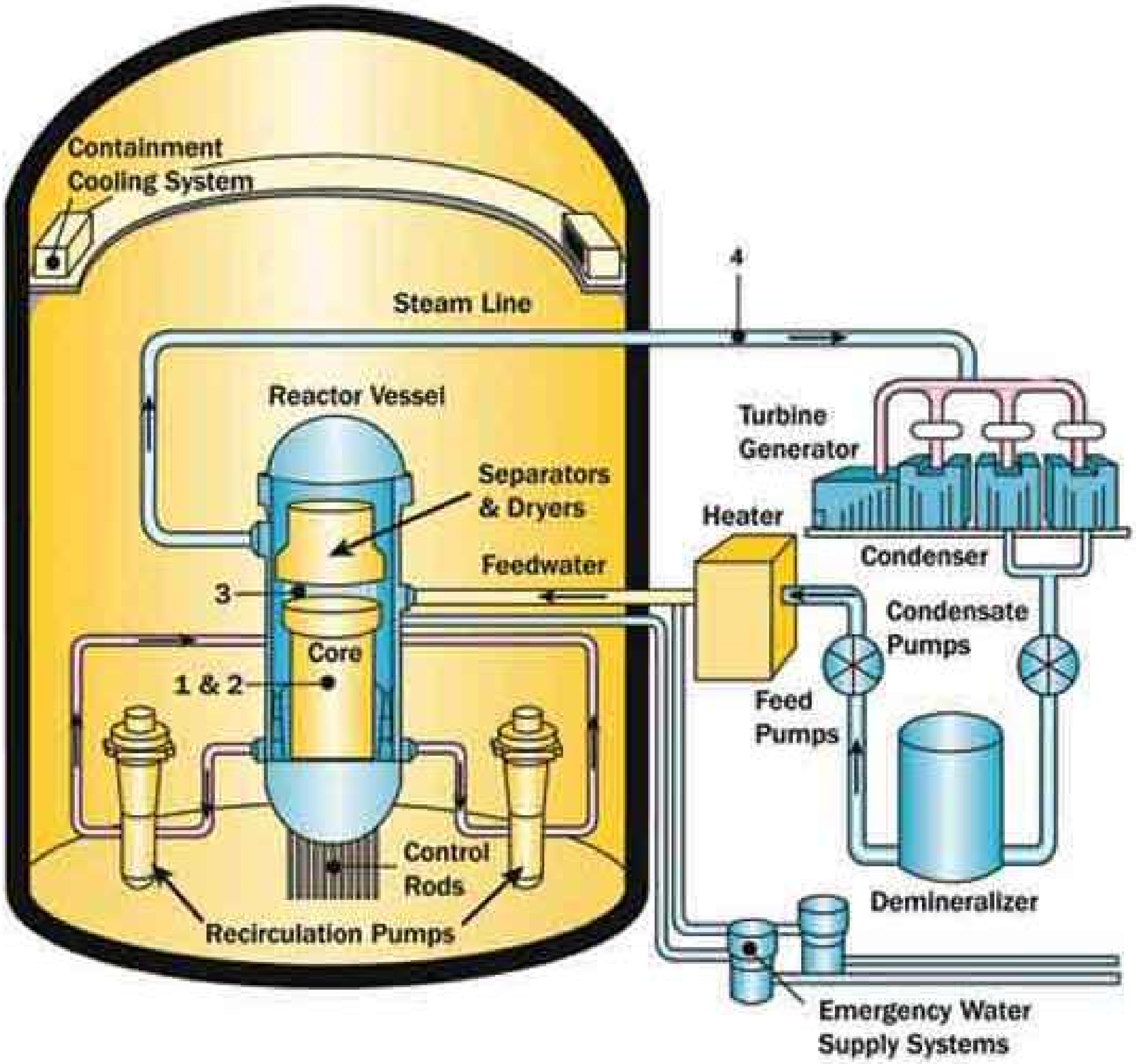

Figure 10. Scheme of typical Boiling Water Reactor (BWR) NPP (courtesy of NRC USA): General basic features – 1) thermal neutron spectrum; 2) uranium-dioxide (UO_2) fuel; 3) fuel enrichment about 3%; 4) direct cycle with steam separator (steam generator and pressurizer are eliminated), i.e., single-flow circuit (single loop); 5) RPV with vertical fuel rods (elements) assembled in bundle strings cooled with upward flow of light water (water and water-steam mixture); 6) reactor coolant, moderator and power-cycle working fluid are the same fluid; 7) reactor coolant outlet parameters: Pressure about 7 MPa and saturation temperature at this pressure is about 286°C; and 8) power cycle - subcritical-pressure regenerative Rankine steam-turbine cycle with steam reheat.

4 For the reheat the primary steam is used. Therefore, the reheat temperature is lower than the primary steam temperature. In general, the reheat parameters at NPPs are significantly lower than those at thermal power plants.

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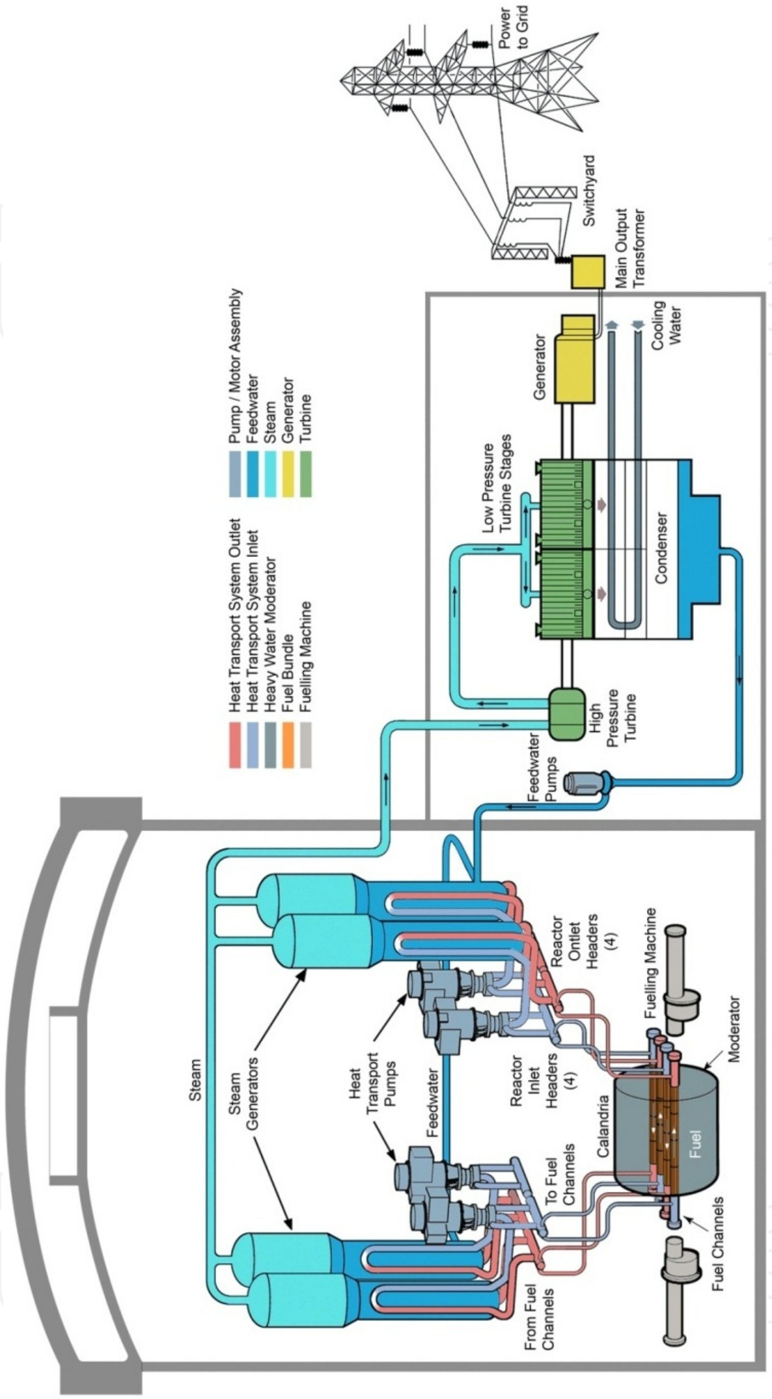

Figure 11. Scheme of CANDU-6 reactor (PHWR) NPP (courtesy of AECL): General basic features – 1) thermal-neutron spectrum; 2) natural uranium-dioxide (UO_2) fuel; 3) fuel enrichment about 0.7%; 4) indirect cycle with steam generator (also, a pressurizer required (not shown)), i.e., double-flow circuit (double loop); 5) pressure-channel design: Calandria vessel with horizontal fuel channels (see Figure 16c); 6) reactor coolant and moderator separated, but both are heavy water; 7) reactor coolant outlet parameters: Pressure about 9.9 MPa and temperature close to saturation (310°C); 8) on-line refuelling; and 9) power cycle - subcritical-pressure regenerative Rankine steam-turbine cycle with steam reheat (working fluid light water, turbine steam inlet parameters: Saturation pressure of ~4.6 MPa and saturation temperature of 259°C).

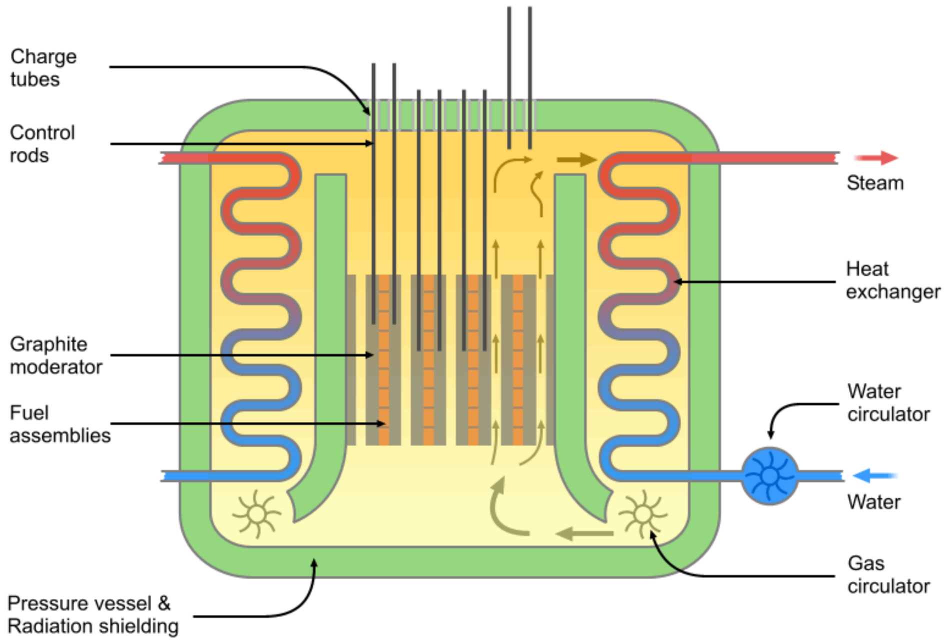

Figure 12. Scheme of Advanced Gas-cooled Reactor (AGR) (Wikimedia, 2012). Note that the heat exchanger is contained within the steel-reinforced concrete combined pressure vessel and radiation shield.

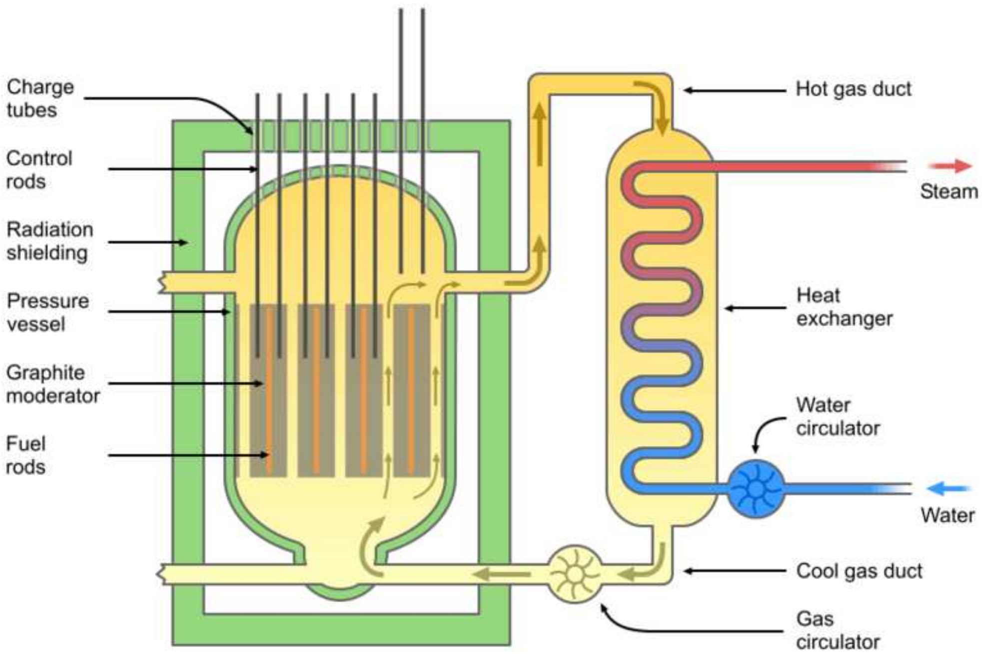

Figure 13. Scheme of Magnox nuclear reactor (GCR) showing gas flow (Wikipidea, 2012). Note that the heat exchanger is outside the concrete radiation shielding. This represents an early Magnox design with a cylindrical, steel, pressure vessel.

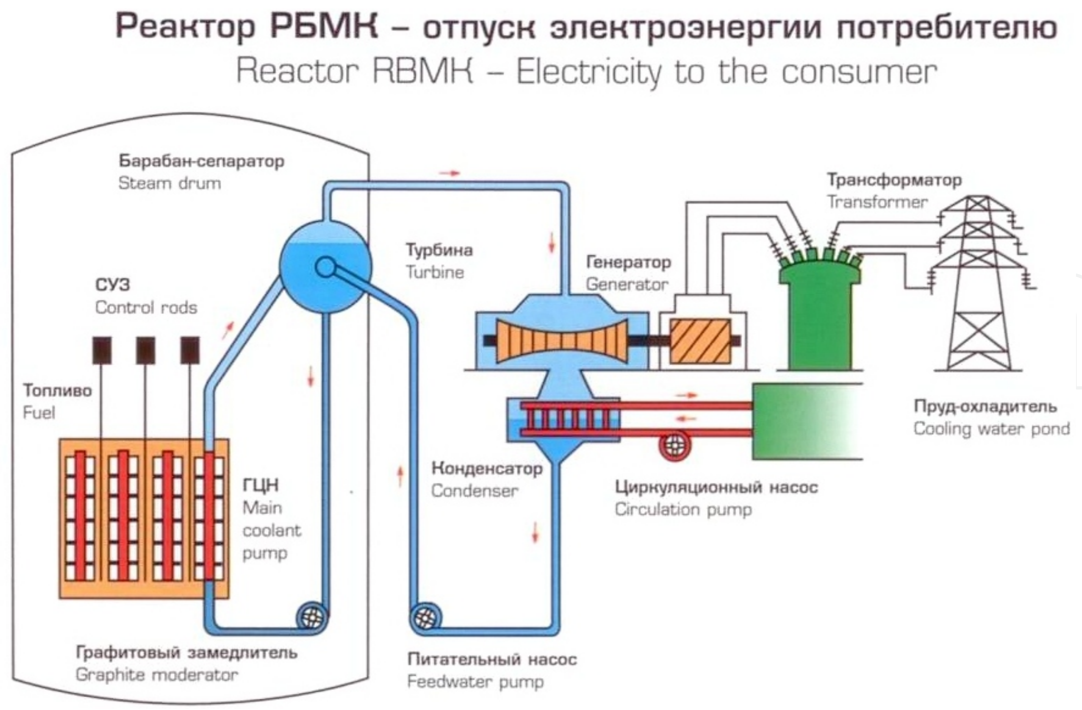

Figure 14. Scheme of Light-water Graphite-moderated Reactor (LGR) (Russian RBMK) NPP (ROSENERGOATOM, 2004) (courtesy of ROSENERGOATOM).

Figure 15. Scheme of Liquid-Metal Fast-Breeder Reactor (LMFBR) or SFR (Russian BN-600) NPP (ROSENERGOATOM, 2004) (courtesy of ROSENERGOATOM).

Parameter	VVER-440	VVER-1000 (Figure 9)	EGP-6	RBMK-1000 (Figure 14)	BN-600 (Figure 15)
Thermal power, MW_{th}	1375	3000	62	3200	1500
Electrical power, MW_{el}	440	1000	12	1000	600
Thermal efficiency, %	32.0	33.3	19.3	31.3	40.0
Coolant pressure, MPa	12.3	15.7	6.2	6.9	~0.1
Coolant massflow rate, t/s	11.3	23.6	0.17	13.3	6.9
Coolant inlet/outlet temperatures, °C	270/298	290/322	265	284	380/550
Steam massflow rate, t/s	0.75	1.6	0.026	1.56	0.18
Steam pressure, MPa	4.3	5.9	6.5	6.6	15.3
Steam temperature, °C	256	276	280	280	505
Reactor core: Diameter/Height m/m	3.8/11.8	4.5/10.9	4.2/3.0	11.8/7	2.1/0.75
Fuel enrichment, %	3.6	4.3	3.0;3.6	2.0-2.4	21;29.4
No. of fuel bundles	349	163	273	1580	369

Table 6. Major Parameters of Russian Power Reactors (Grigor'ev and Zorin, 1988).

Pressure Vessel (PV) ID, m	3.91
PV wall thickness, m	0.19
PV height without cover, m	10.8
Core equivalent diameter, m	2.88
Core height, m	2.5
Volume heat flux, MW/m^3	83
No. of fuel assemblies	349
No of rods per assembly	127
Fuel mass, ton	42
Part of fuel reloaded during year	1/3
Fuel	UO_2

Table 7. Additional parameters of VVER-1000.

Power	
Thermal output, MW_{th}	3830
Electrical output, MW_e	1330
Thermal efficiency, %	34
Specific power, kW/kg(U)	26
Power density, kW/L	56

Power	
Average linear heat flux, kW/m	20.7
Fuel-rod heat flux average/max, MW/m^2	0.51/1.12
Core	
Length, m	3.76
OD, m	4.8
Reactor-coolant system	
Pressure, MPa	7.17
Core massflow rate, kg/s	14,167
Core void fraction average/max	0.37 / 0.75
Feedwater inlet temperature, °C	216
Steam outlet temperature, °C	290
Steam outlet massflow rate, kg/s	2083
Reactor Pressure Vessel	
Inside Diameter, m	6.4
Height, m	22.1
Wall thickness, m	0.15
Fuel	
Fuel pellets	UO^2
Pellet OD, mm	10.6
Fuel rod OD, mm	12.5
Zircaloy sheath (cladding) thickness, mm	0.86

Table 8. Typical parameters of US BWR (Shultis and Faw, 2008).

Analysis of data listed in Table 3 shows that the vast majority nuclear reactors are water-cooled units. Only reactors built in UK are the gas-cooled type, and one reactor in Russia uses liquid sodium for its cooling.

UK carbon-dioxide-cooled reactors consist of two designs (Hewitt and Collier, 2000): 1) older design – Magnox reactor (GCR) (see Figure 13) and 2) newer design – AGR (see Figure 12). The Magnox design is a natural-uranium graphite-moderated reactor with the following parameters: Coolant – carbon dioxide; pressure - 2 MPa; outlet/inlet temperatures – 414/250°C; core diameter – about 14 m; height – about 8 m; magnesium-alloy sheath with fins; and thermal efficiency – about 32%. AGRs have the following parameters: Coolant – carbon dioxide; pressure - 4 MPa; outlet/inlet temperatures – 650/292°C; secondary-loop steam – 17 MPa and 560°C; stainless-steel sheath with ribs and hollow fuel pellets (see Figure 16b); enriched fuel 2.3%; and thermal efficiency – about 42% (the highest in nuclear-power industry so far). However, both these reactor designs will not be constructed anymore. They will just operate to the end of their life term and will be shut down. The same is applied to Russian RBMKs and EGPs.

Just for reference purposes, typical fuel elements (rods) / bundles of various reactors are shown in Figure 16, and typical sheath temperatures, heat transfer coefficients and heat fluxes are listed below.

Typical maximum sheath temperatures for steady operation (Hewitt and Collier, 2000)

BWR	300°C
PWR	320°C
Magnesium alloy (Magnox reactor)	450°C
AGR stainless steel	750°C
SFR	750°C

Typical heat-transfer-coefficient values for reactor coolants (Hewitt and Collier, 2000)

High-pressure carbon dioxide (forced convection)	1 kW/m²K
Boiling water in a kettle (nucleate pool boiling)	10 kW/m²K
Water (single-phase forced convection)	30 kW/m²K
PHWR	46 kW/m²K
Liquid sodium (forced convection)	55 kW/m²K
Boiling water (flow boiling)	60 kW/m²K

Typical Heat Fluxes (HFs) for steady operation (Hewitt and Collier, 2000)

	HF, MW/m²	T_{sheath}-T_{fluid}, °C
Boiling water in a kettle	0.15	15
CANDU reactor	0.625	14
BWR	1.0	15
PWR	1.5	50
SFR	2.0	35

Scheme 1. Typical maximum sheath temperatures for steady operation (Hewitt and Collier, 2000)

All current NPPs and oncoming Generation III+ NPPs are not very competitive with modern thermal power plants in terms of their thermal efficiency, a difference in values of thermal efficiencies between thermal and NPPs can be up to 20 – 30% (see Table 2). Therefore, new generation NPPs should be designed and built in the nearest future.

7. Next generation nuclear reactors

The demand for clean, non-fossil based electricity is growing; therefore, the world needs to develop new nuclear reactors with higher thermal efficiencies in order to increase electricity generation and decrease detrimental effects on the environment. The current fleet of NPPs is classified as Generation II and III (just a limited number of Generation III+ reactors (mainly, ABWRs) operates in some countries). However, all these designs (here we are talking about

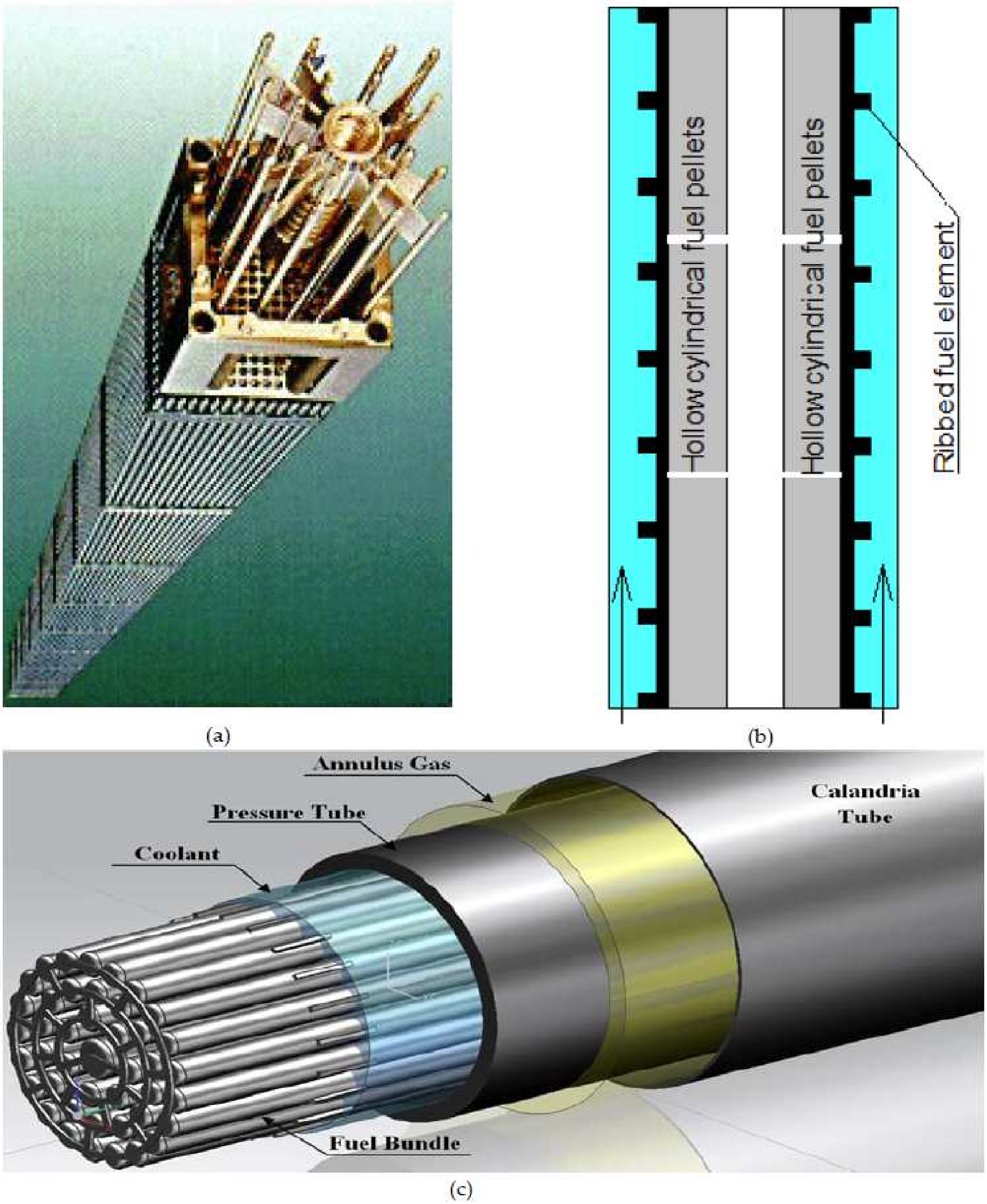

Figure 16. Typical PWR bundle string (courtesy of KAERI, http://www.nucleartourist.com/systems/pwrfuel1.htm) (a); AGR ribbed fuel element with hollow fuel pellet (Hewitt and Collier, 2000) (b); and CANDU reactor fuel channel (based on AECL design) (c).

only water-cooled power reactors) are not as energy efficient as they should be, because their operating temperatures are relatively low, i.e., below 350°C for a reactor coolant and even lower for steam.

Currently, a group of countries, including Canada, EU, Japan, Russia, USA and others have initiated an international collaboration to develop the next generation nuclear reactors (Generation IV reactors). The ultimate goal of developing such reactors is an increase in thermal efficiencies of NPPs from 30 – 36% to 45 - 50% and even higher. This increase in thermal efficiency would result in a higher production of electricity compared to current LWR technologies per 1 kg of uranium.

The Generation IV International Forum (GIF) Program has narrowed design options of nuclear reactors to six concepts. These concepts are: 1) Gas-cooled Fast Reactor (GFR) or just High Temperature Reactor (HTR), 2) Very High Temperature Reactor (VHTR), 3) Sodium-cooled Fast Reactor (SFR), 4) Lead-cooled Fast Reactor (LFR), 5) Molten Salt Reactor (MSR), and 6) SuperCritical Water-cooled Reactor (SCWR). Figures 17 – 24 show schematics of these concepts. These nuclear-reactor concepts differ one from each other in terms of their design, neutron spectrum, coolant, moderator, operating temperatures and pressures. A brief description of each Generation IV nuclear-reactor concept has been provided below.

Reactor Parameter	Unit	Reference Value
Reactor power	MW_{th}	600
Coolant inlet/outlet temperatures	°C	490/850
Pressure	MPa	9
Coolant massflow rate	kg/s	320
Average power density	MW_{th}/m^3	100
Reference fuel compound		UPuC/SiC (70/30%) with about 20% Pu
Net-plant efficiency	%	48

Table 9. Key-design parameters of Gas-cooled Fast Reactor (GFR) concept.

Gas-cooled Fast Reactor (GFR) or High Temperature Reactor (HTR) (see Figure 17 and Table 9.) is a fast-neutron-spectrum reactor, which can be used for the production of electricity and co-generation of hydrogen through thermochemical cycles or high-temperature electrolysis. The coolant is helium with inlet and outlet temperatures of 490 and 850°C, respectively. The net plant efficiency is about 48% with the direct Brayton helium-gas-turbine cycle. Table 9 lists a summary of design parameters for GFR (US DOE, 2002). However, due to some problems with implementation of the direct Brayton helium-gas-turbine cycle, the indirect Rankine steam cycle or even indirect supercritical carbon-dioxide Brayton gas-turbine cycle are also considered. The indirect cycles will be linked to the GFR through heat exchangers.

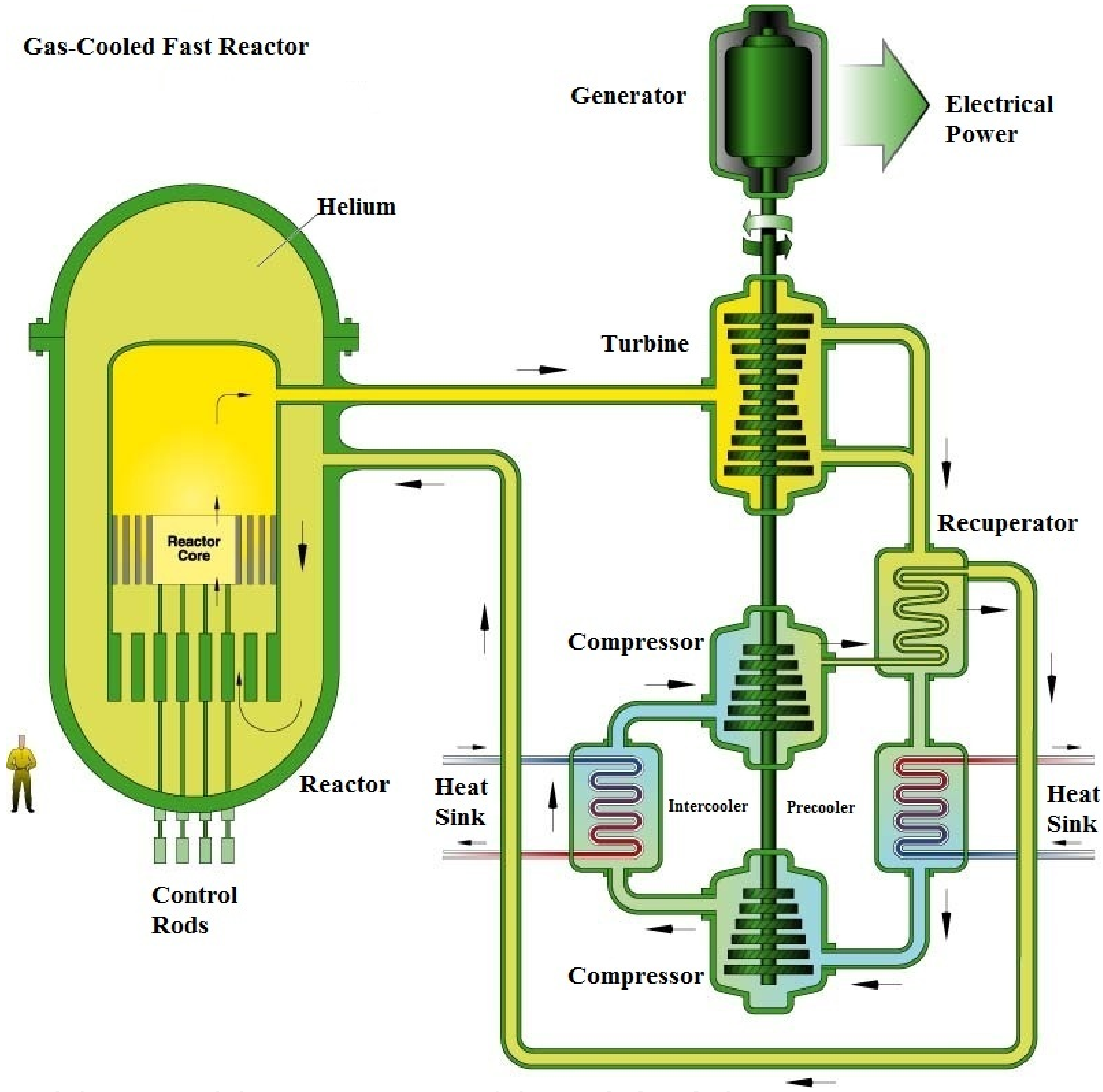

Figure 17. Scheme of Gas-cooled Fast Reactor (GFR) NPP concept (US DOE).

Very High Temperature Reactor (VHTR) (see Figure 18) is a thermal-neutron-spectrum reactor. The ultimate purpose of this nuclear-reactor design is the co-generation of hydrogen through high-temperature electrolysis. In a VHTR, graphite and helium have been chosen as the moderator and the coolant, respectively. The inlet and outlet temperatures of the coolant are 640 and 1000°C, respectively, at a pressure of 7 MPa (US DOE, 2002). Due to such high outlet temperatures, the thermal efficiency of VHTR will be above 50%. A summary of design parameters of VHTR are listed in Table 10 (US DOE, 2002).

In general, the US DOE supports research on several Generation IV reactor concepts (http://nuclear.energy.gov/genIV/neGenIV4.html). However, the priority is being given to the VHTR, as a system compatible with advanced electricity production, hydrogen co-generation and high-temperature process-heat applications.

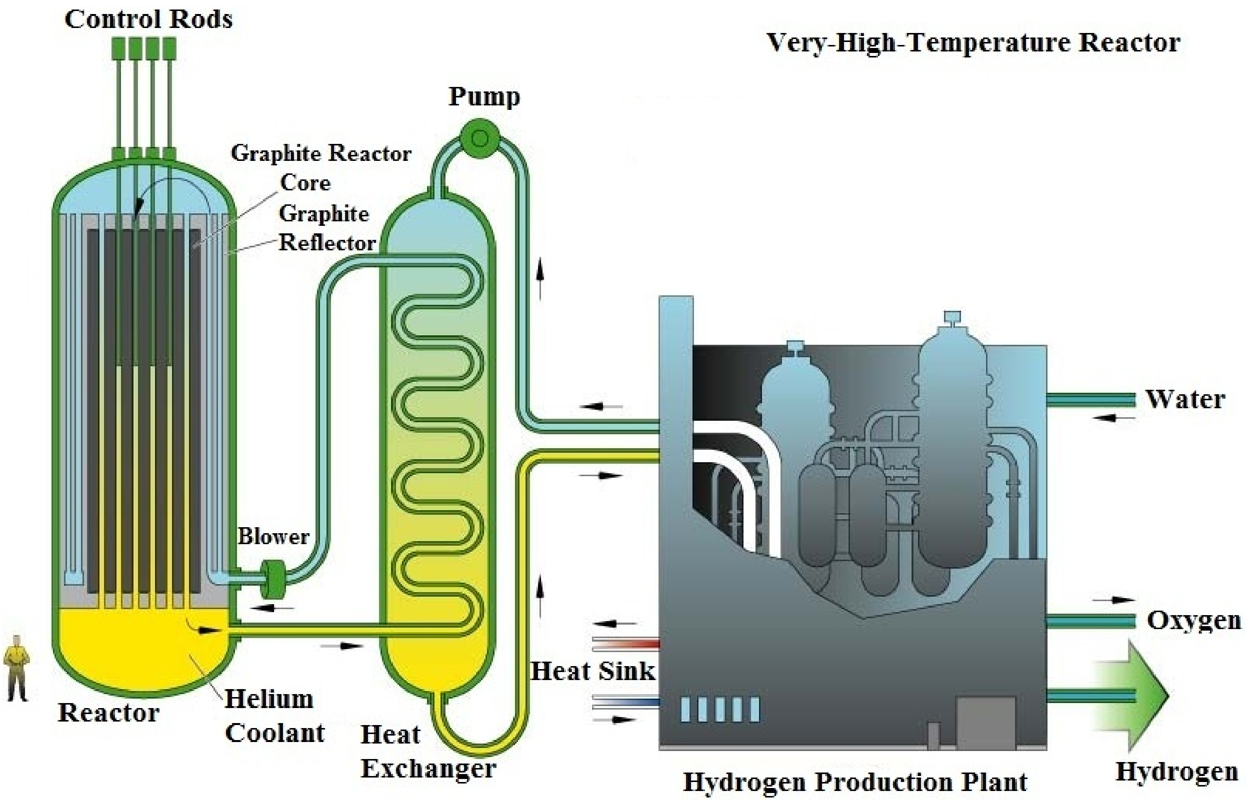

Figure 18. Scheme of Very High Temperature Reactor (VHTR) plant concept with co-generation of hydrogen (US DOE).

Reactor Parameter	Unit	Reference Value
Reactor power	MW_{th}	600
Average power density	MW_{th}/m^3	610
Coolant inlet/outlet temperatures	°C	640/1000
Coolant/Massflow rate	kg/s	Helium/320
Reference fuel compound		ZrC-coated particles in pins or pebbles
Net-plant efficiency	%	>50

Table 10. Key-design parameters of Very High Temperature Reactor (VHTR) concept.

Similar to GFR, SFR (see Figure 19) is a fast-neutron-spectrum reactor. The main objectives of SFR are the management of high-level radioactive wastes and production of electricity. SFR uses liquid sodium as a reactor coolant with an outlet temperature between 530 and 550°C at the atmospheric pressure. The primary choices of fuel for SFR are oxide and metallic fuels.

Table 11 lists a summary of design parameters of SFR (US DOE, 2002). The SFR concept is also on the priority list for the US DOE (http://nuclear.energy.gov/genIV/neGenIV4.html).

Currently, SFR is the only one Generation IV concept implemented in the power industry. Russia and Japan are leaders within this area. In particularly, Russia operates SFR at the Beloyarsk NPP (for details, see BN-600 in Table 6) and constructs even more powerful SFR – BN-850. Japan has operated SFR at the Monju NPP some time ago (http://en.wikipedia.org/wiki/Monju_Nuclear_Power_Plant). In Russia and Japan the SFRs are connected to the subcritical-pressure Rankine steam cycle through heat exchangers (see Figure 19). However, in the US and some other countries a supercritical carbon-dioxide Brayton gas-turbine cycle is considered as the power cycle for future SFRs, because carbon dioxide and sodium are considered to be more compatible than water and sodium. In general, sodium is highly reactive metal. It reacts with water evolving hydrogen gas and releasing heat. Due to that sodium can ignite spontaneously with water. Also, it can ignite and burn in air at high temperatures. Therefore, special precautions should be taken for safe operation of this type reactor. One of them is the triple-flow circuit with the intermediate sodium loop between the reactor coolant (primary sodium) and water as the working fluid in the power cycle.

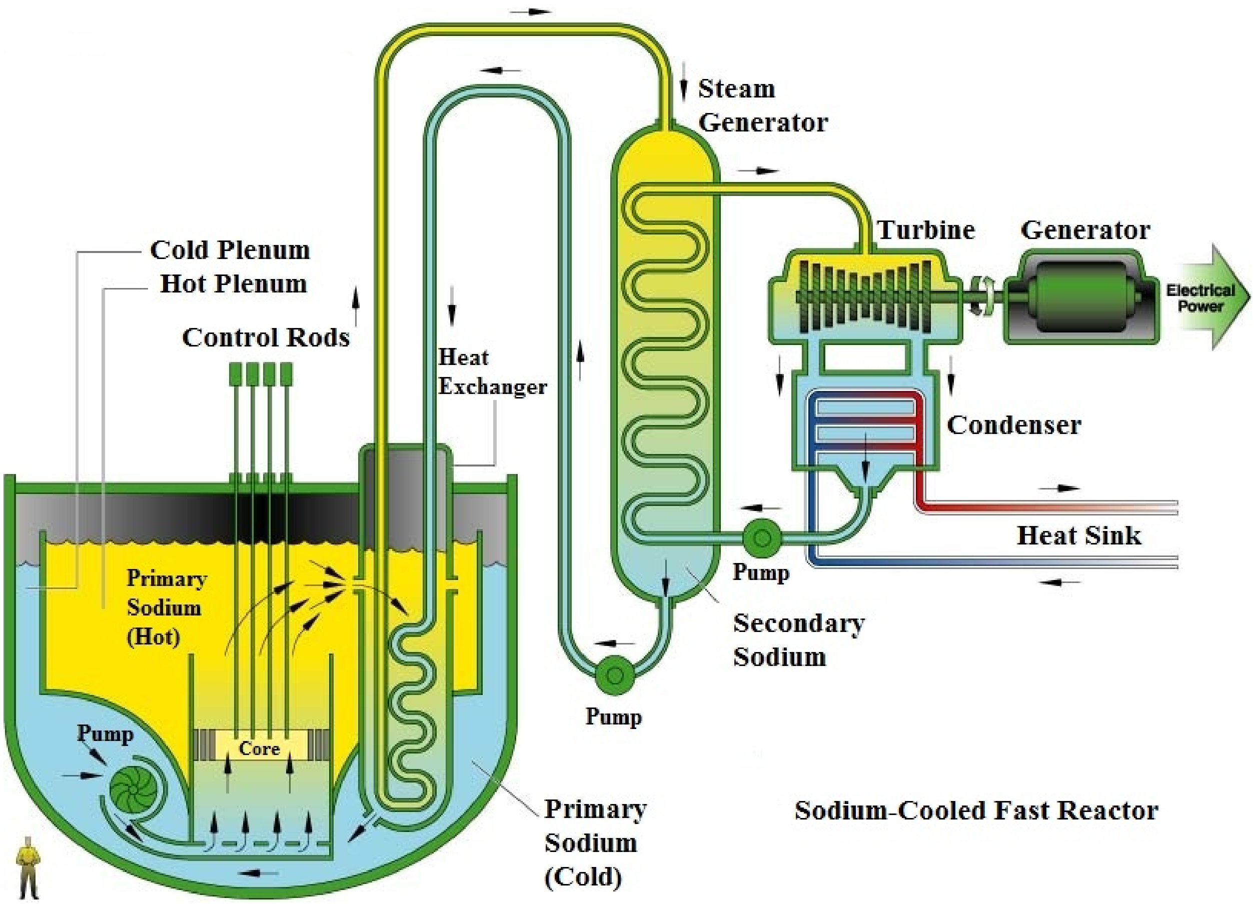

Figure 19. Scheme of Sodium Fast Reactor (SFR) NPP concept (US DOE, 2002).

Reactor Parameter	Unit	Reference Value
Reactor power	MW_{th}	1000–5000
Thermal efficiency	%	40–42%
Coolant		Sodium
Coolant melting/boiling temperatures	°C	98/883
Coolant density at 450°C	kg/m^3	844
Pressure inside reactor	MPa	~0.1
Coolant maximum outlet temperature	°C	530–550
Average power density	MW_{th}/m^3	350
Reference fuel compound		Oxide or metal alloy
Cladding		Ferritic or ODS ferritic
Average burnup	GWD/MTHM	~150–200

Table 11. Key-design parameters of SFR concept (also, see Table 6 for parameters of currently operating SFR BN-600).

Reactor Parameter	Unit	Brest-300	Brest-1200
Reactor power (thermal/electrical)	MW	700/300	2800/1200
Thermal efficiency	%	43	
Primary coolant		Lead	
Coolant melting/boiling temperatures	°C	328/1743	
Coolant density at 450°C	kg/m^3	10,520	
Pressure inside reactor	MPa	~0.1	
Coolant inlet/outlet temperatures	°C	420/540	
Coolant massflow rate	t/s	40	158
Maximum coolant velocity	m/s	1.8	1.7
Fuel		UN+PuN	
Fuel loading	t	16	64
Term of fuel inside reactor	years	5	5–6
Fuel reloading per year		1	
Core diameter/height	m / m	2.3/1.1	4.8/1.1
Number of fuel bundles		185	332
Fuel-rod diameter	mm	9.1; 9.6; 10.4	
Fuel-rod pitch	mm	13.6	
Maximum cladding temperature	°C	650	
Steam-generator pressure	MPa	24.5	
Steam-generator inlet/outlet temperatures	°C	340/520	
Steam-generator capacity	t/s	0.43	1.72
Term of reactor	years	30	60

Table 12. Key-design parameters of LFRs planned to be built in Russia (based on NIKIET data).

LFR (see Figure 20) is a fast-neutron-spectrum reactor, which uses lead or lead-bismuth as the reactor coolant. The outlet temperature of the coolant is about 550°C (but can be as high as 800°C) at an atmospheric pressure. The primary choice of fuel is a nitride fuel. The supercritical carbon-dioxide Brayton gas-turbine cycle has been chosen as a primary choice for the power cycle in US and some other countries, while the supercritical-steam Rankine cycle is considered as the primary choice in Russia (see Table 12).

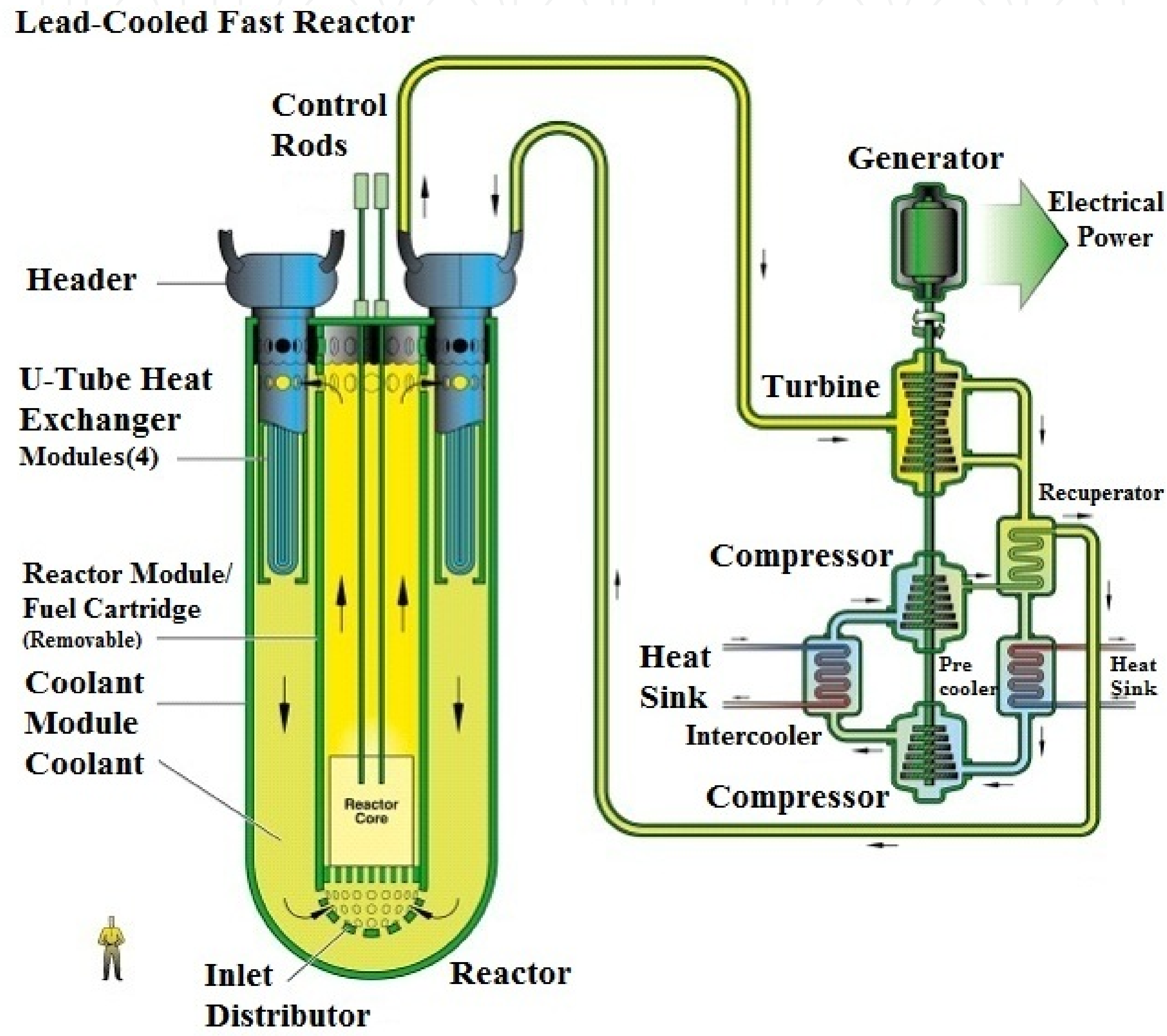

Figure 20. Scheme of Lead Fast Reactor (LFR) NPP concept (US DOE).

MSR (see Figure 21) is a thermal-neutron-spectrum reactor, which uses a molten fluoride salt with dissolved uranium while the moderator is made of graphite. The inlet temperature of the coolant (e.g., fuel-salt mixture) is 565°C while the outlet temperature reaches 700°C. However, the outlet temperature of the fuel-salt mixture can even increase to 850°C when co-generation of hydrogen is considered as an option. The thermal efficiency of the plant is between 45 and 50%. Table 13 lists the design parameters of MSR (US DOE, 2002).

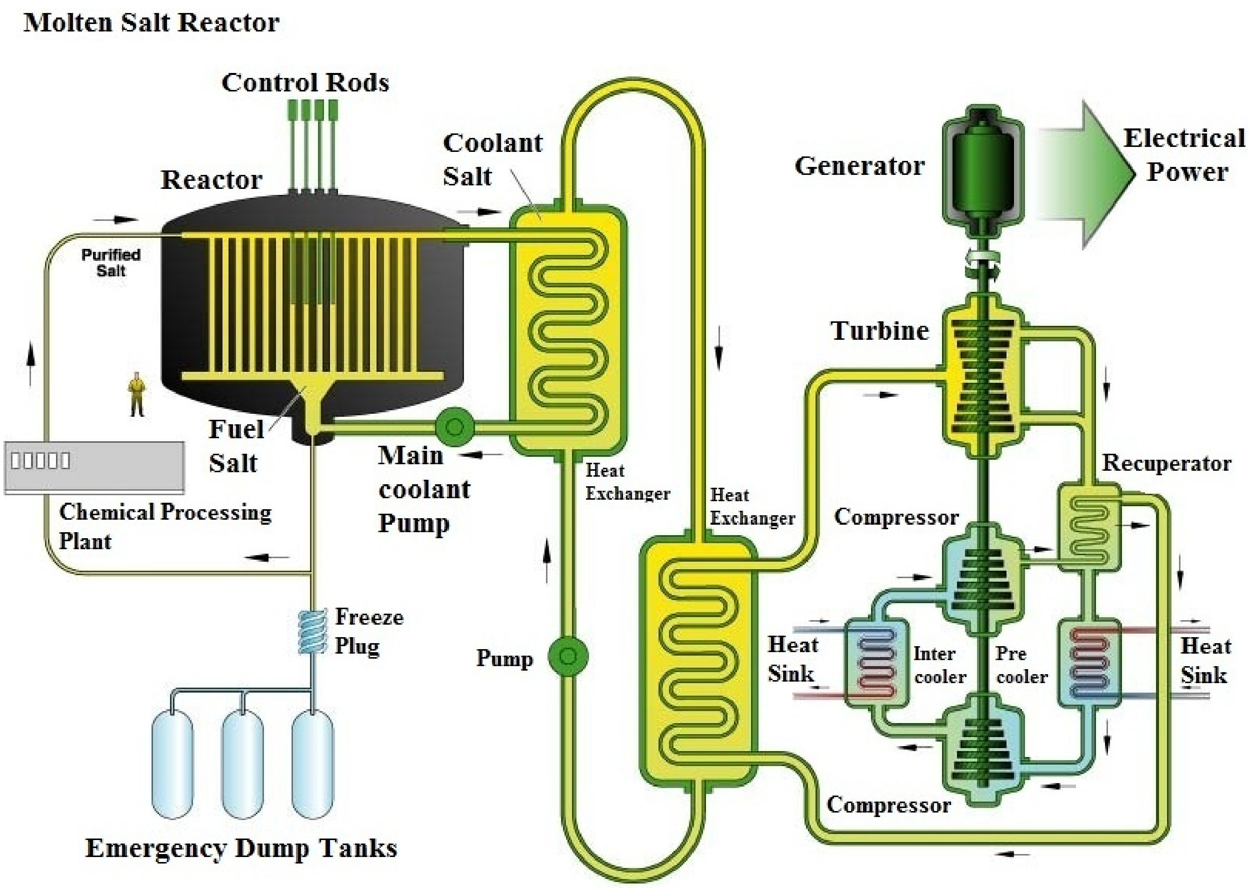

Figure 21. Scheme of Molten Salt Reactor (MSR) NPP concept (US DOE, 2002).

Reactor Parameter	Unit	Reference Value
Reactor power	MW_{el}	1000
Net thermal efficiency	%	4450
Average power density	MW_{th}/m^3	22
Fuel-salt inlet/outlet temperatures	°C	565/700 (800)
Moderator		Graphite
Neutron-spectrum burner		Thermal-Actinide

Table 13. Key-design parameters of MSR concept.

The design of SCWRs is seen as the natural and ultimate evolution of today's conventional water-cooled nuclear reactors (Schulenberg and Starflinger, 2012; Pioro, 2011; Oka et al., 2010; Pioro and Duffey, 2007):

1. Modern PWRs operate at pressures of 15 – 16 MPa.

2. BWRs are the once-through or direct-cycle design, i.e., steam from a nuclear reactor is forwarded directly into a turbine.

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3. Some experimental reactors used nuclear steam reheat with outlet steam temperatures well beyond the critical temperature, but at pressures below the critical pressure (Saltanov and Pioro, 2011). And

4. Modern supercritical-pressure turbines, at pressures of about 25 MPa and inlet temperatures of about 600°C, operate successfully at coal-fired thermal power plants for more than 50 years.

Parameters	Unit	PV SCWR Concepts		
Country	**–**	**Russia**		**USA**
Spectrum	–	Thermal	Fast	Thermal
Power electrical	MW	1500	1700	1600
Thermal efficiency	%	34	44	45
Pressure	MPa	25	25	25
Coolant inlet/outlet temperatures	ºC	280/550	280/530	280/500
Massflow rate	kg/s	1600	1860	1840
Core height/diameter	m/m	3.5/2.9	4.1/3.4	4.9/3.9
Fuel	–	UO_2	MOX	UO_2
Enrichment	$\%_{wt}$	–	–	5
Maximum cladding temperature	ºC	630	630	–
Moderator	–	H_2O	–	H_2O

Table 14. Modern concepts of Pressure-Vessel Super Critical Water-cooled Reactors (PV SCWRs) (Pioro and Duffey, 2007).

In general, SCWRs can be classified based on a pressure boundary, neutron spectrum and/or moderator (Pioro and Duffey, 2007). In terms of the pressure boundary, SCWRs are classified into two categories, a) Pressure Vessel (PV) SCWRs (see Figure 22), and b) Pressure Tube (PT) or Pressure Channel (PCh) SCWRs (see Figures 23 and 24). The PV SCWR requires a pressure vessel with a wall thickness of about 50 cm in order to withstand supercritical pressures. Figure 22 shows a scheme of a PV SCWR NPP. Table 14 lists general operating parameters of modern PV-SCWR concepts. On the other hand, the core of a PT SCWR consists of distributed pressure channels, with a thickness of about 10 mm, which might be oriented vertically or horizontally, analogous to CANDU and RBMK reactors, respectively. For instance, SCW CANDU reactor (Figure 23) consists of 300 horizontal fuel channels with coolant inlet and outlet temperatures of 350 and 625°C at a pressure of 25 MPa (Pioro and Duffey, 2007). It should be noted that a vertical-core option (Figure 24) has not been ruled out; both horizontal and vertical cores are being studied by the Atomic Energy of Canada Limited (AECL). Table 15 provides information about modern concepts of PT SCWR.

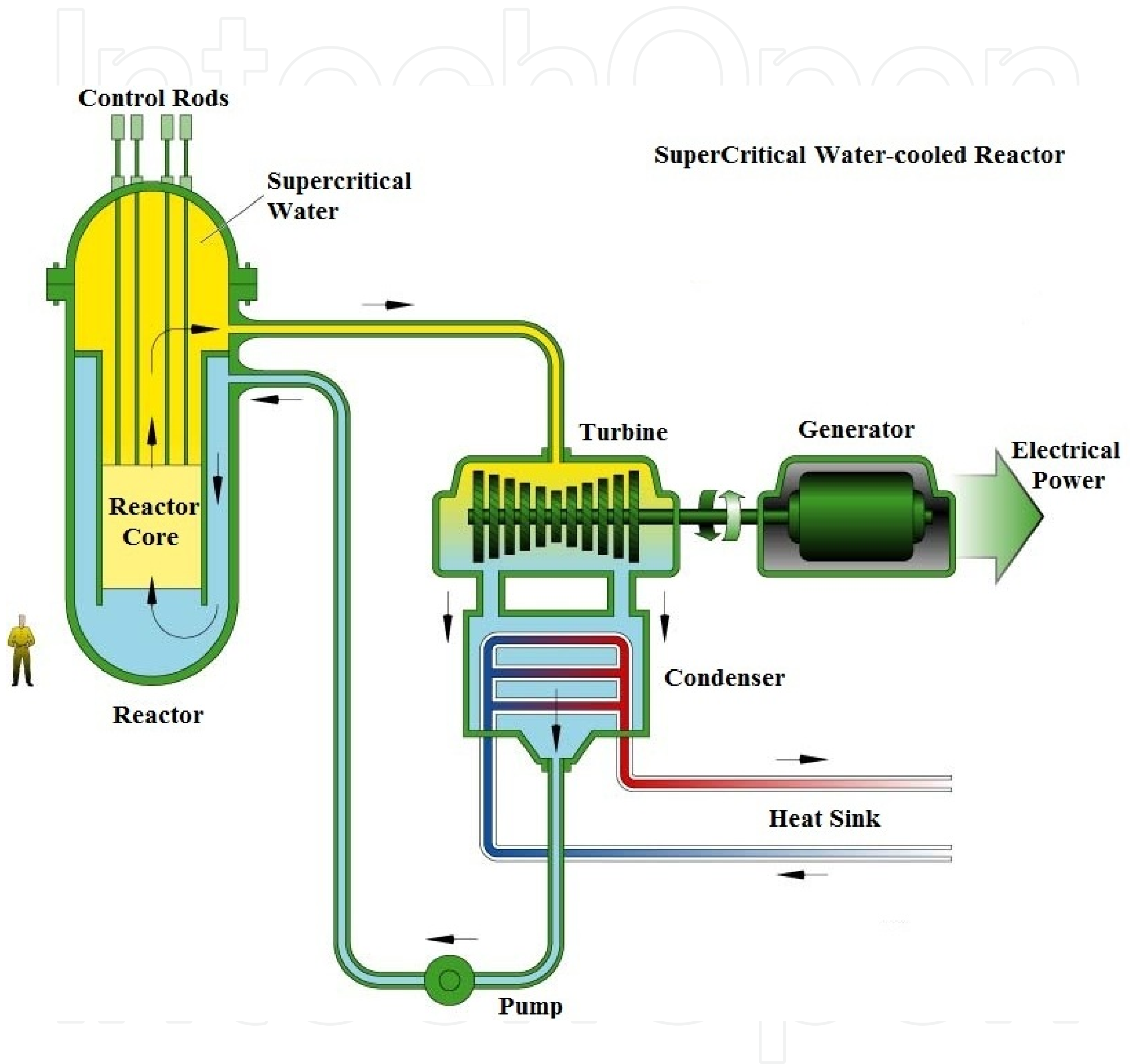

Figure 22. Schematic of Pressure-Vessel Super Critical Water-cooled Reactor (PV SCWR) NPP concept (US DOE, 2002).

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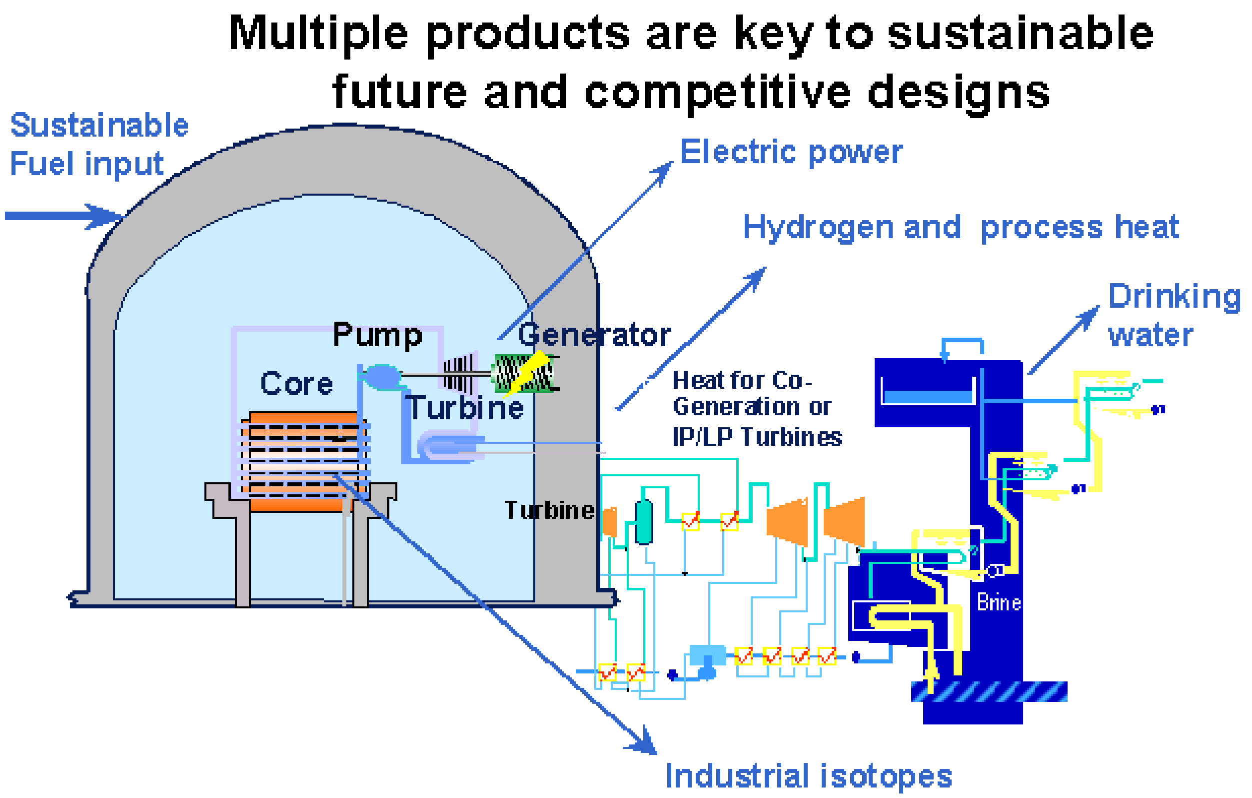

Figure 23. General scheme of Pressure-Tube (PT) SCW-CANDU-reactor NPP concept (courtesy of Dr. R. Duffey, AECL).

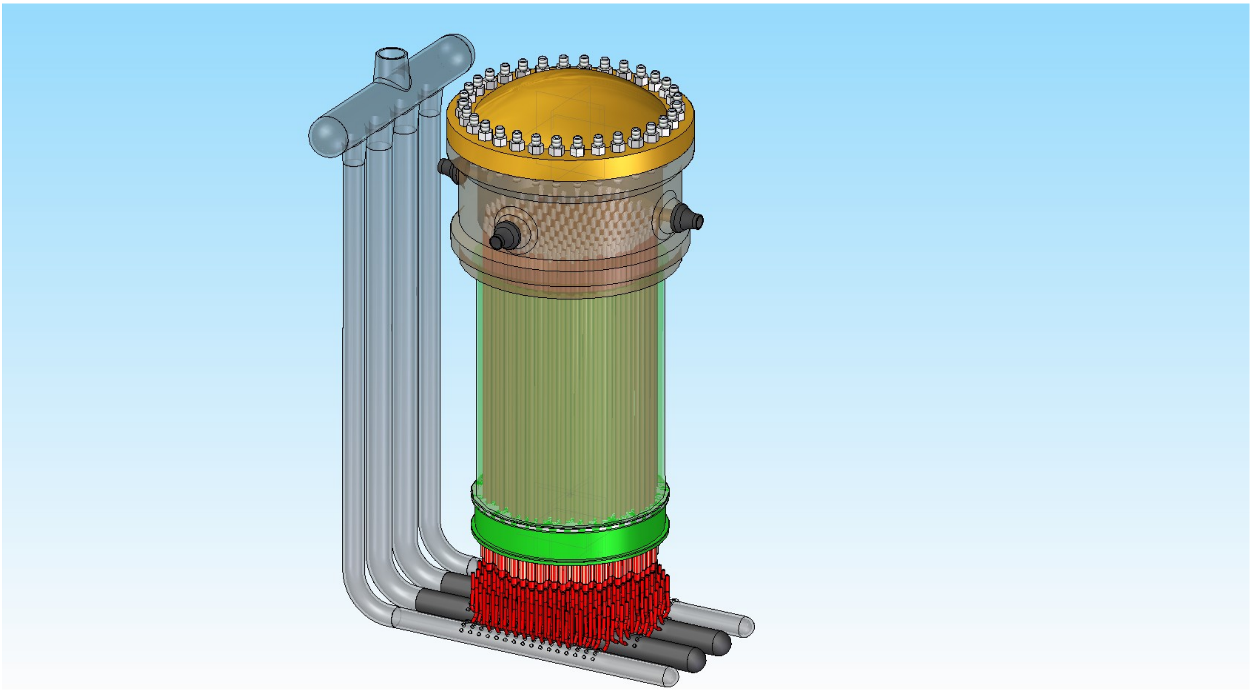

Figure 24. Vertical core-configuration option of PT SCW-CANDU-reactor concept (courtesy of AECL).

Parameters	Unit	PT-SCWR concepts			
Country	**–**	**Canada (Figures 23 and 24)**	**Russia (NIKIET)**		
Spectrum	–	Thermal	Thermal	Fast	Thermal
Power electrical	MW_{el}	1220	1200	1200	850
Thermal efficiency	%	48	44	43	42
Coolant pressure	MPa	25	24,5	25	25
Coolant temperature	ºC	350–625	270–545	400–550	270–545
Mas flowrate	kg/s	1320	1020	–	922
Core height/diameter	m/m	/7	6/12	3.5/11	5/6.5
Fuel	–	UO_2/Th	UCG	MOX	UO_2
Enrichment	$\%_{wt}$	4	4,4	–	6
Maximum cladding temperature	ºC	850	630	650	700
Moderator	–	D_2O	Graphite	–	D_2O

Table 15. Modern concepts of PT SCWRs (Pioro and Duffey, 2007).

In terms of the neutron spectrum, most SCWR designs are a thermal spectrum; however, fast-spectrum SCWR designs are possible (Oka et al., 2010). In general, various liquid or solid moderator options can be utilized in thermal-spectrum SCWRs. These options include light-water, heavy-water, graphite, beryllium oxide, and zirconium hydride. The liquid-moderator concept can be used in both PV and PT SCWRs. The only difference is that in a PV SCWR, the moderator and coolant are the same fluid. Thus, light-water is a practical choice for the moderator. In contrast, in PT SCWRs the moderator and coolant are separated. As a result, there are a variety of options in PT SCWRs.

One of these options is to use a liquid moderator such as heavy-water. One of the advantages of using a liquid moderator in PT SCWRs is that the moderator acts as a passive heat sink in the event of a Loss Of Coolant Accident (LOCA). A liquid moderator provides an additional safety feature[5], which enhances the safety of operation. On the other hand, one disadvantage of liquid moderators is an increased heat loss from the fuel channels to the liquid moderator, especially, at SCWR conditions.

The second option is to use a solid moderator. Currently, in RBMK reactors and some other types of reactors such as Magnox, AGR, and HTR, graphite is used as a moderator. However, graphite may catch fire at high temperatures at some conditions. Therefore, other materials such as beryllium, beryllium oxide and zirconium hydride may be used as solid moderators.

5 Currently, such option is used in CANDU-6 reactors.

In this case, heat losses can be reduced significantly. On the contrary, the solid moderators do not act as a passive-safety feature.

High operating temperatures in SCWRs lead to high fuel centreline temperatures. Currently, UO_2 has been used in LWRs, PHWRs, etc. However, the uranium-dioxide fuel has a lower thermal conductivity, which results in high fuel centerline temperatures. Therefore, alternative fuels with high thermal-conductivities such as UO_2-BeO, UO_2-SiC, UO_2 with graphite fibre, UC, UC_2, and UN might be used (Peiman et al., 2012).

However, the major problem for SCWRs development is reliability of materials at high pressures and temperatures, high neutron flux and aggressive medium such as supercritical water. Unfortunately, up till now nobody has tested candidate materials at such severe conditions.

8. Conclusions

1. Major sources for electrical-energy production in the world are: 1) thermal - primary coal and secondary natural gas; 2) nuclear and 3) hydro.
2. In general, the major driving force for all advances in thermal and nuclear power plants is thermal efficiency. Ranges of gross thermal efficiencies of modern power plants are as the following: 1) Combined-cycle thermal power plants – up to 62%; 2) Supercritical-pressure coal-fired thermal power plants – up to 55%; 3) Carbon-dioxide-cooled reactor NPPs – up to 42%; 4) Sodium-cooled fast reactor NPP – up to 40%; 5) Subcritical-pressure coal-fired thermal power plants – up to 38%; and 6) Modern water-cooled reactors – 30 – 36%.
3. In spite of advances in coal-fired thermal power-plants design and operation worldwide they are still considered as not environmental friendly due to producing a lot of carbon-dioxide emissions as a result of combustion process plus ash, slag and even acid rains.
4. Combined-cycle thermal power plants with natural-gas fuel are considered as relatively clean fossil-fuel-fired plants compared to coal and oil power plants, but still emits a lot of carbon dioxide due to combustion process.
5. Nuclear power is, in general, a non-renewable source as the fossil fuels, but nuclear resources can be used significantly longer than some fossil fuels plus nuclear power does not emit carbon dioxide into atmosphere. Currently, this source of energy is considered as the most viable one for electrical generation for the next 50 – 100 years.
6. However, all current and oncoming Generation III+ NPPs are not very competitive with modern thermal power plants in terms of thermal efficiency, the difference in values of thermal efficiencies between thermal and nuclear power plants can be up to 20 – 25%.
7. Therefore, new generation (Generation IV) NPPs with thermal efficiencies close to those of modern thermal power plants, i.e., within a range of 45 – 50% at least, should be designed and built in the nearest future.

Nomenclature

P, *p* pressure, Pa

H specific enthalpy, J/kg

m massflow rate, kg/s

T temperature, °C

V specific volume, m^3/kg

Greek letters

ρ density, kg/m^3

Subscripts

cr critical

el elctrical

in inlet

pc pseudocritical

s, sat saturation

th thermal

Abbreviations

ABWR Advanced Boiling Water Reactor

AECL Atomic Energy of Canada Limited

AGR Advanced Gas-cooled Reactor

BN Fast Neutrons (reactor) (in Russian abbreviation)

BWR Boiling Water Reactor

CANDU CANada Deuterium Uranium

DOE Department Of Energy (USA)

EGP Power Heterogeneous Loop reactor (in Russian abbreviations)

EU European Union

GCR Gas-Cooled Reactor

GFR Gas Fast Reactor

http://dx.doi.org/10.5772/51916

HP High Pressure

HTR High Temperature Reactor

ID Inside Diameter

IP Intermediate Pressure

KAERI Korea Atomic Energy Research Institute (South Korea)

LFR Lead-cooled Fast Reactor

LGR Light-water Graphite-moderated Reactor

LMFBR Liquid-Metal Fast-Breeder Reactor

LP Low Pressure

LWR Light-Water Reactor

MOX Mixed OXides

NIKIET Research and Development Institute of Power Engineering (in Russian abbreviations) or RDIPE, Moscow, Russia

NIST National Institute of Standards and Technology (USA)

NPP Nuclear Power Plant

NRC National Regulatory Commission (USA)

NRU National Research Universal (reactor), AECL, Canada

PCh Pressure Channel

PHWR Pressurized Heavy-Water Reactor

PT Pressure Tube

PV Pressure Vessel

PWR Pressurized Water Reactor

RBMK Reactor of Large Capacity Channel type (in Russian abbreviations)

RPV Reactor Pressure Vessel

SC SuperCritical

SCW SuperCritical Water

SCWR SuperCritical Water Reactor

SFR Sodium Fast Reactor

UK United Kingdom

USA United States of America

VHTR Very High Temperature Reactor

VVER Water-Water Power Reactor (in Russian abbreviation)

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