УДК 620.9

Pioro I.
University of Ontario Institute of Technology Faculty of Energy Systems and Nuclear Science.
Canada, Oshawa, Ontario

NUCLEAR POWER AS A BASIS FOR FUTURE ELECTRICITY PRODUCTION IN THE WORLD: PART 1. GENERATION III AND IV REACTORS

Анотація

У першій частині статті розглядаються різні засоби генерації енергії, технічні і конструктивні особливості, характерні для атомних реакторів III і IV поколінь, даються порівняльні оцінки важливих параметрів їх характеристик.

Abstract

In Part 1 of the paper various methods of electrical-energy generating are listed together with technical and design specifics of nuclear power plants with Generation III and IV power reactors.

Introduction

It is well known that the electrical-power generation is the key 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 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 and solar are not really reliable sources for industrial power generation, because they depend on Mother nature plus 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.

Table 1
Electrical energy consumption per capita in selected countries
(Wikipedia, http://en.wikipedia.org/wiki/List_of_countries_by_electric_energy_consumption)

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: Very high — 42 countries; high — 43; medium — 42; and low — 42

Thermal Power Plants
Coal-fired power plants

For thousands years, mankind used and still is using wood and coal for heating purposes. For about

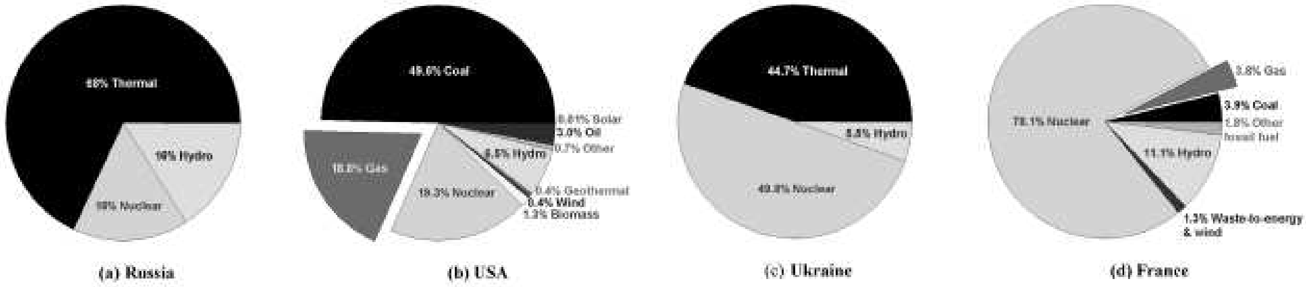
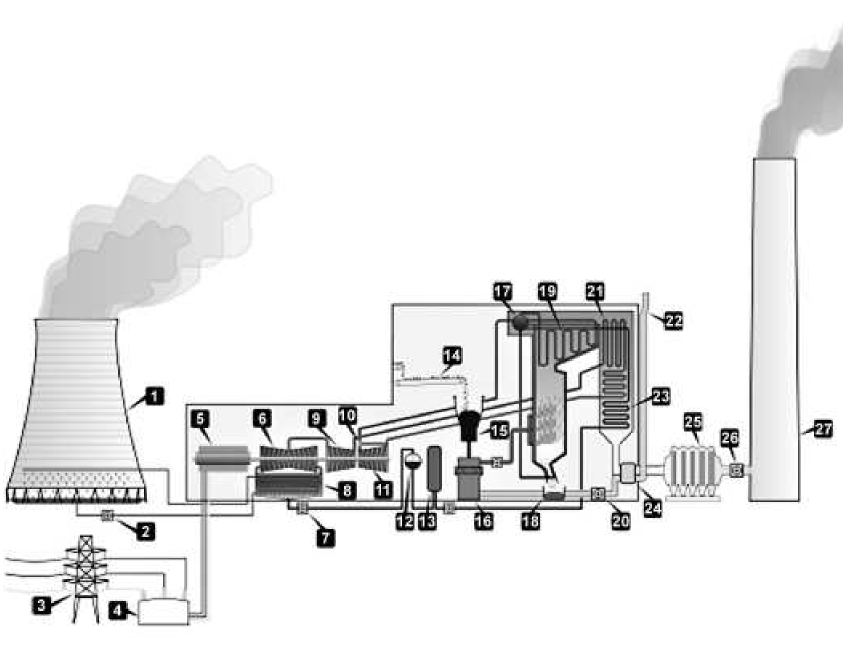


Fig. 1. Electricity production by source in selected countries (data from 2005–2010 presented here just for reference purposes) (Wikipedia)



Key to Diagram

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.
27. Flue gas stack

Fig. 2. Typical diagram of coal-fired thermal power plant (Wikipedia)

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 MPa (~157 technical atmospheres) and 2) modern large capacity power plants operate at supercritical pressures from 23.5 MPa and up to 38 MPa (see Figure *3* and Tables *2* and *3*). Supercritical pressures 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. Therefore, usually subcritical-pressure plants have thermal efficiencies of about 34–38 % and modern supercritical-pressure plants — 43–50 % and even slightly above. Steam-generators outlet temperatures

Table 2

Major parameters of selected Hitachi supercritical plants (turbines) (Mokry et al., 2008)

First Year of Operation	Power Rating MW_{el}	Pressure MPa(g)	T_{main}/T_{reheat} °C
2011	495	24.1	566/566
2010	809	25.4	579/579
	790	26.8	600/600
2009	1000	25.0	600/620
	1000	25.5	566/566
	600	24.1	600/620
2008	1000	24.9	600/600
	887	24.1	566/593
	677	25.5	566/566
2007	1000	24.9	600/600
	870	25.3	566/593
2006	600	24.1	566/566
2005	495	24.1	566/566
2004	700	24.1	538/566
2003	1000	24.5	600/600
2002	700	25.0	600/600
1998	1000	24.5	600/600
1994	1000	24.1	538/566

First Year of Operation	Power Rating MW_{el}	Pressure MPa(g)	T_{main}/T_{reheat} °C
1992	700	24.1	538/566
1991	600	24.1	538/566
1989	1000	24.1	538/566
	700	24.1	538/566
1984–1985	600	24.1	538/538
1983	700	24.1	538/538
	600	24.1	538/566
	350	24.1	538/566
1981	500	24.1	538/538
1979	600	24.1	538/566
1977	1000	24.1	538/566
	600	24.1	538/552/566*
1975	450	24.1	538/566
1974	500	24.1	538/566
1973	600	24.1	538/552/566*
	450	24.1	538/566
1971–1972	600	24.1	538/566

* Double-steam-reheat-cycle turbines

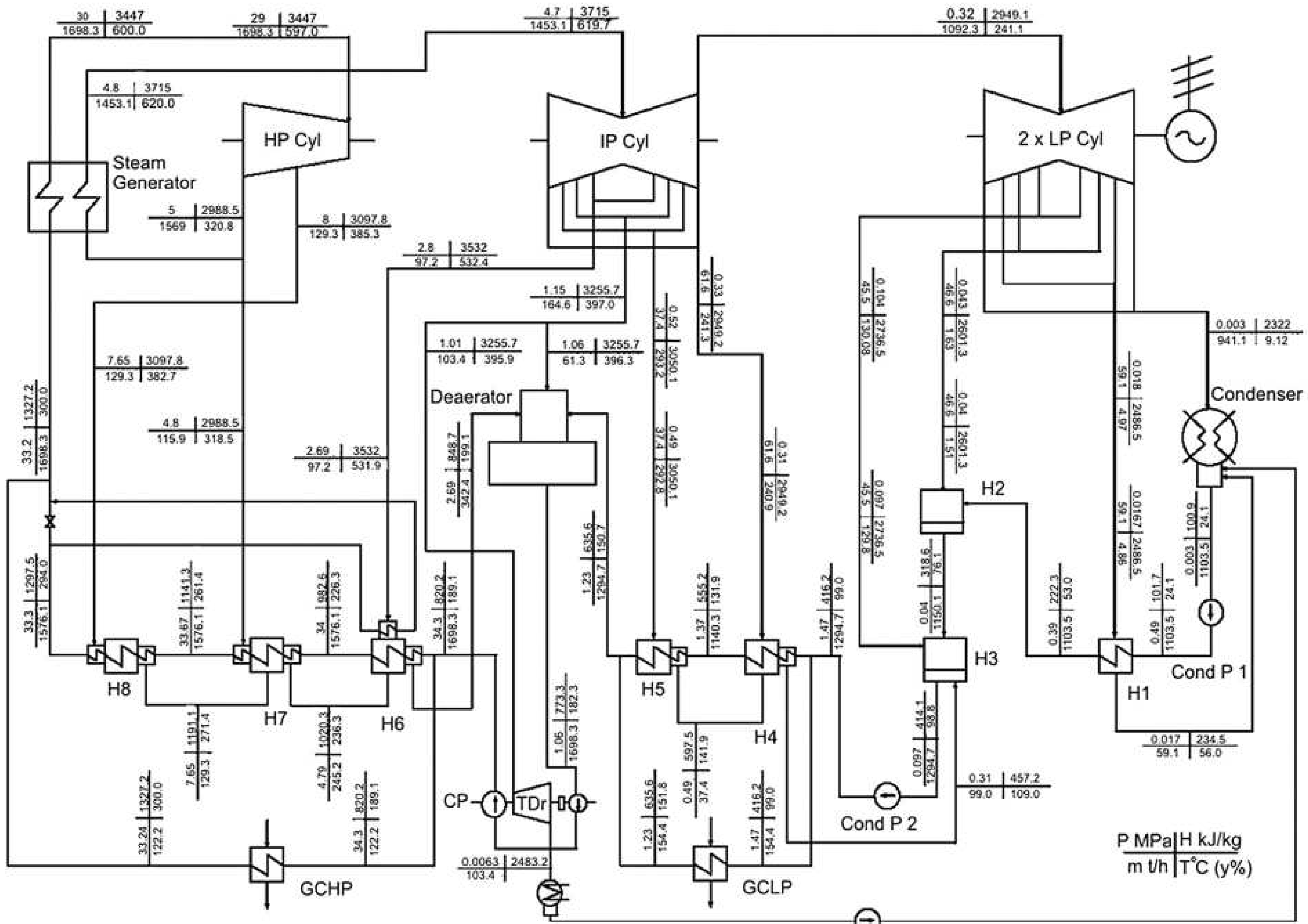


Fig. 3. Single-reheat-regenerative cycle 600-MWel Tom'-Usinsk thermal power plant (Russia) layout (Kruglikov et al., TsKTI, Russia, 2009):
Cyl — Cylinder; H — Heat exchanger (feedwater heater); CP — Circulation Pump; TDr — Turbine Drive; Cond P — Condensate Pump;
GCHP — Gas Cooler of High Pressure; and GCLP — Gas Cooler of Low Pressure

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 Tables *2* and *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 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 several figures (Figs. *4–7*) and Table *4* are shown below.

Definitions of Selected Terms and Expressions Related to Critical and Supercritical Regions (Pioro and Mokry, 2011)

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 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*.

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.

Table 3

Parameters of largest Russian supercritical-pressure turbines (Grigor'ev and Zorin, 1982)

Parameters	K-1200-240	K-800-240	K-800-240*
Power, MW_{el} (max power)	1200 (1380)	800 (850)	800 (835)
Main Steam			
Pressure, MPa	23.5	23.5	23.5
Temperature, °C	540	540	560
Max Flow Rate Through HP Turbine, t/h	3950	2650	2500
Reheat Steam			
Pressure, MPa	3.5	3.2	3.4
Temperature, °C	540	540	565
No. of Steam Extractions	9	8	8
Outlet Pressure, kPa	3.6	3.4	2.9
Cooling Water			
Temperature, °C	12	12	12
Flow Rate, m^3/h	108,000	73,000	85,000
Feedwater Temperature, °C	274	274	270
Turbine Layout			
No. of Cylinders	5	5	6
No. of High Pressure (HP) Cylinders	1	1	–
No. of Intermediate Pressure (IP) Cylinders	2	2	–
No. of Low Pressure (LP) Cylinders	2	2	–
Turbine Mass and Dimensions			
Total Mass, t	1900	1300	1600
Total Length, m	48	40	40
Total Length with Electrical Generator, m	72	60	46
Average Diameter of HP Turbine, m	3.0	2.5	2.5

* Double-shaft turbine

Table 4

Critical parameters of selected fluids (Pioro and Duffey, 2007)

Fluid	P_{cr}, MPa	T_{cr}, °C	ρ_{cr}, kg/m³
Carbon dioxide (CO_2)	7.3773	30.978	467.6
Helium (He)	0.22746	– 267.95	69.641
Water (H_2O)	22.064	373.95	322.39

General trends of various properties near the critical and pseudocritical points (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 Fig. *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).

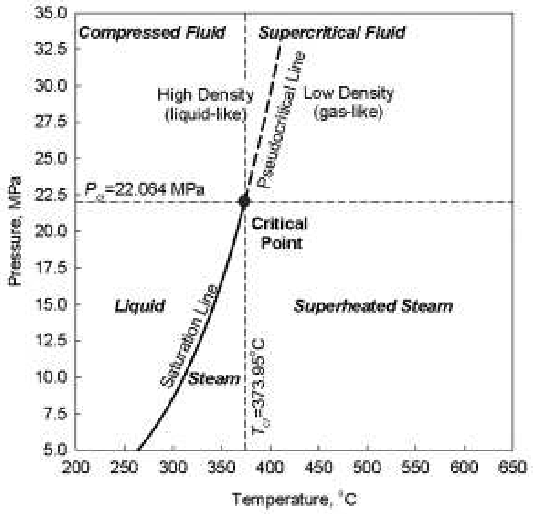


Fig. 4. Pressure-Temperature diagram for water

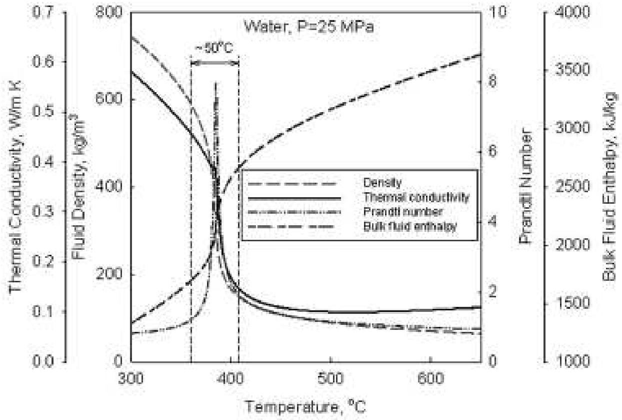


Fig. 5. Variations of selected thermophysical properties of water near pseudocritical point: Pseudocritical region at 25 MPa is about ~50 °C

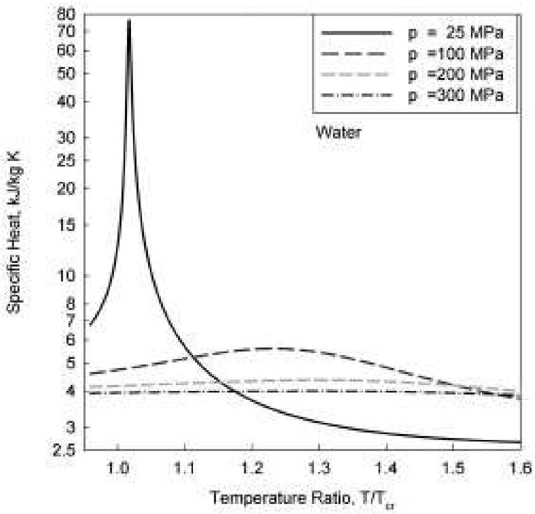


Fig. 6. Specific heat variations at various supercritical pressures: Water

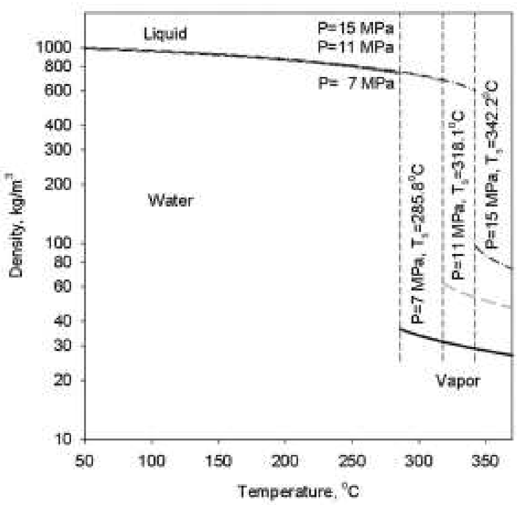


Fig. 7. Density variations at various subcritical pressures for water: Liquid and vapour

At the 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 the critical and pseudocritical regions (see Fig. *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 Fig. *6*).

At supercritical pressures properties such as density and dynamic viscosity undergo a significant drop (near the critical point this drop is almost vertical) within a very narrow temperature range (see Fig. *5*), while the kinematic viscosity and specific enthalpy 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 Fig. *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 undergo through the minimum right after the critical and pseudocritical points (see Fig. *5*).

The specific heat of water (as well as of other fluids) has the maximum value at the critical point. The exact temperature that corresponds to the specific-heat peak above the critical pressure is known as the pseudocritical temperature (see Fig. *4*). At pressures approximately above 300 MPa (see Fig. *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.

In general, crossing the pseudocritical line from left to right (see Fig. *4*) is quite similar as crossing the sat-

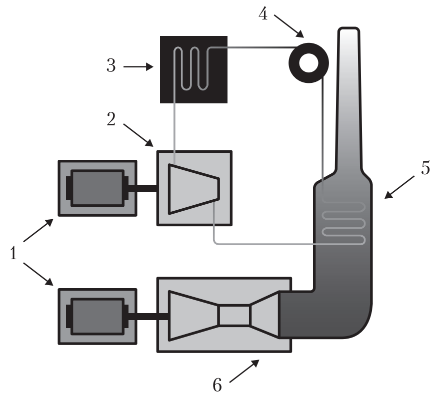


Fig. 8. Working principle of combined-cycle power plant (gas turbine (Brayton cycle) and steam turbine (Rankine cycle) plant) (Wikipedia):
1. Electrical generators; 2. Steam turbine; 3. Condenser; 4. Circulation pump; 5. Steam generator; 6. Gas turbine

uration 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 Fig. *5*). On the contrary, at subcritical pressures there are properties discontinuation on the saturation line: one value for liquid and another for vapour (see Fig. *7*). Therefore, supercritical fluids behave as single-phase substances. Also, when dealing with supercritical fluids we usually apply the term *"pseudo"* in front of a *critical point*, *boiling*, *film boiling*, etc.

Combined-cycle power plants

Natural gas is considered as a clean fossil fuel compared to coal and oil, but still due to combustion process emits a lot of carbon dioxide when it used for

Table 5

Operating and forthcoming nuclear power reactors (in total — 444 (net 378 GWel); these data don't account for reactors shut down during and after the Japan earthquake and tsunami disaster in spring of 2011) (Nuclear News, 2011)

Pressurized light-Water Reactors (PWRs) — 268 (248 GWel); forthcoming — 90 (94 GWel)
Boiling light-Water Reactors (BWRs or ABWRs) — 92 (85 GWel); forthcoming — 6 (8 GWel)
Gas-Cooled Reactors (GCRs) — 18 (9 GWel), UK (AGRs — 14 and Magnox — 4); forthcoming — 1 (0.2 GWel)
Pressurized Heavy-Water Reactors (PHWRs) — 50 (25 GWel), Argentina 3, Canada 22, China 2, India 18, Pakistan 1, Romania 2, S. Korea 4; forthcoming — 9 (5 GWel)
Light-water, Graphite-moderated Reactors (LGRs) — 15 (10 GWel), Russia 11 RBMKs and 4 EGPs (earlier prototype of RBMK)
Liquid-Metal Fast-Breeder Reactors (LMFBRs) — 1 (0.6 GWel), Russia; forthcoming — 3 (1.5 GWel)

Table 6

Current nuclear power reactors by nation (10 first nations) (these data don't account for reactors shut down during and after 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	54	47
4	Russia	32	23
5	Germany	17	20
6	Canada	22	15
7	S. Korea	20	18
8	Ukraine	15	13
9	UK	19	10
10	China	13	10

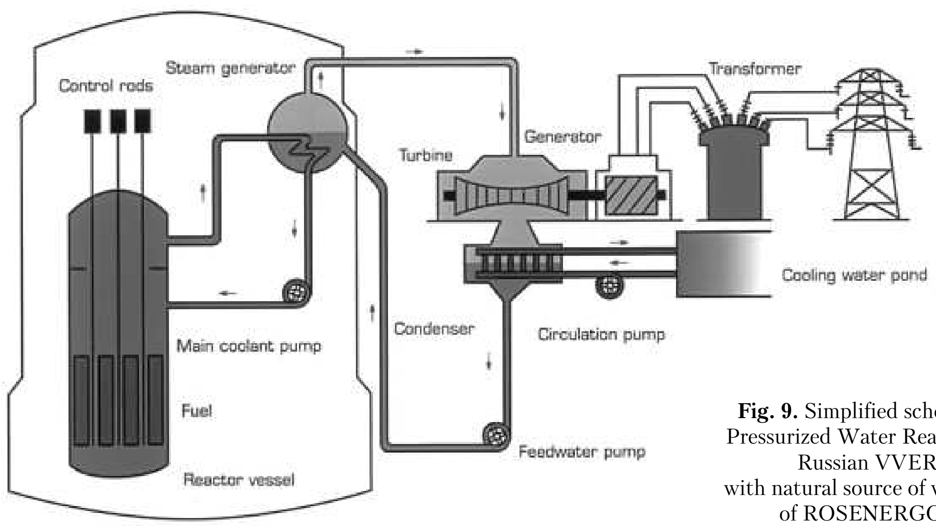


Fig. 9. Simplified scheme of typical Pressurized Water Reactor (PWR) — Russian VVER cooled with natural source of water (courtesy of ROSENERGOATOM)

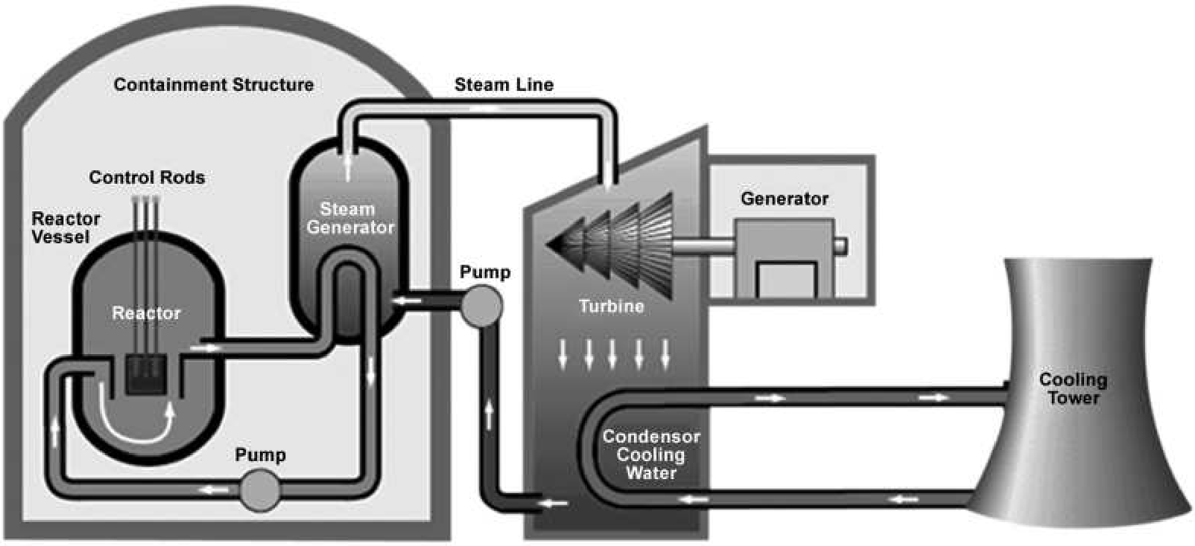


Fig. 10. Simplified scheme of typical Pressurized Water Reactor (PWR) with cooling tower (Zberecki, 2001)

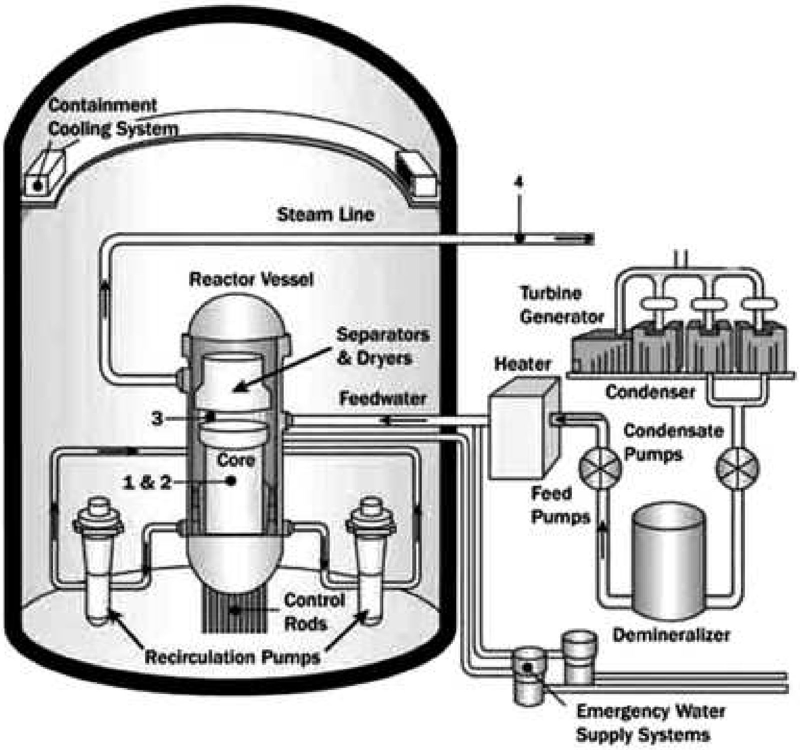


Fig. 11. Simplified scheme of typical Boiling Water Reactor (BWR) (courtesy of NRC USA)

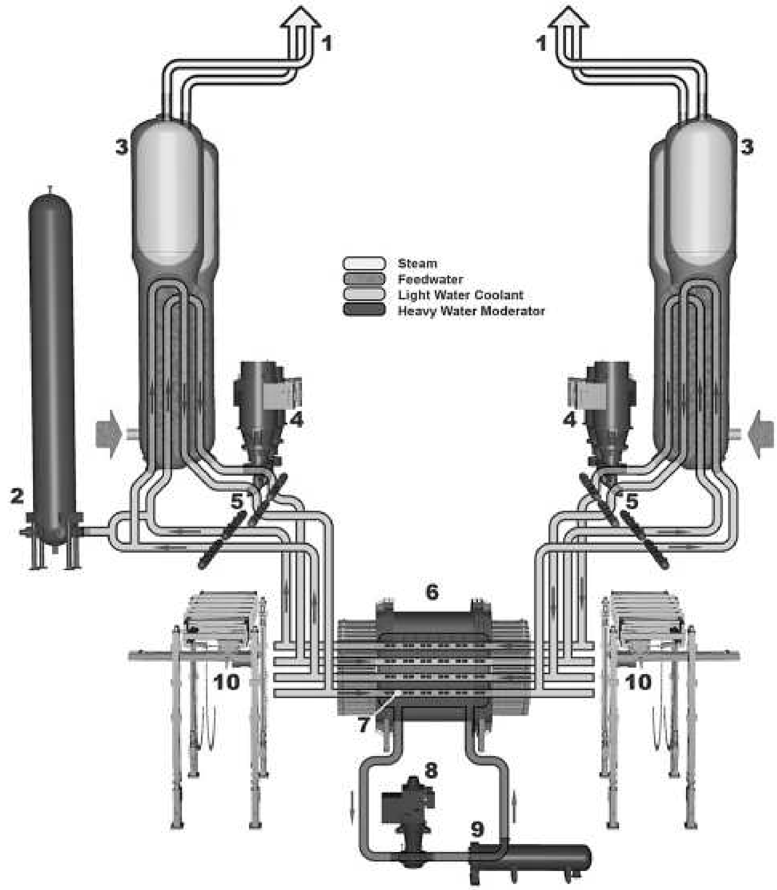


Fig. 12. Simplified scheme of Pressurized Heavy Water Reactor (PHWR) – CANDU reactor (courtesy of AECL):
1. Main Steam Pipes; 2. Pressurizer; 3. Steam Generators (4); 4. Heat Transport Pumps (4); 5. Headers; 6. Calandria; 7. Fuel; 8. Moderator Pumps (2); 9. Moderator Heat Exchangers (2); 10. Fuelling Machines (2)

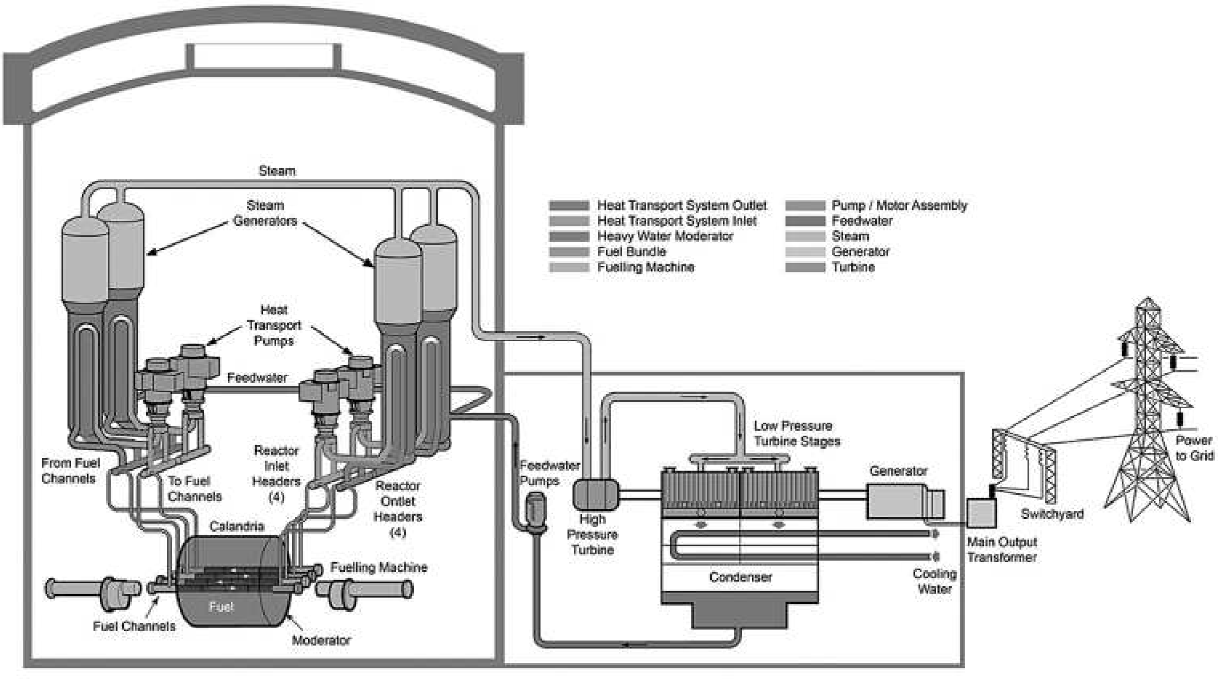


Fig. 13. Simplified scheme of CANDU-6 reactor nuclear power plant (courtesy of AECL)

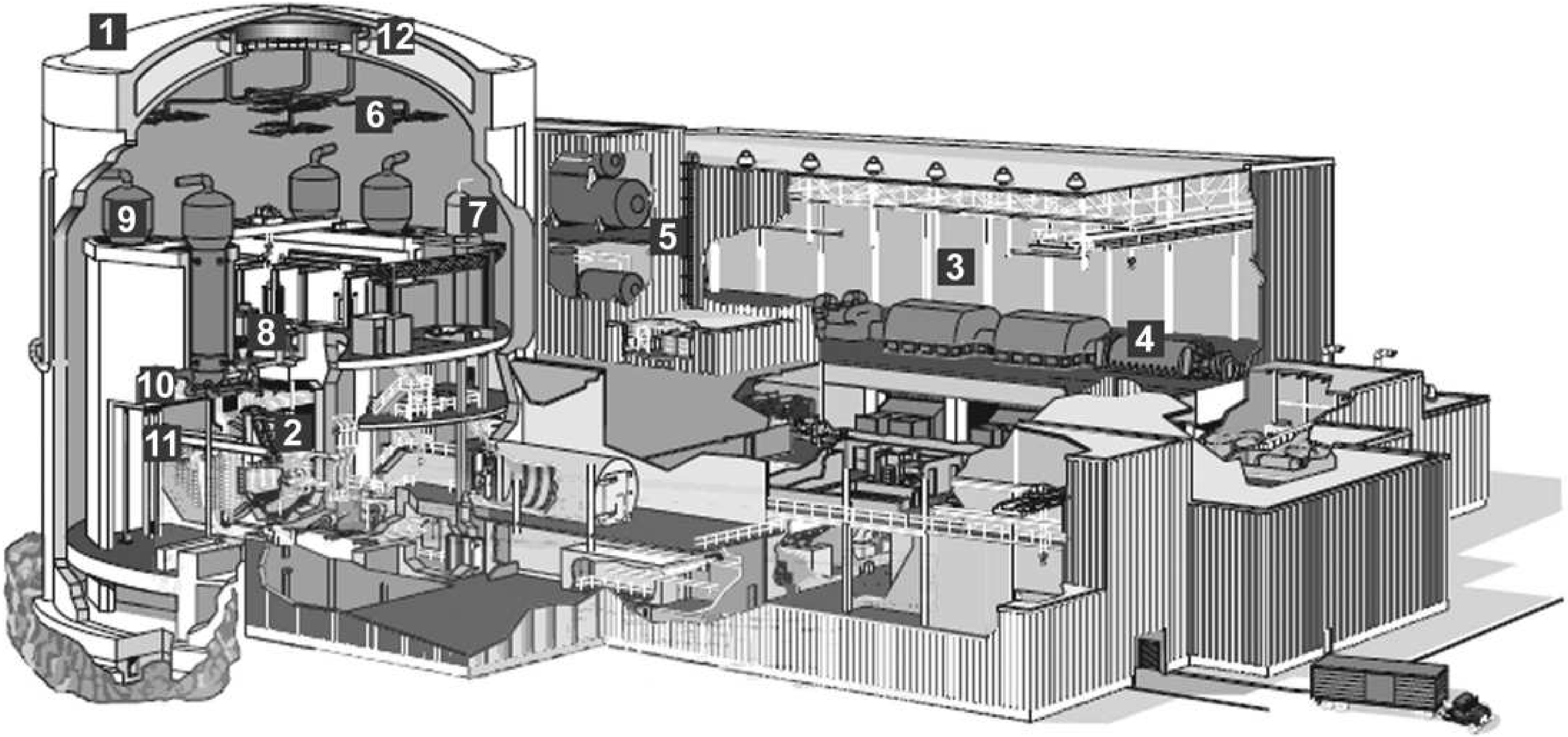


Fig. 14. Enhanced CANDU-6 (EC6) Nuclear Power Plant (NPP) (courtesy of AECL):
1. Reactor building; 2. Calandria vessel; 3. Turbine building; 4. Turbine generator; 5. Service building; 6. Spray system; 7. Pressurizer; 8. Heat-transport pumps; 9. Steam generators; 10. Heat-transport system; 11. Fuelling machine and 12. Reserve water tank

Fig. 15. CANDU-6 reactor main control room (courtesy of AECL)

electrical generation. The most efficient modern thermal-power plants with thermal efficiencies within a range of 50–60 % are, so-called, combined-cycle power plants, which use natural gas as a fuel (see Figure *8*). As an example, a 1350-MW combined-cycle power plant built in Turkey in 1988 by Siemens (Germany) has efficiency of 52.5 %. This plant has six 150-MW gas turbines and three 173-MW steam turbines.

However, natural gas combustion still emits carbon dioxide into atmosphere, which is currently considered as one of the major reasons for a climate change. In addition, all fossil-fuels resources are depleting quite fast. Therefore, a new reliable and environmental friendly source for the electrical-energy generation should be considered.

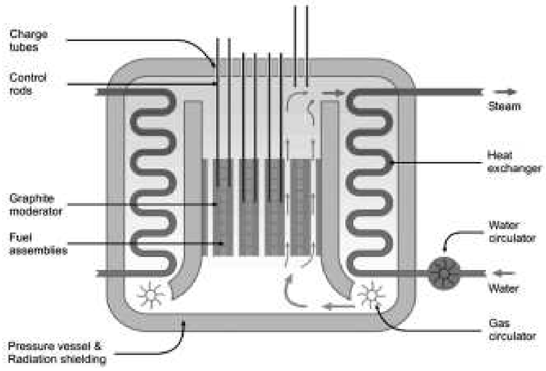


Fig. 16. Simplified scheme of Advanced Gas-cooled Reactor (AGR) (source: Wikimedia). Note that the heat exchanger is contained within the steel-reinforced concrete combined pressure vessel and radiation shield

Nuclear Power Plants
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.

First success of using nuclear power for the electrical generation was achieved in several countries within 50-s, and currently, Generations II and III nuclear reactors are operating around the world (see Tables *5* and 6 and Figures *9–19*). Definitions of nuclear reactors generations are as the following: 1) Generation I

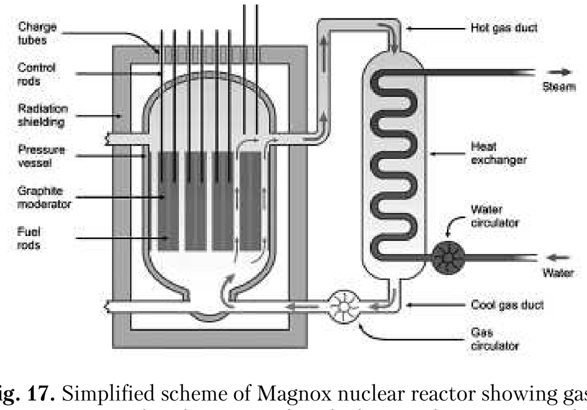


Fig. 17. Simplified scheme of Magnox nuclear reactor showing gas flow (source: Wikipidea). Note that the heat exchanger is outside the concrete radiation shielding. This represents an early Magnox design with a cylindrical, steel, pressure vessel

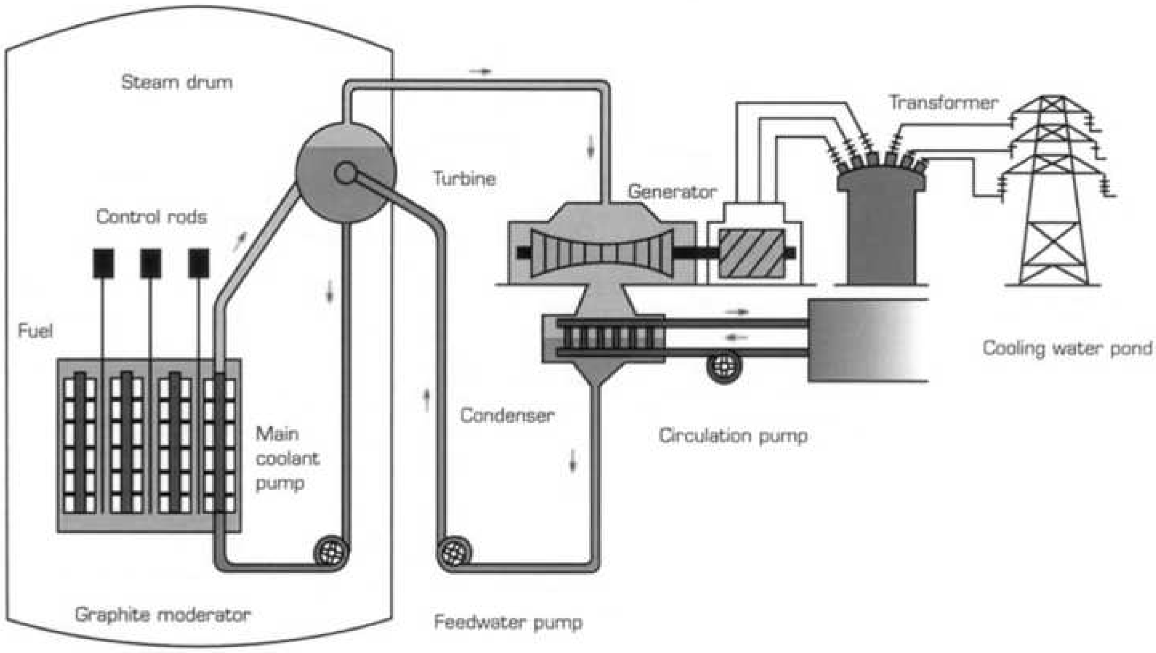


Fig. 18. Simplified scheme of Light-water, Graphite-moderated Reactor (LGR) — Russian RBMK (boiling reactor) (courtesy of ROSENERGOATOM)

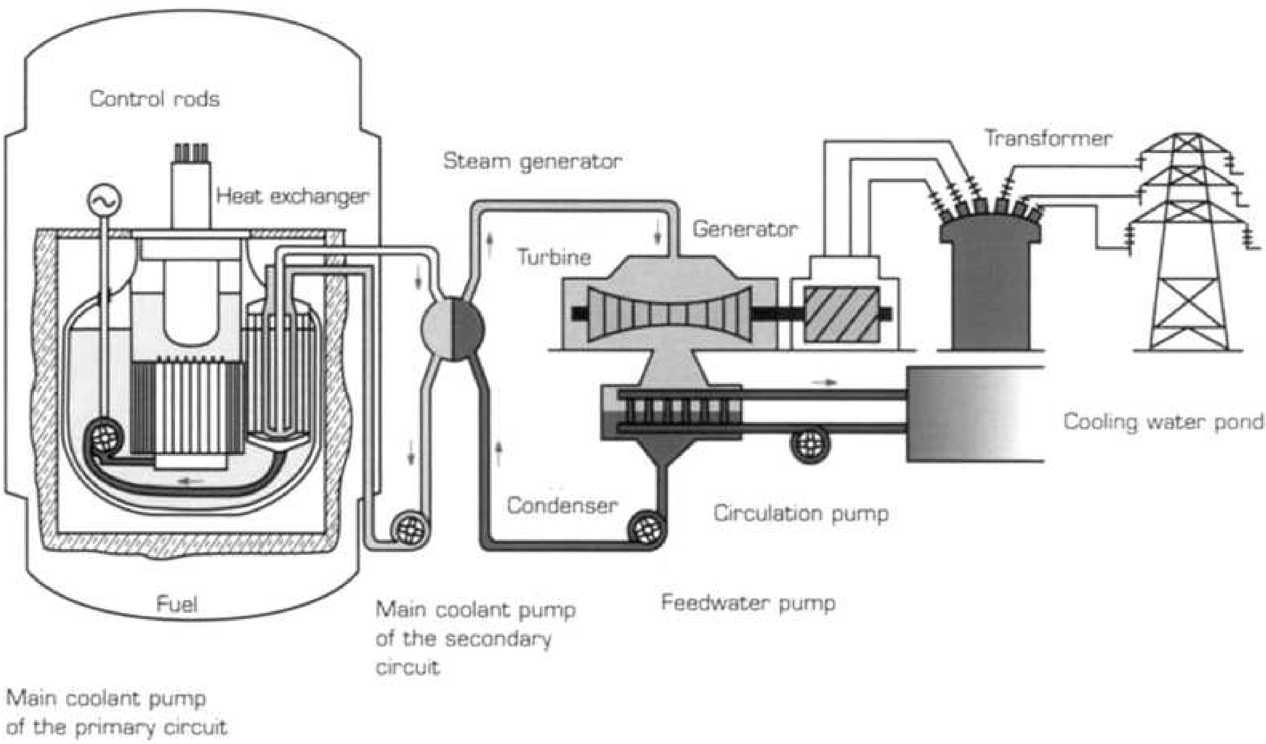
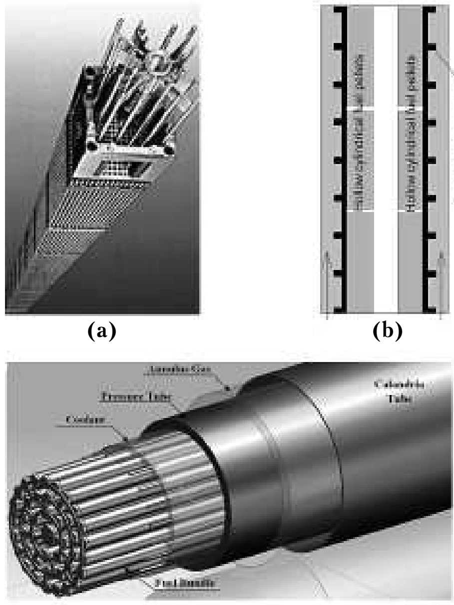


Fig. 19. Simplified scheme of Liquid-Metal Fast-Breeder Reactor (LMFBR) — Russian BN-600 (sodium-cooled) (courtesy of ROSENERGOATOM)

(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 the thermal efficiency of 30–35 %; carbon-dioxide-cooled NPPs with the thermal efficiency up to 42 % и 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 *7*); 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).

Analysis of data listed in Table *5* shows that the vast majority of nuclear reactors are water-cooled units. Only reactors built in UK are gas-cooled type, and one reactor in Russia uses liquid sodium for its cooling.

UK Gas-Cooled Reactors (GCRs) consist of 2 designs (Hewitt and Collier, 2000): 1) older design — Magnox reactor (see Fig. *17*) and 2) newer design — Advanced Gas-cooled Reactor (AGR) (see Fig. *16*). 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 31.5 %. Advanced Gas-cooled Reactors (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 Fig. *12*); enriched fuel 2.3 %; and thermal efficiency — about 41.6 % (the highest in nuclear-power industry so far). However, these reactors will not be reproduced 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.



(a) (b)

(c)

Fig. 20. 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 (c)

Table 7

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

ABWR — Toshiba, Mitsubishi Heavy Industries and Hitachi-GE (Japan, USA) (the only one Generation III+ reactor currently under operation)
Advanced CANDU Reactor (ACR-1000) AECL, Canada
Advanced Plant (AP-1000) — Toshiba-Westinghouse (Japan-USA) (12 planned to be built in China and 6 in USA)
European Pressurized-water Reactor (EPR) Areva, France (1 under construction in Finland and 1 in France, 2 planned to be built in China and 2 — in USA)
VVER (Generation III+) — Gidropress, Russia (several planned to be built in various countries)

Table 8

Table 8. Major Parameters of Russian Power Reactors (Grigor'ev and Zorin, 1982)

Parameter	VVER-440	VVER-1000 (Fig. 1)	EGP-6	RBMK-1000 (Fig. 10)	BN-600 (Fig. 11)
Thermal power, MWth	1375	3000	62	3200	1500
Electrical power, MWel	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 flow, t/h	40,800	84,800	600	48,000	25,000
Coolant temperature, °C	270/298	290/322	265	284	380/550
Steam flow rate, t/h	2700	5880	92	5600	660
Steam pressure, MPa	4.3	5.9	6.0	6.6	14.0
Steam temperature, °C	256	276	–	280	505
Core: Diameter/Height m/m	3.8/11.8	4.5/10.9	–	11.8/7	2.1/0.75
Fuel enrichment, %	3.6	4.3	3.0–3.6	2.0–2.4	17–33
No. of fuel assemblies (bundles)	349	163	273	1580	369

Just for reference purposes, typical fuel elements (rods)/bundles of various reactors are shown in Figure *20*; typical maximum fuel-sheath temperatures for reactor steady operation are listed in Table *12*, typical heat-transfer-coefficient values for reactor coolants and typical heat fluxes for steady operation are listed in Tables *13* and *14*, respectively.

Table 9

Additional parameters of VVER-1000

Pressure vessel 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

All current reactors and oncoming Generation III+ reactors are not very competitive with modern power plants in terms of thermal efficiency, a difference in values of thermal efficiencies between thermal and nuclear power plants can be up to 20–25 %. Therefore, new generation nuclear power plants should be designed and built in the nearest future.

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 efficiency in order to increase electricity generation and decrease detrimental effects on the environment. The current fleet of nuclear power plants 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 only water-cooled 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 lower for steam.

Table 10

Typical parameters of US PWR (Shultis and Faw, 2008)

Power		**Steam generators**	
Thermal output, MW$_{th}$	3800	Number	103
Electrical output, MW$_{el}$	1300	Outlet pressure, MPa	63
Thermal efficiency, %	34	Outlet temperature, °C	47
Specific power, kW/kg(U)	33	Mass flow rate, kg/s	23
Power density, kW/L	102	**Fuel**	
Average linear heat flux, kW/m	17.5	Fuel pellets	UO_2
Rod heat flux ave/max, MW/m^2	0.584/ 1.46	Pellet outside diameter, mm	8.19
Reactor pressure vessel		Fuel loading, t	115
Outside diameter, m	4.4	Enrichment, %	3.2
Height, m	13.6	Rod outside diameter, mm	9.5
Wall thickness, m	0.22	Zircaloy sheath thickness, mm	0.57
Core		Rods per bundle (17x17)	264
Length, m	4.17	Bundles in core	193
Outside diameter, m	3.37	**Reactivity control**	
Reactor coolant system		No. of control assemblies	68
Pressure, MPa	15.5	Shape — rod clusters	
Inlet temperature, °C	292	Absorber rods per assembly	24
Outlet temperature, °C	329	Neutron absorber Ag-In-Cd and/or B_4C	
Mass flow rate, kg/s	531	Soluble poison shim boric acid H_3BO_3	

Table 11

Typical parameters of US PWR (Shultis and Faw, 2008)

Power		**Reactor coolant system**	
Thermal output, MW$_{th}$	3800	Core flow rate, kg/s	14,167
Electrical output, MW$_{el}$	1330	Core void fraction ave/max	0.37/ 0.75
Thermal efficiency, %	34	**Fuel**	
Specific power, kW/kg(U)	26	Fuel pellets	UO_2
Power density, kW/L	56	Pellet outside diameter, mm	10.6
Average linear heat flux, kW/m	20.7	Fuel loading, t	168
Rod heat flux ave/max, MW/m^2	0.51/ 1.12	Enrichment, %	1.9
Reactor pressure vessel		Rod outside diameter, mm	12.5
Inside diameter, m	6.4	Zircaloy sheath thickness, mm	0.86
Height, m	22.1	Rods per bundle (8x8)	62
Wall thickness, m	0.15	Bundles in core	760
Core		**Reactivity control**	
Length, m	3.76	No. of control assemblies	193
Outside diameter, m	4.8	Shape — rod clusters	
Reactor coolant system		Overall length, m	4.42
Pressure, MPa	7.17	Shape — rod clusters	
Feedwater temperature, °C	216	Length of poison section, m	3.66
Steam outlet temperature, °C	290	Neutron absorber — boron carbide	
Outlet steam flow rate, kg/s	2083	Soluble poison shim — gadolinium	

Table 12

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

AGR stainless steel	750 °C
Sodium-cooled fast reactor	750 °C
Magnesium alloy (Magnox)	450 °C
PWR	320 °C
BWR	300 °C

Table 14

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

	Heat Flux, MW/m^2	T_{sheath}-T_{fluid}, °C
Sodium-cooled fast reactor (LMFBR)	2.0	35
PWR	1.5	50
BWR	1.0	15
CANDU reactor	0.625	14
Boiling water in kettle	0.15	15

Table 13

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

Boiling water (flow boiling) (BWR)	60 kW/m^2K
Liquid sodium (forced convection) (LMFBR)	55 kW/m^2K
CANDU reactor	46 kW/m^2K
Water (single-phase forced convection) (PWR)	30 kW/m^2K
Boiling water in kettle	10 kW/m^2K
High-pressure carbon dioxide (forced convection) (Magnox, AGR)	1 kW/m^2K

Table 15

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

Reactor Parameter	Unit	Reference Value
Reactor Power	MW_{th}	600
Coolant Inlet/ Outlet Temperature	°C	490/850
Pressure	MPa	9
Core Inlet/ Outlet Pressure	MPa	Dependent on process
Coolant Mass Flow Rate	kg/s	320
Average Power Density	MW_{th}/m^3	100
Reference Fuel Compound	–	UPuC/SiC (70/30 %) with about 20 % Pu
Volume fraction, Fuel, Gas, SiC	%	50/40/10
Boiling water in kettle	%	48

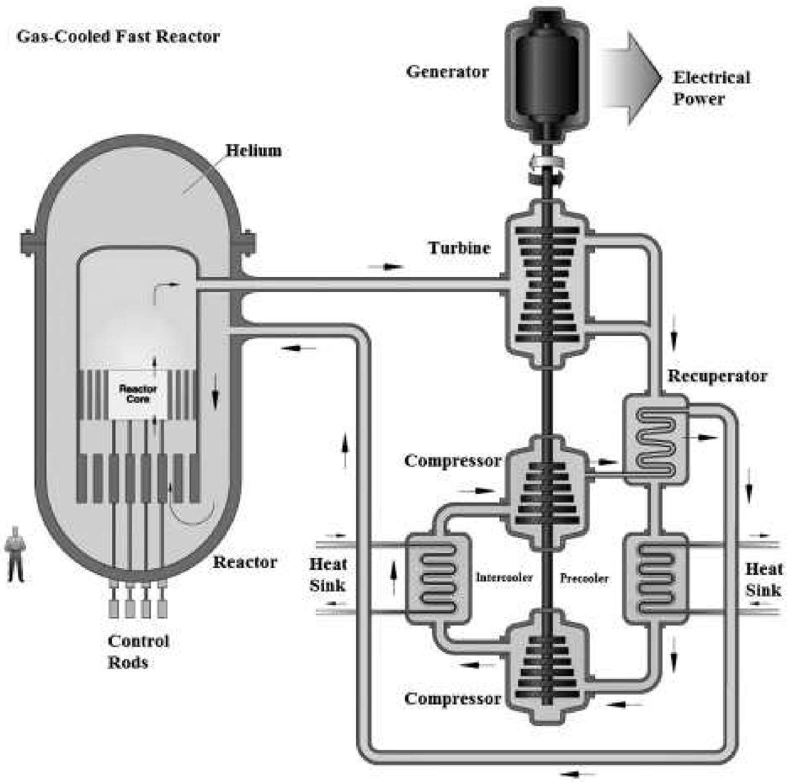


Fig. 21. Schematic diagram of Very High Temperature Reactor (US DOE)

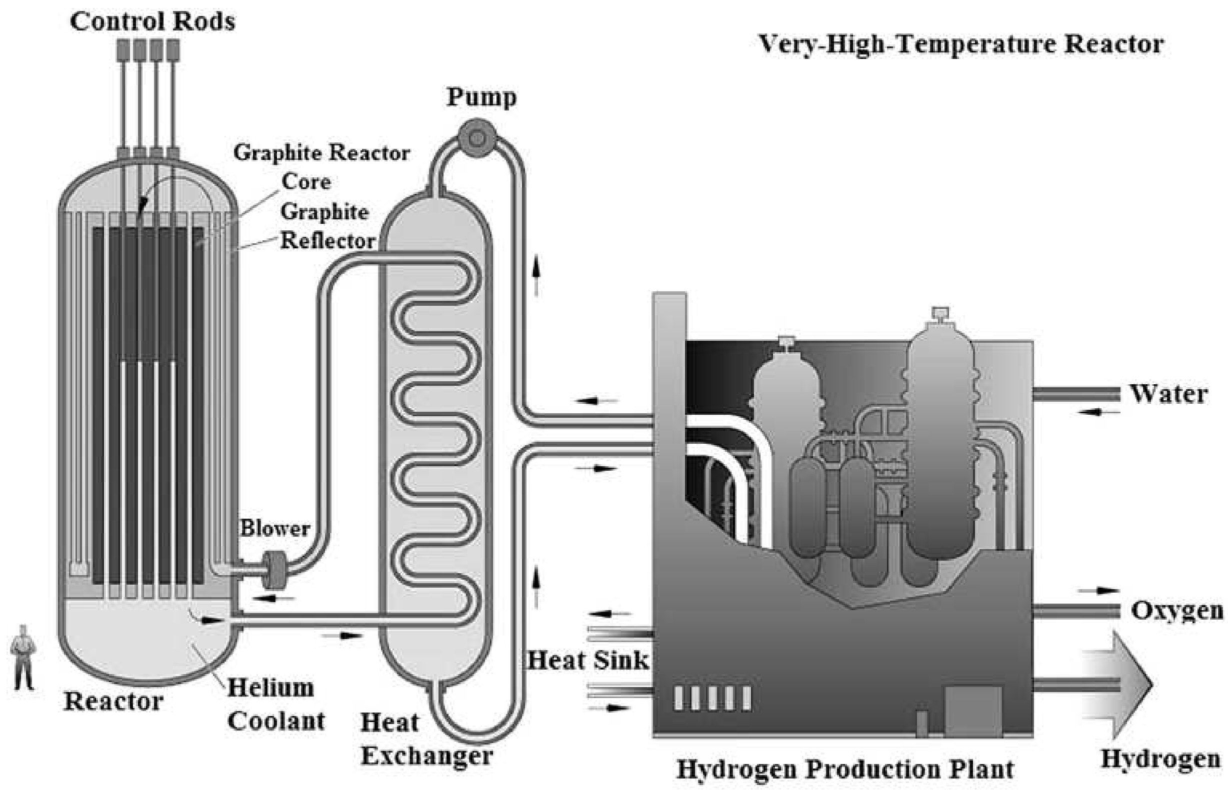


Fig. 22. Schematic diagram of Very High Temperature Reactor (US DOE)

Table 16

Key design parameters of Very High Temperature Reactor

Reactor Parameter	Unit	Reference Value
Reactor Power	MW_{th}	600
Average Power Density	MW_{th}/m^3	6–10
Coolant Inlet/Outlet Temperature	°C	640/1000
Coolant: Mass Flow Rate	kg/s	Helium: 320
Reference Fuel Compound	–	ZrC-coated particles in pins or pebbles
Net Plant Efficiency	%	>50

Table 17

Key design parameters of Sodium Cooled Reactor

Reactor Parameter	Unit	Reference Value
Reactor Power	MW_{th}	1000–5000
Coolant Outlet Temperature	°C	530–550
Pressure	MPa	~0.1
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 18

Key design parameters of Lead-Cooled Fast Reactor

Reactor Parameter	Unit	SSTAR	ELSY
Reactor Power	MW_{th}	19.8	600
Thermal Efficiency	%	44	42
Primary Coolant	–	Lead	Lead
Coolant Inlet/ Outlet Temperature	°C	420/567	400/480
Peak Cladding Temperature	°C	650	550
Fuel	–	Nitrides	MOX, (Nitrides)
Fuel Pin Diameter	mm	25	10.5

Table 19

Key design parameters of Molten Salt Reactor

Reactor Parameter	Unit	Reference Value
Reactor Power	MW_e	1000
Net thermal Efficiency	%	44–50
Average Power Density	MW_{th}/m^3	22
Fuel-Salt Inlet/ Outlet Temperature	°C	565/700
Vapor Pressure	kPa	~0.7
Moderator	–	Graphite
Neutron Spectrum Burner	–	Thermal-Actinide

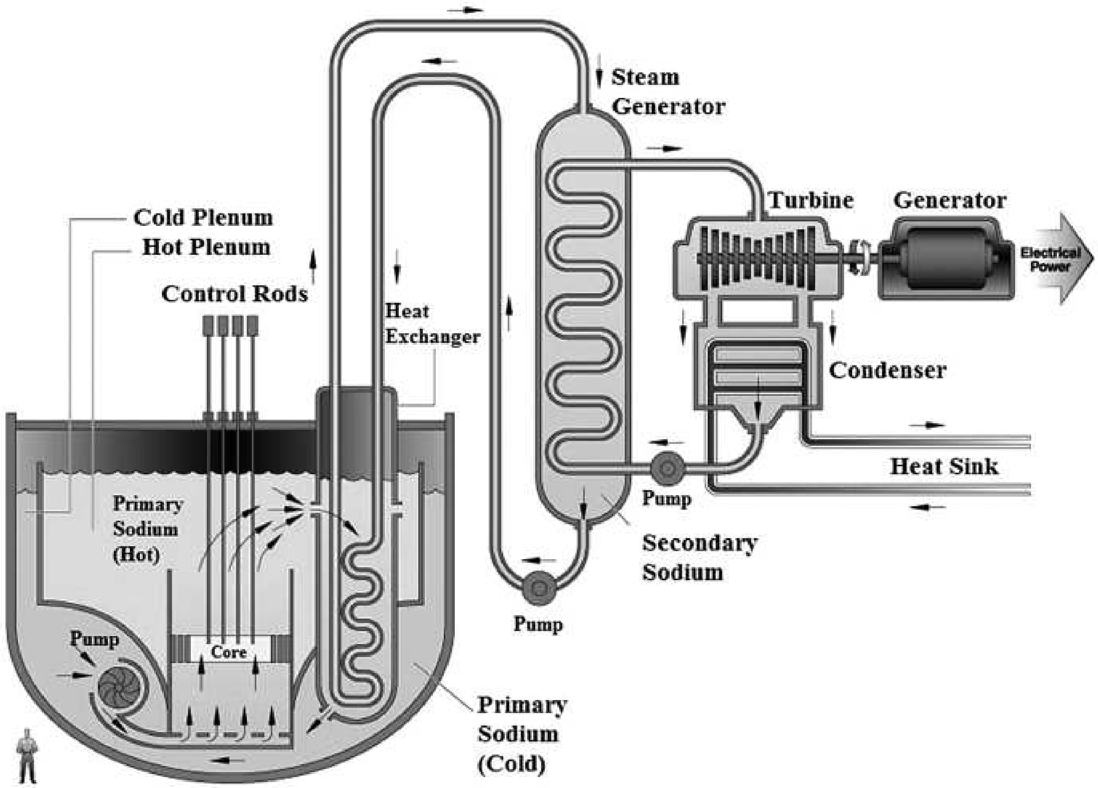


Fig. 23. Schematic diagram of Sodium-Cooled Reactor (US DOE, 2002)

Currently, a group of countries, including Russia, USA, Japan, EU, Canada 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 to increase the thermal efficiency from 30–35% to 45–50%. This increase in thermal efficiency would result in a higher production of electricity compared to current Pressurized Water Reactor (PWR) or Boiling Water Reactor (BWR) technologies per 1 kg of uranium.

The Generation IV International Forum (GIF) Program has narrowed design options of the nuclear reactors to six concepts. These concepts are: 1) Gas-cooled Fast Reactor (GFR), 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 *21–25* show schematic images 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.

Gas-cooled Fast Reactor (GFR) is a fast-neutron-spectrum reactor, which can be used for the production of electricity and co-generation of hydrogen through thermochemical processes. The coolant is helium with inlet and outlet temperatures of 490 and 850 °C, respectively. The net plant efficiency is 48 % with a direct Brayton cycle. Table *15* lists a summary of design parameters for GFR (US DOE, 2002).

Very High Temperature Reactor (VHTR) is a thermal-neutron-spectrum reactor. The ultimate purpose of this nuclear-reactor design is the co-generation of hydrogen through thermochemical processes. 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 *16* (US DOE, 2002).

Similar to GFR, Sodium-cooled Fast Reactor (SFR) is a fast-neutron-spectrum reactor. The main objectives of SFR are the management of high-level radioactive waste and production of electricity. SFR uses liquid sodium as its coolant with an outlet temperature between 530 and 550 °C at an atmospheric pressure. The primary choices of fuel for SFR are oxide and metallic fuels. Table *17* lists a summary of design parameters of SFR (US DOE, 2002).

Lead-cooled Fast Reactor (LFR) is a fast-neutron-spectrum reactor, which uses lead or lead-bismuth as the coolant. The outlet temperature of the coolant is 480–567°C at an atmospheric pressure. The primary choice of fuel is a nitride fuel. The Brayton cycle has

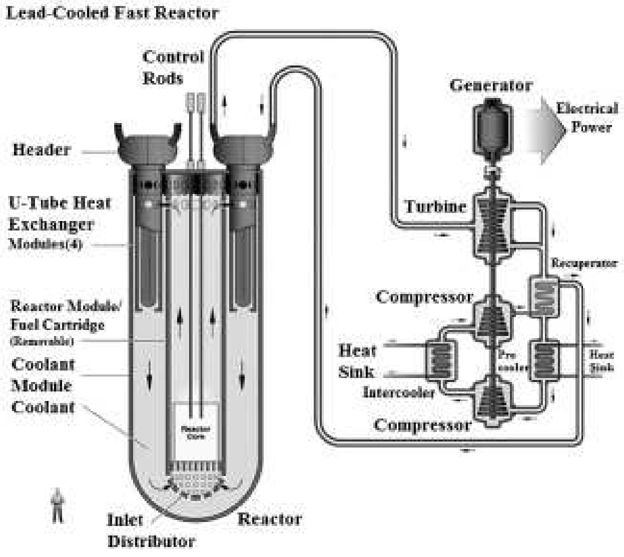


Fig. 24. Schematic diagram of Lead-Cooled Fast Reactor (US DOE)

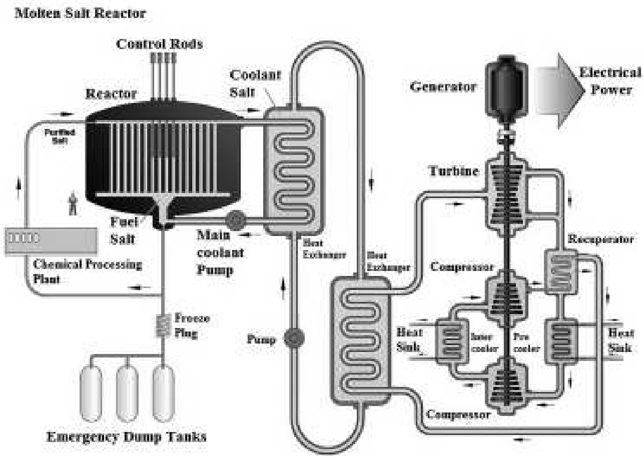


Fig. 25. Schematic diagram of Molten Salt Reactor (US DOE, 2002)

been chosen as a primary choice for the power cycle while the supercritical Rankine cycle is considered as the secondary choice. Table *18* lists several of key design parameters of LFR (OECD Nuclear Energy Agency, 2010; US DOE, 2002).

Molten Salt Reactor (MSR) 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 *19* lists the design parameters of MSR (US DOE, 2002).

A SuperCritical Water-cooled Reactor (SCWR) will be discussed in Part 2 of this paper.

Nomenclature

P, p — pressure, Pa
H — specific enthalpy, J/kg
m — mass flow rate, kg/s
T — temperature,°C
V — specific volume, m^3
Greek letters
ρ — density, kg/m^3

Subscripts

cr — critical
el — elctrical
pc — pseudocritical
s — saturation
th — thermal

Abbreviations

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 Graphite with Steam Reheat (reactor) (in Russian abbreviations)
EU — European Union
GCR — Gas-Cooled Reactor
GFR — Gas Fast Reactor
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
NIST — National Institute of Standards and Technology (USA)
NPP — Nuclear Power Plant
NRC — National Regulatory Commission (USA)
PHWR — Pressurized Heavy-Water Reactor
PV — Pressure Vessel
PWR — Pressurized Water Reactor

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

References

1. *Grigor'ev, V. A. and Zorin, V. M.*, Editors, 1982. Thermal and Nuclear Power Plants, (In Russian), Energoatomizdat Publ. House, Moscow, Russia.

2. *Hewitt, G. F.* and *Collier, J. G.*, 2000. Introduction to Nuclear Power, 2nd ed., Taylor & Francis, New York, NY, USA, 304 pages.

3. *Kruglikov, P. A.*, *Smolkin, Yu. V.* and *Sokolov, K. V.*, 2009. Development of engineering solutions for thermal scheme of power unit of thermal power plant with supercritical parameters of steam, (In Russian), Proc. Int. Workshop "Supercritical Water and Steam in Nuclear Power Engineering: Problems and Solutions", Moscow, Russia, October 22–23, 6 pages.

4. *Mokry, S.*, *Naidin, M.*, *Baig, F.*, et al., 2008. Conceptual Thermal-Design Options for Pressure-Tube SCWRs with Thermochemical Co-Generation of Hydrogen, Proceedings of the 16th International Conference on Nuclear Engineering (ICONE-16), Orlando, Florida, USA, May 11–15, Paper #48313, 13 pages.

5. *National* Institute of Standards and Technology, 2010. NIST Reference Fluid Thermodynamic and Transport Properties-REFPROP. NIST Standard Reference Database 23, Ver. 9.0. Boulder, CO, U.S.: Department of Commerce.

6. *Nuclear* News, 2011, March, A Publication of the American Nuclear Society (ANS), pp. 45 78.

7. *Pioro, I. L.* and *Duffey, R.B.*, 2007. Heat Transfer and Hydraulic Resistance at Supercritical Pressures in Power Engineering Applications, ASME Press, New York, NY, USA, 328 pages.

8. *Pioro, I.* and *Mokry, S.*, 2011. Thermophysical Properties at Critical and Supercritical Conditions, Chapter in book "Heat Transfer. Theoretical Analysis, Experimental Investigations and Industrial Systems", Editor: A. Belmiloudi, INTECH, Rijeka, Croatia, pp. 573 592.

9. *Pioro, L. S.*, *Pioro, I. L.*, *Soroka, B. S.* and *Kostyuk, T. O.* 2010. Advanced Melting Technologies with Submerged Combustion, RoseDog Publ. Co., Pittsburg, PA, USA, 420 pages.

10. *ROSENERGOATOM*, 2004. Russian Nuclear Power Plants. 50 Years of Nuclear Power, Moscow, Russia, 120 pages.

11. *Shultis, J. K.* and *Faw, R. E.*, 2008. Fundamentals of Nuclear Science and Engineering, 2nd ed., CRC Press, Boca Raton, FL, USA, 591 pages.

12. *Zberecki, C.*, 2001. Introduction and Classification of Nuclear Reactors. http://www.if.pw.edu.pl/ ~pluta/pl/dyd/mtj/zal00/Zberecki/reaktory.htm, June.