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INTRODUCTION AND HISTORICAL DEVELOPMENT OF SCWRS

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LECTURE SC02
INTRODUCTION AND HISTORICAL DEVELOPMENT OF SCWRS

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CONTENT

PART 1: THERMOPHYSICAL PROPERTIES OF WATER AND CARBON DIOXIDE AT CRITICAL AND SUPERCRITICAL PRESSURES

1.1 INTRODUCTION

Prior to a general discussion on specifics of thermophysical properties and forced-convective heat transfer at critical and supercritical pressures, it is important to define special terms and expressions used at these conditions. For a better understanding of these terms and expressions their definitions are listed below together with complimentary Figures 1.1 and 1.2.

Definitions of Selected Terms and Expressions Related to Critical and Supercritical Regions

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.

Deteriorated Heat Transfer (DHT) is characterized with lower values of the wall heat transfer coefficient compared to those for normal heat transfer; and hence has higher values of wall temperature within some part of a test section or within the entire test section.

Improved Heat Transfer (IHT) is characterized with higher values of the wall heat transfer coefficient compared to those for normal heat transfer; and hence lower values of wall temperature within some part of a test section or within the entire test section. In our opinion, the improved heat-transfer regime or mode includes peaks or "humps" in the heat transfer coefficient near the critical or pseudocritical points.

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

Normal Heat Transfer (NHT) can be characterized in general with wall heat transfer coefficients similar to those of subcritical convective heat transfer far from the critical or pseudocritical regions, when they are calculated according to the conventional single-phase Dittus-Boelter-type correlations: $\mathbf{Nu} = 0.0023\,\mathbf{Re}^{0.8}\mathbf{Pr}^{0.4}$.

Pseudo-boiling is a physical phenomenon similar to subcritical pressure nucleate boiling, which may appear at supercritical pressures. Due to heating of a supercritical fluid with a bulk-fluid temperature below the pseudocritical temperature (high-density fluid, i.e., "liquid"), some layers near the heating surface may attain temperatures above the pseudocritical temperature (low-density fluid, i.e., "gas"). This low-density "gas" leaves the heating surface in the form of variable density (bubble) volumes. During the pseudo-boiling, the wall heat transfer coefficient usually increases (improved heat-transfer regime).

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.

Pseudo-film boiling is a physical phenomenon similar to subcritical-pressure film boiling, which may appear at supercritical pressures. At pseudo-film boiling, a low-density fluid (a fluid at temperatures above the pseudocritical temperature, i.e., "gas") prevents a high-density fluid (a fluid at temperatures below the pseudocritical temperature, i.e., "liquid") from contacting ("rewetting") a heated surface. Pseudo-film boiling leads to the deteriorated heat-transfer regime.

Supercritical fluid is a fluid at pressures and temperatures that are higher than the critical pressure and critical temperature. However, in the present chapter, the 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.

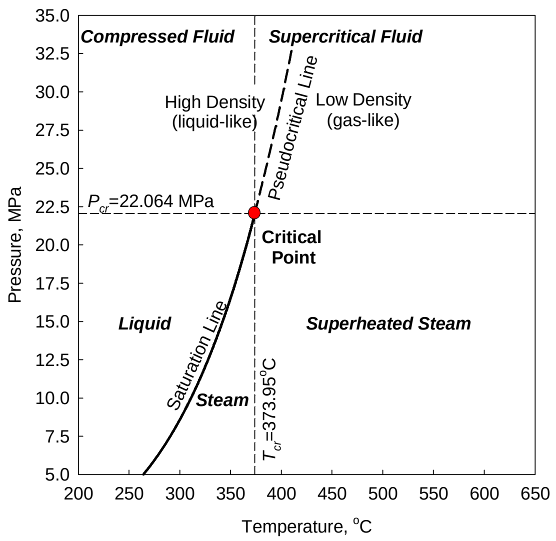

Figure 1.1. Pressure-Temperature diagram for water.

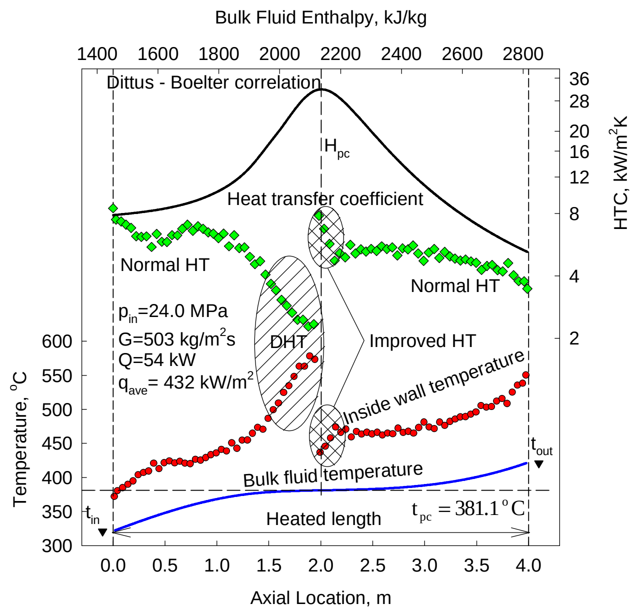

Figure 1.2. Temperature and HTC profiles along heated length of vertical tube (Kirillov et al., 2003): Water, D=10 mm and L_h=4 m.

1.2. THERMOPHYSICAL PROPERTIES AT CRITICAL AND SUPERCRITICAL PRESSURES

The general trends of various properties near the critical and pseudocritical points (Pioro, 2008; Pioro and Duffey, 2007) can be illustrated on a basis of those of water and carbon dioxide (Figs. 1.3-1.6). Figures 1.3 and 1.4 show variations in the basic thermophysical properties of water at the critical (P_{cr} = 22.064 MPa) and three supercritical pressures (P = 25.0, 30.0, and 35.0 MPa) (also, in addition see Fig. 1.5) and those of carbon dioxide at the equivalent pressures to those of water (the conversion is based on $\left(\frac{P}{P_{cr}}\right)_{H_2O} = \left(\frac{P}{P_{cr}}\right)_{CO_2}$). 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). Critical parameters of selected fluids are listed in Table 1.1.

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 the critical and pseudocritical regions. Near the critical point, these changes are dramatic (see Figures 1.3-1.5). In the vicinity of pseudocritical points, with an increase in pressure, these changes become less pronounced (see Figs. 1.3, 1.4 and 1.7).

Table 1.1. Critical parameters of selected fluids (Pioro and Duffey, 2007).

Fluid	P_{cr}, MPa	T_{cr}, °C	ρ_{cr}, kg/m^3
Carbon dioxide (CO_2)	7.3773	30.978	467.6
Freon-134a (1,1,1,2-tetrafluoroethane, CH_2FCF_3)	4.0593	101.06	511.9
Water (H_2O)	22.064	373.95	322.39

Also, it can be seen that 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 Figs. 1.3a,b, 1.4a,b and 1.5), while the kinematic viscosity and specific enthalpy undergo a sharp increase (see Figs. 1.3d,g, 1.4d,g and 1.5). The volume expansivity, specific heat, thermal conductivity and Prandtl number have peaks near the critical and pseudocritical points (see Figs. 1.3c,e,f,h, 1.4c,e,f,h, 1.5 and 1.6). The magnitude of these peaks decreases very quickly with an increase in pressure (see Fig. 1.7). 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. 1.3b,d,f, and 1.4b,d,f).

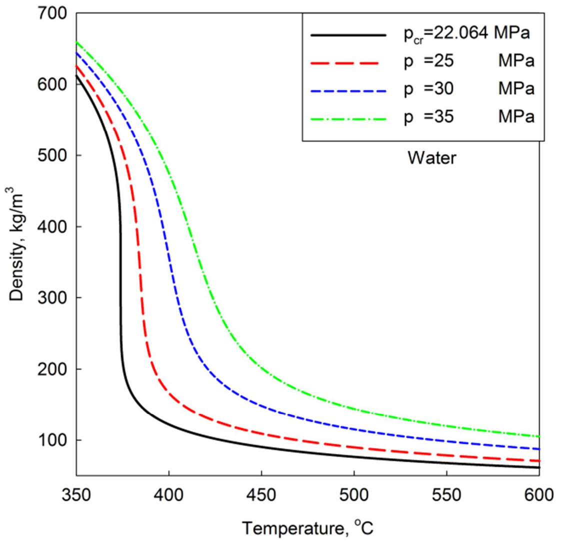

Figure 1.3a. Density vs. Temperature: Water.

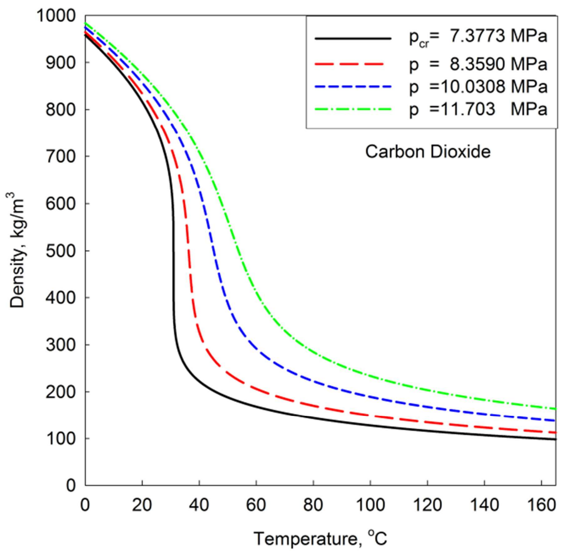

Figure 1.4a. Density vs. Temperature: Carbon Dioxide.

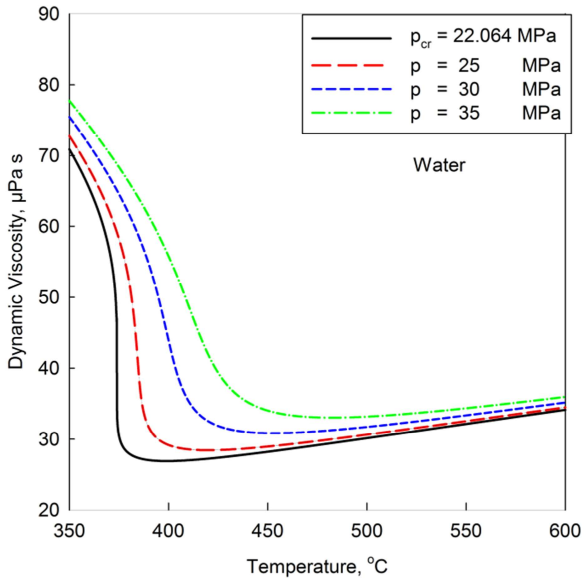

Figure 1.3b. Dynamic viscosity vs. Temperature: Water.

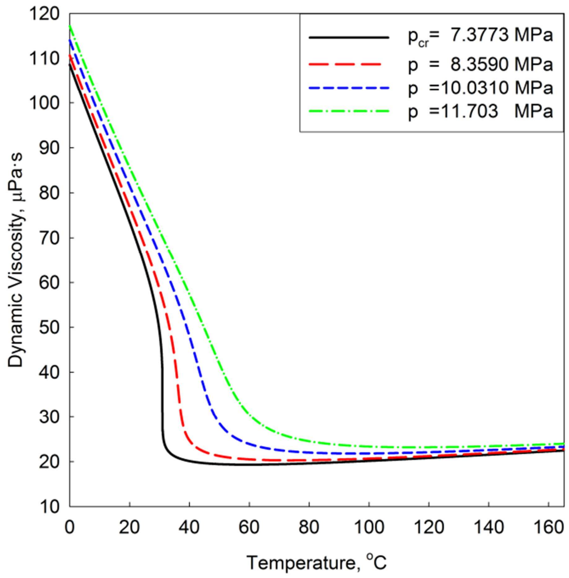

Figure 1.4b. Dynamic viscosity vs. Temperature: Carbon Dioxide.

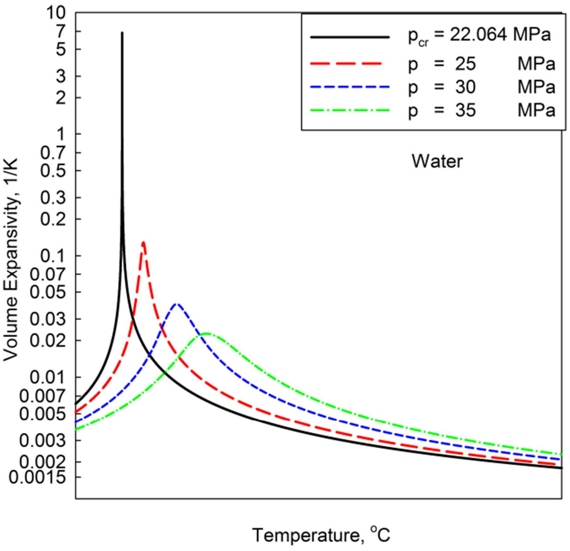

Figure 1.3c. Volume expansivity vs. Temperature: Water.

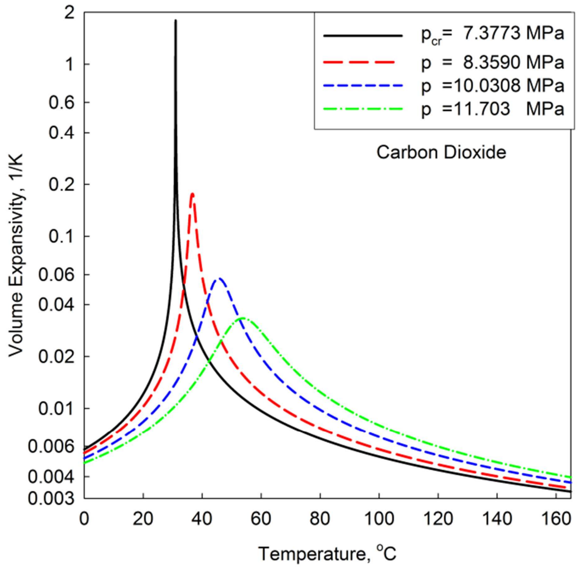

Figure 1.4c. Volume expansivity vs. Temperature: Carbon Dioxide.

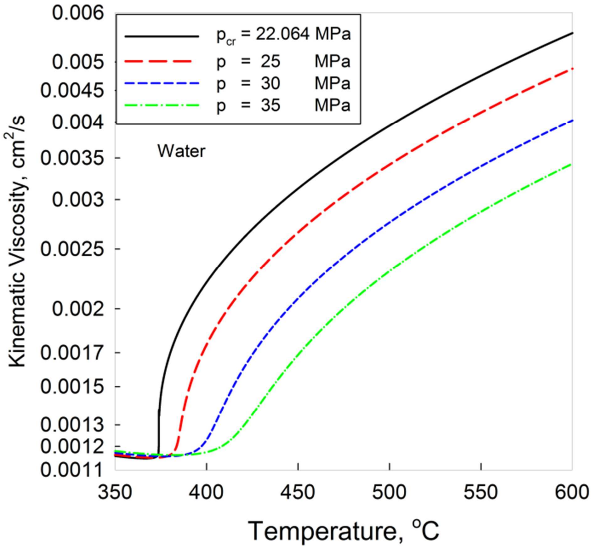

Figure 1.3d. Kinematic viscosity vs. Temperature: Water.

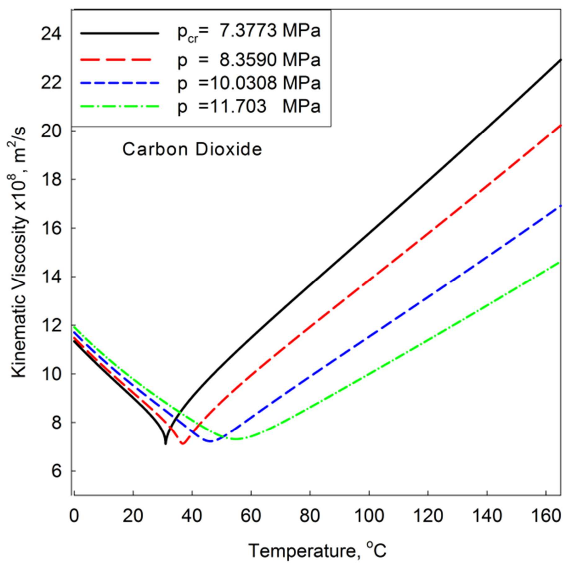

Figure 1.4d. Kinematic viscosity vs. Temperature: Carbon Dioxide.

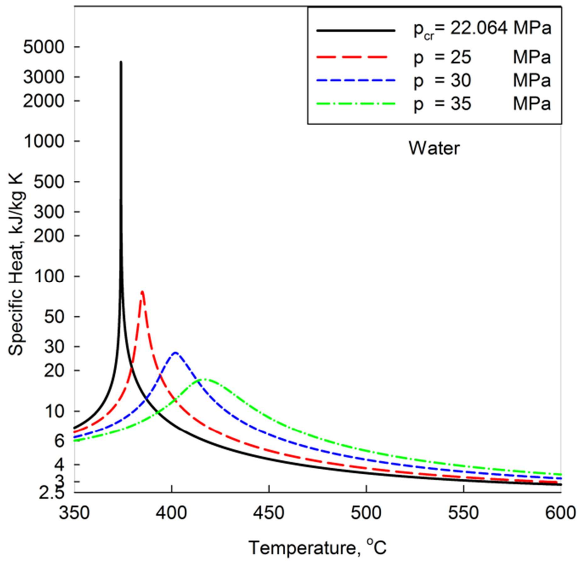

Figure 1.3e. Specific heat vs. Temperature: Water.

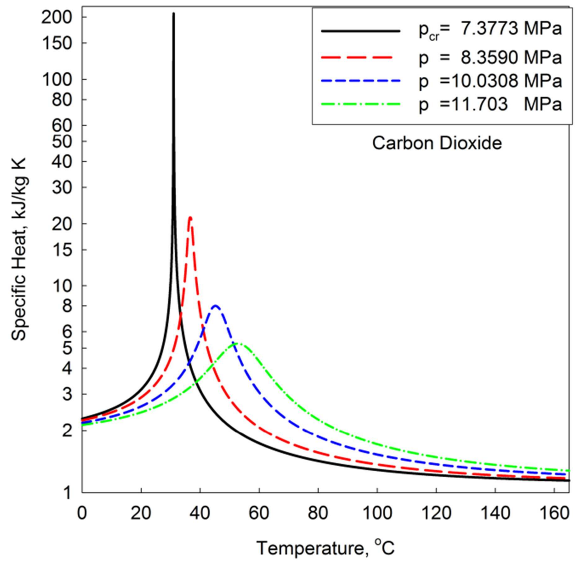

Figure 1.4e. Specific heat vs. Temperature: Carbon Dioxide.

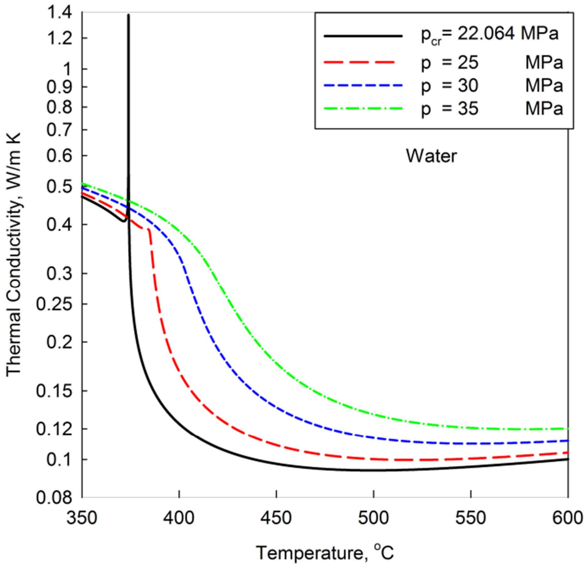

Figure 1.3f. Thermal conductivity vs. Temperature: Water.

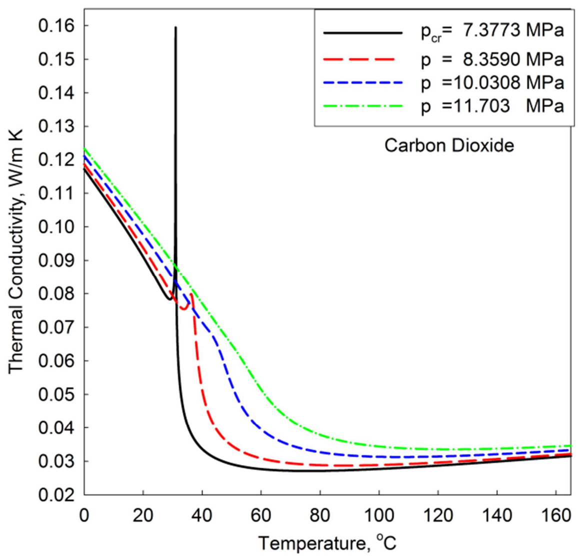

Figure 1.4f. Thermal conductivity vs. Temperature: Carbon Dioxide.

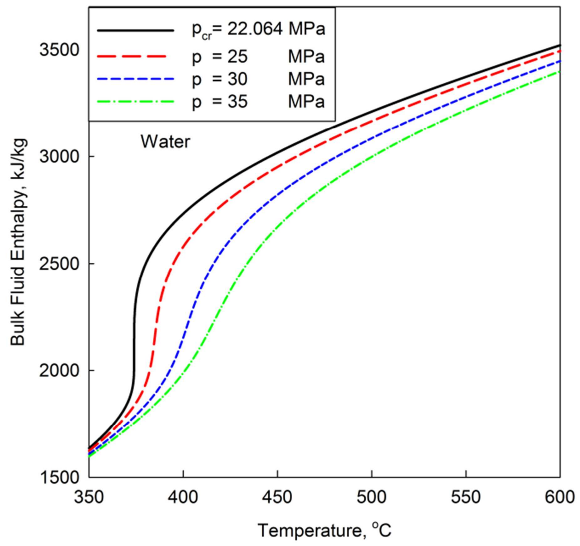

Figure 1.3g. Specific enthalpy vs. Temperature: Water.

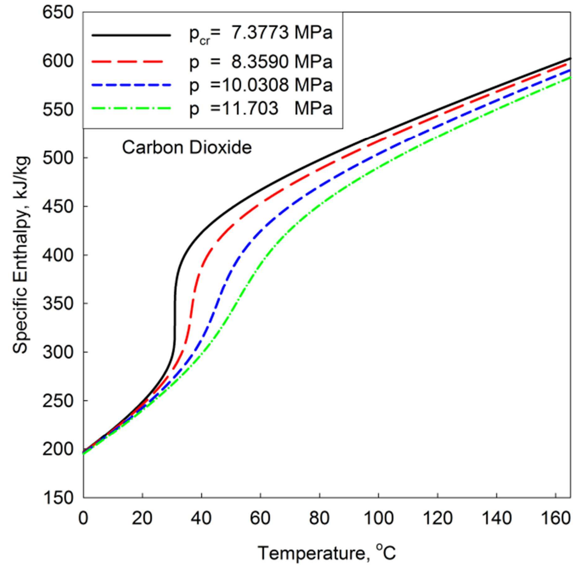

Figure 1.4g. Specific enthalpy vs. Temperature: Carbon Dioxide.

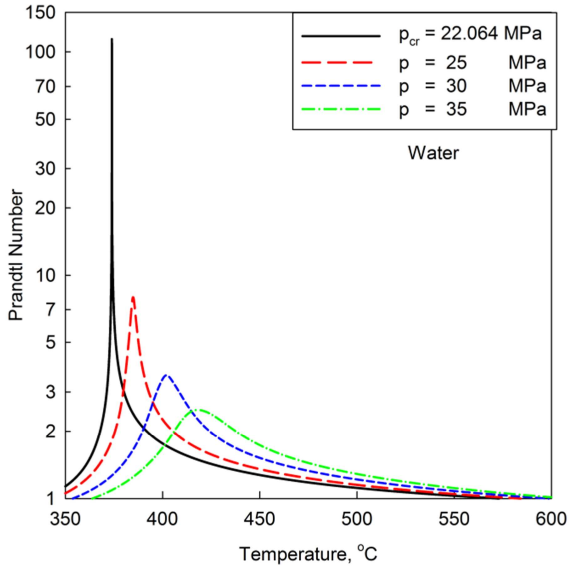

Figure 1.3h. Prandtl number vs. Temperature: Water.

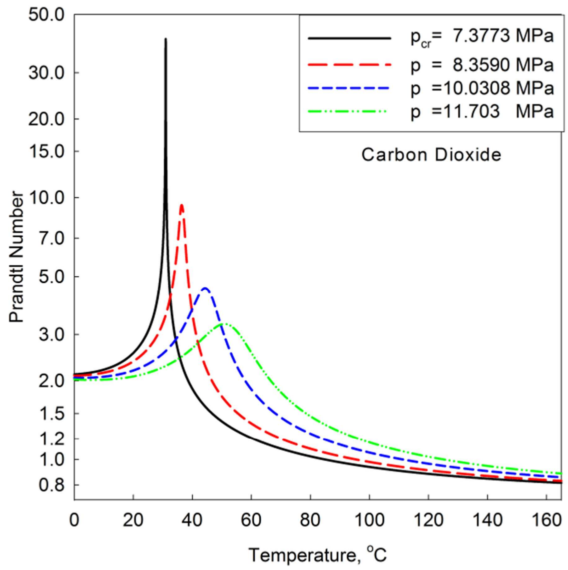

Figure 1.4h. Prandtl number vs. Temperature: Carbon Dioxide.

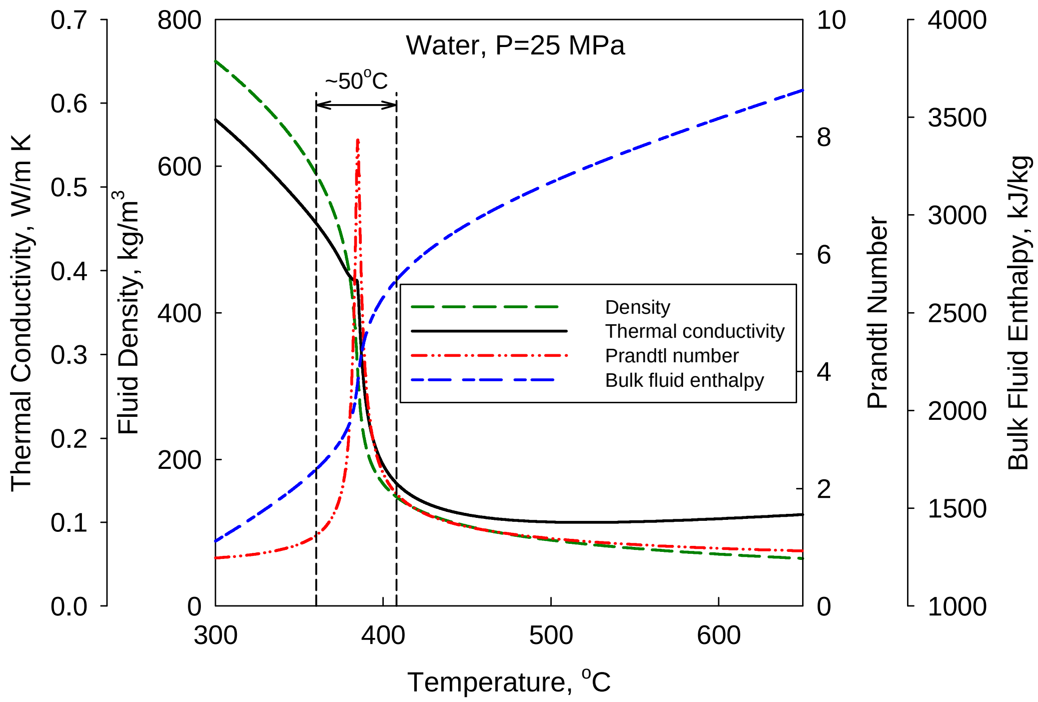

Figure 1.5. Variations of selected thermophysical properties of water near pseudocritical point: Pseudocritical region at 25 MPa is about ±25°C around pseudocritical point.

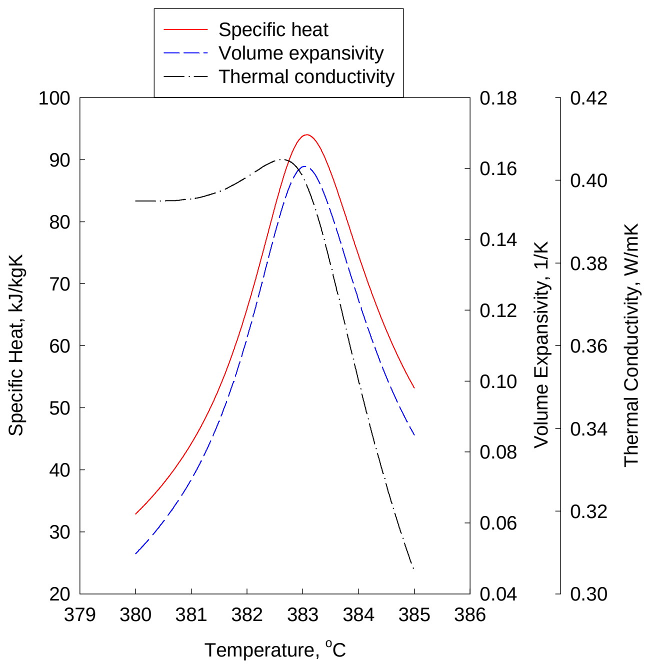

Figure 1.6. Specific Heat, Volume Expansivity and Thermal Conductivity vs. Temperature: Water, *P* = 24.5 MPa.

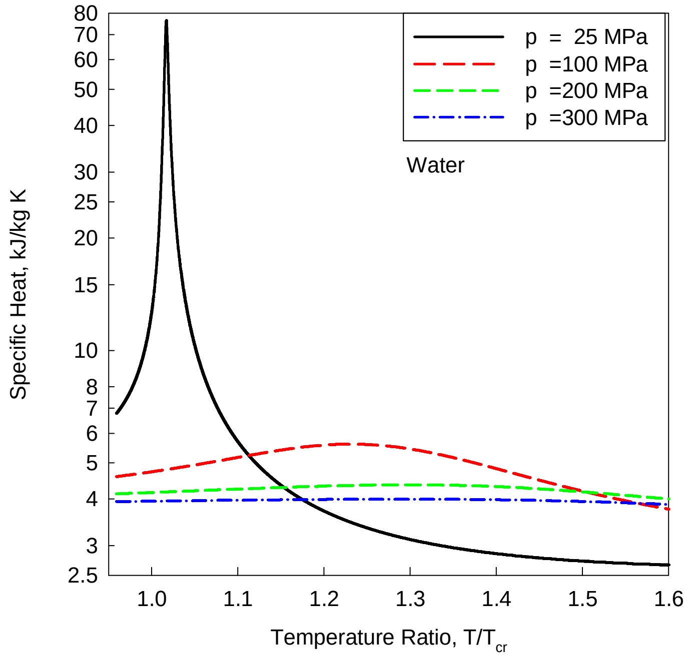

Figure 1.7. Specific heat variations at various pressures: Water.

Table 1.2. Values of pseudocritical temperature and corresponding peak values of specific heat within wide range of pressures (Pioro and Duffey, 2007).

Pressure, MPa	Pseudocritical temperature, °C	Peak value of specific heat, kJ/kg·K
23	377.5	284.3
24	381.2	121.9
25	384.9	76.4
26	388.5	55.7
27	392.0	43.9
28	395.4	36.3
29	398.7	30.9
30	401.9	27.0
31	405.0	24.1
32	408.1	21.7
33	411.0	19.9
34	413.9	18.4
35	416.7	17.2

The specific heat of water (see Fig. 1.3e) (as well as of other fluids, for example, for carbon dioxide, see Fig. 1.4e) has a 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. 1.1 and Table 1.2). At pressures approximately above 300 MPa (see Fig. 1.7) a peak (here it is better to say "a hump") in specific heat almost disappears, therefore, such term as a *pseudocritical point* no longer exists. 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 (see Table 1.3 and Figure 1.6).

Table 1.3. Peak values of specific heat, volume expansivity and thermal conductivity in critical and near pseudocritical points (Pioro and Duffey, 2007).

Pressure, MPa	Pseudocritical temperature, °C	Temperature, °C	Specific heat, kJ/kg·K	Volume expansivity, 1/K	Thermal conductivity, W/m·K
p_{cr}=22.064	t_{cr}=374.1	–	∞	∞	∞
22.5	375.6	–	690.6	1.252	0.711
23.0	–	377.4	–	–	0.538
	377.5	–	284.3	0.508	–
23.5	–	379.2	–	–	0.468
	–	379.3	–	0.304	–
	379.4	–	171.9	–	–
24.0	–	381.0	–	–	0.429
	381.2	–	121.9	0.212	–
24.5	–	382.6	–	–	0.405
	–	383.0	–	0.161	–
	383.1	–	93.98	–	–
25.0	–	384.0	–	–	0.389
	384.9	–	76.44	–	–
	–	385.0	–	0.128	–
25.5	386.7	–	64.44	0.107	no peak
26.0	388.5	-	55.73	0.090	0.355
27.0	392.0	-	43.93	0.069	0.340
28.0	395.4	-	36.29	0.056	0.329
29.0	398.7	-	30.95	0.046	0.321
30.0	401.9	-	27.03	0.039	0.316

In early studies, i.e., approximately before 1990, a peak in thermal conductivity was not taken into account. Later, this peak was well established (see Figs. 1.3f and 1.4f) and included into thermophysical data and software. The peak in thermal conductivity diminishes at about 25.5 MPa for water (see Fig. 1.3f and Table 1.3) and at about 8.4 MPa for carbon dioxide (see Fig. 1.4f).

In general, crossing the pseudocritical line from left to right (see Fig. 1.1) 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 Fig. 1.5). On the contrary, at subcritical pressures there are properties discontinuation on the saturation line: one value for liquid and another for vapour (see Fig. 1.8). Therefore, supercritical fluids behave as single-phase substances. Also, dealing with supercritical fluids we apply usually a term “*pseudo*” in front of a *critical point*, *boiling*, *film boiling*, etc.

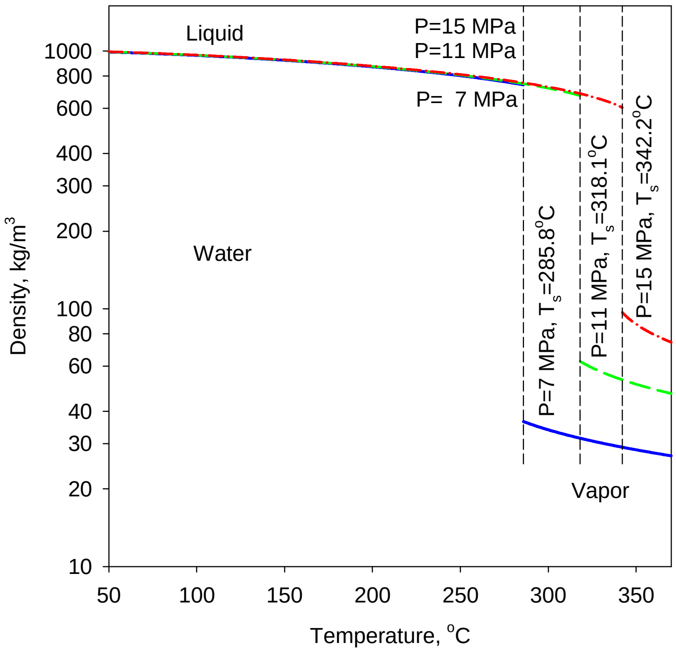

Figure 1.8. Density variations at various subcritical pressures for water: Liquid and vapour.

Properties of supercritical helium and R-134a are shown in Pioro and Duffey (2007).

PART 2: HISTORY OF SUPERCRITICAL PRESSURES APPLICATION IN POWER INDUSTRY

2.1. INTRODUCTION

The use of supercritical fluids in different processes is not new and, actually, is not a human invention. Nature has been processing minerals in aqueous solutions at near or above the critical point of water for billions of years (Levelt Sengers 2000). In the late 1800s, scientists started to use this natural process in their labs for creating various crystals. During the last 50 – 60 years, this process, called hydrothermal processing (operating parameters: water pressure from 20 to 200 MPa and temperatures from 300 to 500ºC), has been widely used in the industrial production of high-quality single crystals (mainly gem stones) such as sapphire, tourmaline, quartz, titanium oxide, zircon and others (Levelt Sengers 2000).

The first works devoted to the problem of heat transfer at supercritical pressures started as early as the 1930s (Pioro and Pioro 1997; Hendricks et al. 1970). E. Schmidt and his associates investigated free convection heat transfer of fluids at the near-critical point with the application to a new effective cooling system for turbine blades in jet engines. They found (Schmidt 1960; Schmidt et al. 1946) that the free convection heat transfer coefficient (HTC) at the near-critical state was quite high and decided to use this advantage in single-phase thermosyphons with an intermediate working fluid at the near-critical point (Pioro and Pioro 1997). (In general, thermosyphons are used to transfer heat flux from a heat source to a heat sink located at some distance.)

In the 1950s, the idea of using supercritical steam-water appeared to be rather attractive for “steam” generators. At supercritical pressures, there is no liquid-vapour phase transition; therefore, there is no such phenomenon as critical heat flux or dryout. Only within a certain range of parameters a deterioration of heat transfer may occur. The objective of operating “steam” generators at supercritical pressures was to increase the total efficiency of a power plant. Work in this area was mainly done in the USA and former USSR in the 1950s – 1980s (International Encyclopedia of Heat & Mass Transfer 1998).

At the end of the 1950s and the beginning of the 1960s, some studies were conducted to investigate the possibility of using supercritical fluids in nuclear reactors (Oka 2000; Wright and Patterson 1966; Bishop et al. 1962; Skvortsov and Feinberg 1961; Marchaterre and Petrick 1960; Supercritical pressure power reactor 1959). Several designs of nuclear reactors using water as the coolant at supercritical pressures were developed in the USA and USSR. However, this idea was abandoned for almost 30 years and regained support in the 1990s.

Use of supercritical water in power-plant “steam” generators is the largest application of a fluid at supercritical pressures in industry. However, other areas exist where supercritical fluids are used or will be implemented in the near future.

The latest developments in these areas focus on:

- increasing the efficiency of the existing ultra-supercritical and supercritical “steam” generators (Smith 1999);

- developing supercritical water-cooled nuclear reactors (Kirillov 2001a,b; Oka 2000);
- using supercritical water in the Rankine cycle for lead-cooled nuclear reactors (Boehm et al. 2005) and in the Brayton cycle (Sohn et al. 2005) including the Brayton cycle for future Sodium Fast Reactors (SFRs);
- using supercritical carbon dioxide in an indirect cycle of the gas cooled fast reactors (Hejzlar et al. 2005; Kato et al. 2005);
- using supercritical carbon dioxide for cooling of a printed circuit (Ishizuka et al. 2005);
- the use of near-critical helium to cool the coils of superconducting electromagnets, superconducting electronics and power-transmission equipment (Hendricks et al. 1970a);
- the use of supercritical hydrogen as a fuel for chemical and nuclear rockets (Hendricks et al. 1970a);
- the use of methane as a coolant and fuel for supersonic transport (Hendricks et al. 1970a);
- the use of liquid hydrocarbon coolants and fuels at supercritical pressures in the cooling jackets of liquid rocket engines and in fuel channels of air-breathing engines (Altunin et al. 1998; Kalinin et al. 1998, Dreitser 1993, Dreitser et al. 1993);
- the use of supercritical carbon dioxide as a refrigerant in air-conditioning and refrigerating systems (Lorentzen 1994; Lorentzen and Pettersen 1993);
- the use of a supercritical cycle in the secondary loop for transformation of geothermal energy into electricity (Abdulagatov and Alkhasov 1998);
- the use of supercritical water oxidation technology (SCWO) for treatment of industrial and military wastes (Levelt Sengers 2000; Lee 1997);
- the use of carbon dioxide in the supercritical fluid leaching (SFL) method for removal uranium from radioactive solid wastes (Tomioka et al. 2005) and in decontamination of surfaces (Shadrin et al. 2005); and
- the use of supercritical fluids in chemical and pharmaceutical industries in such processes as supercritical fluid extraction, supercritical fluid chromatography, polymer processing and others (Supercritical Fluids 2002; Levelt Sengers 2000).

Experiments at supercritical pressures are very expensive and require sophisticated equipment and measuring techniques (Pioro and Duffey, 2007). Therefore, some of these studies (for example, heat transfer in bundles) are proprietary and hence were not published in the open literature.

The majority of the studies deal with heat transfer and hydraulic resistance of working fluids, mainly *water* (Pioro and Duffey 2005, 2003a; Pioro et al. 2004), *carbon dioxide* (Duffey and Pioro 2005b), and *helium, in circular tubes*. In addition to these fluids, forced- and free-convection heat transfer experiments were conducted at supercritical pressures, using:

- *liquefied gases* such as air and argon (Budnevich and Uskenbaev 1972), hydrogen (International Encyclopedia of Heat & Mass Transfer 1998; Hess and Kunz 1965; Thompson and Geery 1962); nitrogen (Popov et al. 1977; Akhmedov et al. 1974; Uskenbaev and Budnevich 1972), nitrogen tetra-oxide (Nesterenko et al. 1974; McCarthy et al. 1967), oxygen (Powel 1957) and sulphur hexafluoride (Tanger et al. 1968);
- *alcohols* such as ethanol and methanol (Kafengauz 1983; Alad'yev et al. 1967, 1963);
- *hydrocarbons* such as n-heptane (Isayev 1983; Alad'ev et al. 1976; Kaplan and Tolchinskaya 1974a, 1971), n-hexane (Isaev et al. 1995), di-iso-propyl-cyclo-hexane

(Kafengauz 1983, 1969, 1967; Kafengauz and Fedorov 1970, 1968, 1966), n-octane (Yanovskii 1995), iso-butane, iso-pentane and n-pentane (Abdulagatov and Alkhasov 1998; Bonilla and Sigel 1961);
- *aromatic hydrocarbons* such as benzene and toluene (Rzaev et al. 2003; Abdullaeva et al. 1991; Kalbaliev et al. 1983, 1978; Isaev and Kalbaliev 1979; Mamedov et al. 1977; 1976), and poly-methyl-phenyl-siloxane (Kaplan et al. 1974b);
- *hydrocarbon coolants* such as kerosene (Kafengauz 1983), TS-1 and RG-1 (Altunin et al. 1998), jet propulsion fuels RT and T-6 (Kalinin et al. 1998; Yanovskii 1995; Valueva et al. 1995; Dreitser et al. 1993); and
- *refrigerants* (Abdulagatov and Alkhasov 1998; Gorban' et al. 1990; Tkachev 1981; Beschastnov et al. 1973; Nozdrenko 1968; Holman and Boggs 1960; Griffith and Sabersky 1960).

A limited number of studies were devoted to heat transfer and pressure drop in annuli, rectangular-shaped channels and bundles.

2.2. SUPERCRITICAL THERMAL POWER PLANTS: REVIEW AND STATUS

2.2.1. Russian Supercritical Units

Early studies in Russia related to the heat transfer at supercritical pressures started in the late 1940s. In the 1950s, Podol'sk Machine-Building plant "ЗиО" ("ZiO") (Plant by the name of S. Ordzhonikidze) manufactured several small experimental supercritical "steam" generators for research institutions such as: *(i)* "ВТИ" ("VTI") – All-Union Heat Engineering Institute (Moscow) with "steam" parameters of 29.4 MPa and 600ºC (Shvarts et al., 1963), *(ii)* "ЦКТИ" ("TsKTI") – Central Boiler-Turbine Institute by Polzunov (St.-Petersburg) and *(iii)* Kiev Polytechnic Institute with "steam" parameters of 39 MPa and 700ºC (Kirillov 2001).

The implementation of supercritical power-plant "steam" generators in Russia (the former USSR) started with units having thermal powers of 300 MW (Ornatskiy et al. 1980). Two leading Russian manufacturing plants: "ТКЗ" ("TKZ") – Taganrog Power-Plant Steam Generator's Manufacturing plant (Taganrog, Ukraine) and "ЗиО" (Podol'sk, Russia) developed and manufactured the first units, with the assistance of research institutions such as "ЦКТИ" and "ВТИ". Supercritical "steam" generators are usually the once–through type boilers (Belyakov 1995).

Power-plant "steam" generator ТПП-110 (TPP-110) manufactured at "ТКЗ" in 1961 was the first industrial unit operating at supercritical conditions in the former USSR, and was used at coal-fired power plant. Its design included a liquid slug drain. A total of six units were put into operation.

Also, a power-plant "steam" generator (model ПК-39 (PK-39)) was built at "ЗиО" in 1961. Next year, "ЗиО" designed a new unit, ПК-41, to work with residual fuel oil and natural gas. Later, in 1964 and 1967, upgraded designs of ТПП-110 (units ТПП-210 and ТПП-210А) were developed and manufactured. In these units, it was decided to decrease the temperature of the primary "steam" from 585 to 565ºC.

Based on industrial experience, several upgraded designs were manufactured at "ТКЗ" (units ТГМП-144 (TGMP-144) for residual fuel oil and natural gas, ТПП-312 (1970) for coal, ТПП-314 (1970) for residual fuel oil and natural gas, and ТГМП-144 (1971) for residual fuel oil and natural gas with pressurized combustion chamber) and "ЗиО" (units ПК-50 (1963) for coal, ПК-59 (1972) for brown coal (lignite), and П-64 (P-64) (1977) for Yugoslavian lignites).

The 300-MW power-"steam"-generating units have the following characteristics:

• "steam" capacity, t/h	950–1000
• Pressure (primary "steam"), MPa	25
• Temperature (primary "steam"), °C	545–585
• Pressure (secondary steam), MPa	3.5–3.9
• Feed-water temperature, °C	260–265
• Thermal efficiency[1], %	88–93

The next stage in further development of supercritical "steam" generators involved an increase in their thermal capacity to 500 MW (units manufactured at "ЗиО": П-49 (1965) and П-57 (1972)) and 800 MW (units manufactured at "ТКЗ": ТПП-200 (1964), ТГМП-204 (1973) and ТГМП-324; unit manufactured at "ЗиО": П-67 (1976)).

The 500-MW power-"steam"-generating units have the following characteristics:

• "steam" capacity, t/h	1650
• Pressure (primary "steam"), MPa	25
• Temperature (primary "steam"), °C	545
• Pressure (secondary steam), MPa	3.95
• Temperature (secondary steam), °C	545
• Feed-water temperature, °C	277
• Thermal efficiency, %	92

The 800-MW power-"steam"-generating units have the following characteristics:

• "steam" capacity, t/h	2650
• Pressure (primary "steam"), MPa	25
• Temperature (primary "steam"), °C	545
• Pressure (secondary steam), MPa	3.44
• Temperature (secondary steam), °C	545
• Feed-water temperature, °C	275
• Thermal efficiency, %	92–95

In 1966, the first 1000-MW ultra-supercritical plant started its operation in Kashira with a primary "steam" pressure of 30.6 MPa, and primary and reheat temperatures of 650 and 565°C, respectively (Smith 1999).

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[1] This thermal efficiency is related only to a "steam"-generator. The total or overall efficiency of a power plant will be significantly less (43 – 50%) due to a number of energy converting devices: $\eta_{total} = \eta_{steam\ gen.}\ \eta_{turbine}\ \eta_{el.gen.} \dots$ In other words, the total or overall efficiency of a power plant is actually the ratio of net electrical power output to the rate of fuel energy input.

In modern designs of supercritical units, the thermal capacity was upgraded to 1200 MW (unit manufactured at "ТКЗ": ТГМП-1204 (1978), "steam" generating capacity of 3950 t/h (Ornatskiy et al. 1980). This, one of the largest in the world, supercritical power-generating unit operates with a single-shaft turbine at the Kostroma district power plant and is a gas-oil-fired "steam" generator (Belyakov 1995).

Over the last 25 years, more than 200 supercritical units were manufactured and put into operation in Russia (Smith 1999). Supercritical "steam" generators manufactured in Russia are listed in Table 2.1 and 2.2.

Table 2.1. Supercritical "steam" generators manufactured in Russia (Belyakov 1995).

Capacity, MW	Manufacturer				Total
	"ТКЗ" (Taganrog)		"ЗиО" (Podol'sk)		
	gas-oil	coal	gas-oil	coal	
300	91	49	19	36	195
500	–	–	–	16	16
800	17	2	–	1	20
1200	1	–	–	–	1
In all	109	51	19	53	–
	160		72		232

Figure 2.1 shows modern single-reheat-cycle 600-MW_{el} Tom'-Usinsk (Russia) thermal power plant layout.

Table 2.2. Parameters of largest Russian SC 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 HP Cylinders	1	1	-
No. of IP Cylinders	2	2	-
No. of 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
Turbine Specific Performance			
Specific Heat Rate, kJ/kW·h	7660	7720	7590

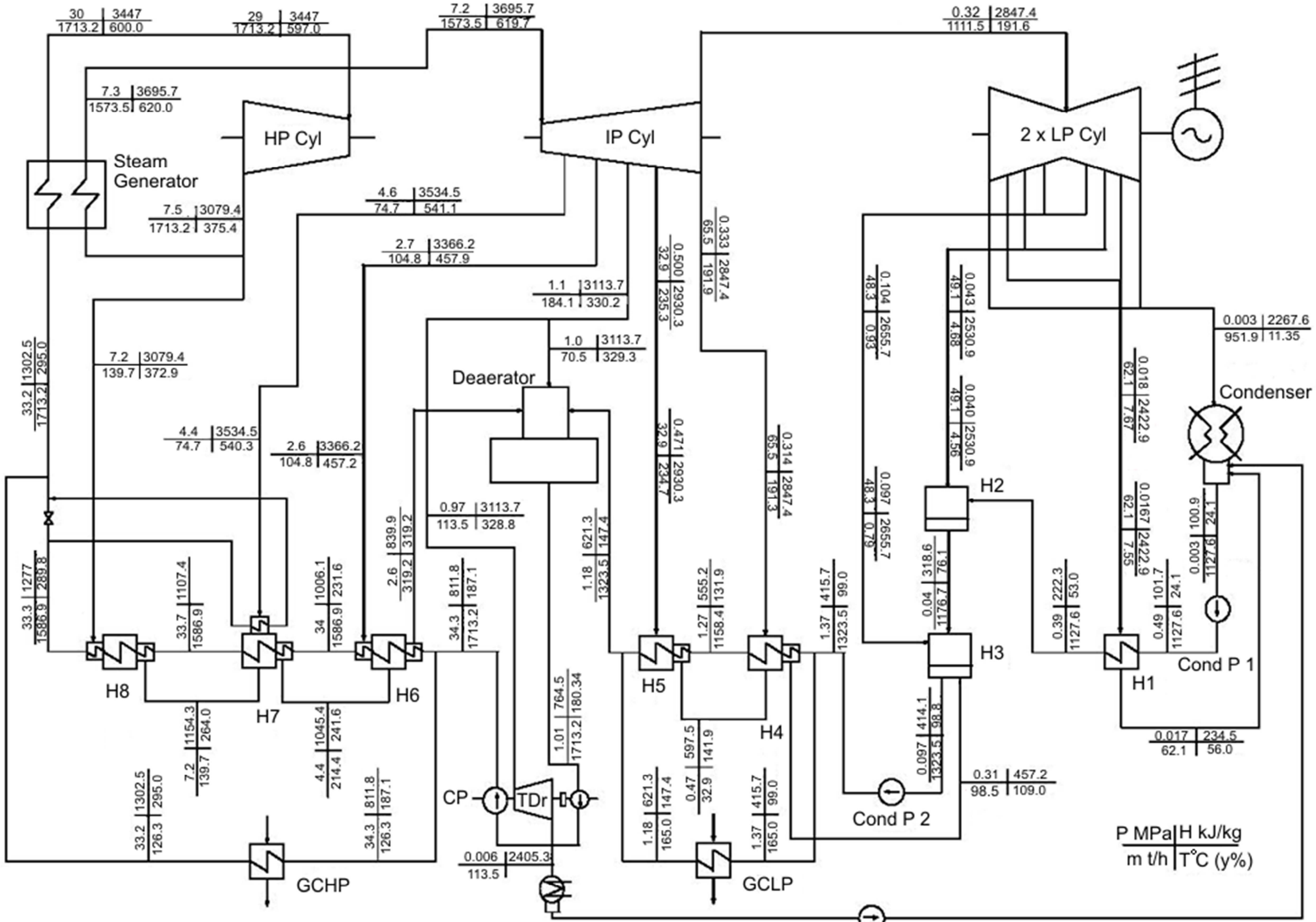

Figure 2.1. Single-reheat-cycle 600-MW$_{el}$ Tom'-Usinsk thermal power plant (Russia) thermal layout (Kruglikov et al., 2009): Cyl – Cylinder; H – Heat exchanger (feedwater heater); CP – Circulation Pump; TDr – Turbine Drive; Cond P – Condensate Pump; GCHP – Gas Cooler of High Pressure; and GCLP – Gas Cooler of Low Pressure.

2.2.2. US Supercritical-Pressure Units

In the early 1950s, the development work on supercritical "steam" generators started in the USA (Lee and Haller 1974). The first supercritical "steam" generator was put into operation at the Philo Plant of American Electric Power in 1957. The capacity of this unit was 120 MW with "steam" parameters of 31 MPa and 620/566/538ºC (main/reheat/reheat) (Retzlaff and Ruegger 1996).

Later on, supercritical power-plant "steam" generators in the USA were developed, manufactured and put into operation with a "steam" generating capacity of 500 MW (1961) (Ornatskiy et al. 1980).

In the early sixties, another plant was built with ultra-supercritical "steam" parameters (pressure of 30 MPa, temperatures (primary and reheat) of 650ºC) (Smith 1999).

Major USA manufacturers of power-plant "steam" generators such as Babcock & Wilcox, Combustion Engineering, Inc., Foster Wheeler, and others were involved in the development and manufacturing of the supercritical units. The supercritical units found their application at thermal capacities from 400 to 1380 MW. Often the subcritical units for 1000 MW and higher were replaced with supercritical "steam" generators in the USA (Ornatskiy et al. 1980).

US power "steam"-generating units have the following averaged characteristics (Ornatskiy et al. 1980):

- "Steam" capacity, t/h 1110–4440
- Pressure (primary "steam"), MPa 23–26
- Temperature (primary "steam"), ºC 538–543
- Temperature (secondary steam), ºC 537–551

The characteristics of two supercritical units built by "Babcock & Wilcox" are listed below (Ornatskiy et al. 1980).

Power-plant "steam" generator put into operation at the "Paradise" power plant (USA) in 1969 (for 1130 MW unit):

- "Steam" capacity, t/h 3630
- Pressure (primary "steam"), MPa 24.2
- Temperature (primary "steam"), ºC 537
- Steam capacity (secondary steam), t/h 2430
- Pressure (secondary steam), MPa 3.65
- Temperature (secondary steam), ºC 537
- Feed-water temperature, ºC 288
- Thermal efficiency, % 89

Power-plant "steam" generators put into operation at the "Emos" (1973) and "Gevin" (1974 – 1975) power plants (USA) (for 1130 MW units):

- "steam" capacity, t/h 4438
- Pressure (primary "steam"), MPa 27.3

- Temperature (primary "steam"), ºC 543
- Steam capacity (secondary steam), t/h 3612
- Pressure (secondary steam), MPa 4.7
- Temperature (secondary steam), ºC 538
- Feed-water temperature, ºC 291
- Thermal efficiency, % 93

The largest supercritical units are rated up to 1300 MW with "steam" parameters of 25.2 MPa and 538ºC (Lee and Haller 1974).

2.2.3. Recent Developments in Supercritical "Steam" Generators/Turbines around the World

Recently supercritical power-plant "steam" generators are working around the world with a wide range of "steam" parameters (see Table 2.3).

Table 2.3. Characteristics of modern supercritical "steam" generators (Smith 1999).

Country	"Steam" parameters					
	Capacity	Primary		Reheat		Feed water
	t/h	*p*, MPa	*t*, ºC	*p*, MPa	*t*, ºC	*t*, ºC
China	–	25	538	–	566	–
Denmark	–	30	580	7.5	600	320
Germany	2420	26.8	547	5.2	562	270
Japan*	350–1000	24.1	538	–	566	–
		25	600	–	610	300
		31.1	566	–	566	–

* updated with recent data.

On average, the usage of supercritical "steam" generators instead of subcritical ones increased overall power plant efficiency from 45% to about 50% (Smith 1999).

In Japan, the first supercritical "steam" generator (600 MW) was commissioned in 1967 at the Anegasaki plant (Oka and Koshizuka 2002; Tsao and Gorzegno 1981). Nowadays, many power plants are equipped with supercritical "steam" generators and turbines. Hitachi operating supercritical pressure "steam" turbines have the following average parameters (see also Table 2.4): output – 350 (1 unit), 450 (2 units), 500 (3 units), 600 (11 units), 700 (4 units) and 1000 MW (4 units), "steam" pressure about 24.1 MPa (one unit 24.5 MPa), "steam" temperature (main/reheat) – 538/566ºC (the latest units 600/600ºC (610°C)).

In Germany, at the end of the nineties construction was started on Unit "K" of RWE Energie's Niederauβem lignite-burning power station near Cologne (Heitmüller et al. 1999). This power plant is described as the most advanced lignite-fired power plant in the world with 45.2% planned thermal efficiency. At a later date, with new dry lignite technology introduced, a further increase in efficiency of 3 – 5% is expected. The new Unit "K" will have the following parameters: output of about 1000 MW and “steam” conditions of 27.5 MPa and 580/600ºC (main/reheat).

In Denmark (Noer and Kjaer 1998), the first supercritical power plant started operation in 1984, and today a total of seven supercritical units are in operation. Main parameters of these units are: output – 2 units 250 MW, the rest 350 – 390 MW, “steam” pressure 24.5 – 25 MPa, “steam” temperature 545 – 560ºC, reheat temperature 540 – 560ºC, feed-water temperature 260 – 280ºC and net efficiency 42 – 43.5%. Main parameters of ultra-supercritical units: “steam” pressure 29 – 30 MPa, “steam” temperature 580ºC, steam reheat temperature 580 – 600ºC, feed-water temperature 300 – 310ºC and net efficiency 49 – 53%.

So-called “Ultra-supercritical boilers” are now being researched and deployed world wide, particularly in Japan, Korea and China. Using double steam reheat and advanced high temperature blade materials, the turbine inlet temperature is being extended to 625°C at pressures of up to 34 MPa, with overall efficiencies then approaching 51 – 53%.

Table 2.4. Major parameters of selected Hitachi SC plants (turbines).

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
	677	25.5	566/566
	600	24.1	600/620
2008	1000	24.9	600/600
	887	24.1	566/593
	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
1992	700	24.1	538/566

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

*Double-reheat-cycle turbines.

An analysis of SC-turbine data (Naidin et al., 2009) based on the current review and materials presented by Pioro and Duffey (2007) showed that:

- The vast majority of the modern and upcoming SC turbines are single-reheat-cycle turbines;
- Major "steam" inlet parameters of these turbines are:
  The main or primary SC "steam" – P = 24 – 25 MPa and T = 540 – 600°C; and the reheat or secondary subcritical-pressure steam – P = 3 – 5 MPa and T = 540 – 620°C.
- Usually, the main "steam" and reheat-steam temperatures are the same or very close (for example, 566/566°C; 579/579°C; 600/600°C; 566/593°C; 600/620°C).
- Only very few double-reheat-cycle turbines were manufactured so far. The market demand for double-reheat turbines disappeared due to economic reasons after the first few units were built.

PART 3. SUPERCRITICAL WATER-COOLED NUCLEAR-REACTOR CONCEPTS: REVIEW AND STATUS

3.1. GENERAL CONSIDERATIONS

Concepts of nuclear reactors cooled with water at supercritical pressure were studied as early as the 1950s and 1960s in the USA and former USSR (Oka 2000; Wright and Patterson 1966; Bishop et al. 1962; Skvortsov and Feinberg 1961; Marchaterre and Petrick 1960; Supercritical pressure power reactor 1959). The main characteristics of the first concepts of SCWRs are listed in Table 3.1.

Table 3.1. First concepts of nuclear reactors cooled with supercritical water (Oka 2002; 2000).

Parameters	Company / reactor acronym (year)			
	Westinghouse		GE, Hanford	B & W
	SCR (1957)	SCOTT-R (1962)	SCR (1959)	SCFBR (1967)
Reactor type	Thermal	Thermal	Thermal	Fast
Pressure, MPa	27.6	24.1	37.9	25.3
Power, MW (thermal/electrical)	70/21.2	2300/1010	300/–	2326/980
Thermal efficiency, %	30.3	43.5	~40	42.2
Coolant temperature at outlet, ºC	538	566	621	538
Primary coolant flow rate, kg/s	195	979	850	538
Core height / diameter, m/m	1.52/1.06	6.1/9.0	3.97/4.58	–
Fuel material	UO_2	UO_2	UO_2	MOX
Cladding material	SS	SS	Inconel-X	SS
Rod diameter / pitch, mm/mm	7.62/8.38	–	–	–
Moderator	H_2O	Graphite	D_2O	–

Explanations to the table:
Acronyms: GE – General Electric; B & W – Babcock & Wilcox; SCR – SuperCritical Reactor; SCOTT-R – SuperCritical Once-Through Tube reactor; and SCFBR – SuperCritical Fast Breeder Reactor.

After a 30-year interval, the idea of developing nuclear reactors cooled with supercritical water became attractive as the ultimate development path for water-cooling. Several countries (Canada, Germany, Japan, Korea, Russia, USA and others) have started R&D work in that direction. However, none of these concepts is expected to be implemented in practice before 2015 – 2020.

The main objectives of using supercritical water in nuclear reactors are:
1) Increase the efficiency of modern nuclear power plants (NPPs) from 30 – 35% to about 45 – 50%; and
2) Decrease capital and operational costs and hence decrease electrical energy costs.

Supercritical water NPPs will have much higher operating parameters (see Figure 3.1: pressure about 25 MPa and outlet temperature up to 625ºC) compared to those of modern NPPs’, and a simplified flow circuit (see Figure 3.2), in which steam generators, steam dryers, steam separators, etc., can be reduced or eliminated. Also, higher supercritical water temperatures allow direct thermo-chemical or indirect electrolysis production of hydrogen at low cost, due to increased process efficiencies[2]. According to the IAEA (1999), the optimum required temperature is about 850ºC and the minimum required temperature is around 650 to 700ºC, well within modern materials capability.

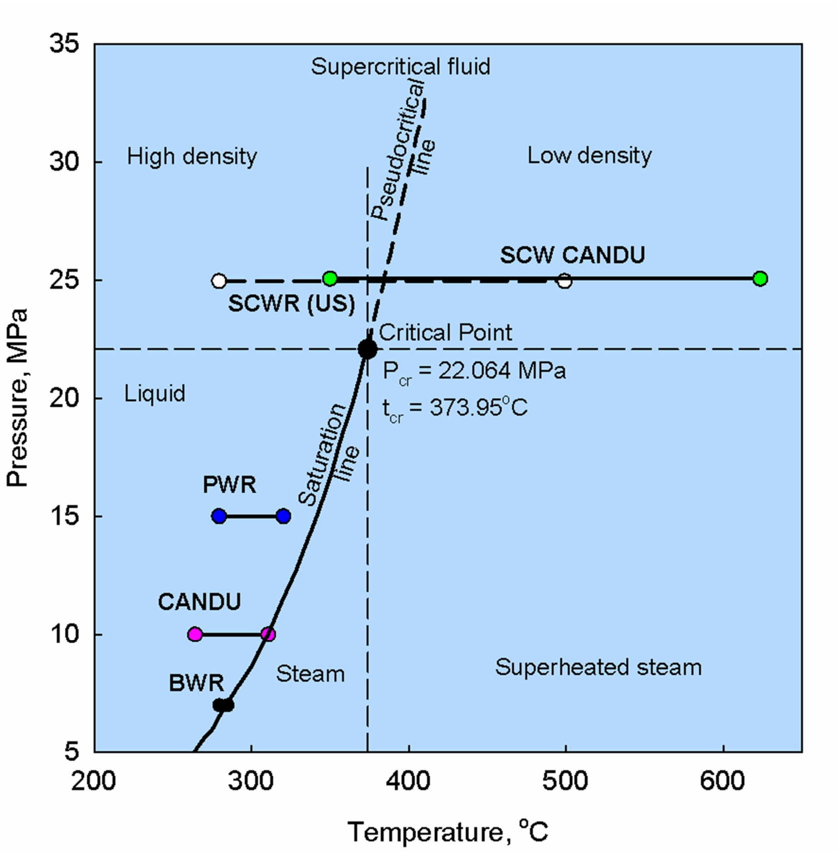

Figure 3.1. Pressure-temperature diagram of water with typical operating conditions of SCWRs, PWRs, CANDU-6 reactors and BWRs.

Also, future nuclear reactors will have high indexes of fuel usage in terms of thermal output per mass of fuel (Kirillov 2000; Alekseev et al. 1989). Therefore, changing over to supercritical pressures increases the thermal output coefficient and decreases the consumption of natural resources of uranium. Due to the considerable reduction in water density in the reactor core, it might be possible to develop fast SCWRs with a breeding factor of more than one for converting fertile (non-fissionable fuel) to fissionable isotopes in a self-sustaining fuel cycle.

3.2. DESIGN CONSIDERATIONS

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[2] IAEA TECDOC-1584 "Advanced Applications of Water Cooled NPPs" has information on hydrogen production with high-temperature electrolysis as well as other processes using 500°C heat.

The design of SCWRs is seen as the natural and ultimate evolution of today's conventional water-cooled nuclear reactors.

- First, some designs of the modern Pressurized Water Reactors (PWRs) operate at pressures of 15 − 16 MPa.
- Second, some Boiling Water Reactors (BWRs) are the once-through or direct-cycle design, i.e., steam from a nuclear reactor is forwarded directly into a turbine.
- Third, some experimental reactors used nuclear steam reheat with outlet steam temperatures well beyond the critical temperature, but at pressures below the critical pressure (DOE USA 2005; Grigor'yants et al. 1979; Baturov et al. 1978; Samoilov et al. 1976; Aleshchenkov et al. 1971; Dollezhal' et al. 1974, 1971, 1958; Kornbichler 1964; Margen 1961; Spalaris et al. 1961; Wallin et al. 1961). And
- Fourth, modern supercritical turbines, at pressures of about 25 MPa and inlet temperatures of about 600ºC, operate successfully at thermal coal-fired power plants for many years.

The SCWR concepts therefore follow two main types, the use of either (a) a large reactor pressure vessel (Figures 3.2 and 3.3) (Buongiorno and MacDonald 2003) with wall thickness of about 0.5 m to contain the reactor core (fuelled) heat source, analogous to conventional PWRs and BWRs, or (b) distributed pressure tubes or channels analogous to conventional CANDU®[3] and RBMK nuclear reactors (Duffey et al. 2006, 2005, 2003; Duffey and Pioro 2006, 2005a,b, 2004; Khartabil et al. 2005; Torgerson et al. 2005; Gabaraev et al. 2005, 2004, 2003a,b; Kuznetsov and Gabaraev 2004).

The pressure-vessel SCWR design is developed largely in the USA, EU, Japan (Oka et al. 2010), Korea and China and allows using a traditional high-pressure circuit layout.

The pressure-channel SCWR design is developed largely in Canada (Figure 3.4) and in Russia (Figure 3.5) to avoid a thick wall vessel, and allows, in principle, the following key features for safety and performance:

a) Passive accident and decay heat removal by radiation and convection from the distributed channels even with no active cooling and no fuel melting. Thus, the system is potentially inherently safe.
b) Use of multi-pass reactor flows, so that reheat and superheat are possible while still keeping the pressure tube cool. Thus, the system can produce process heat on demand. Reactor size (and thermal power) can be adjusted from 300 MW_e to 1400 MW_e depending on the customer site, financing and product mix application.

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[3] CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL).

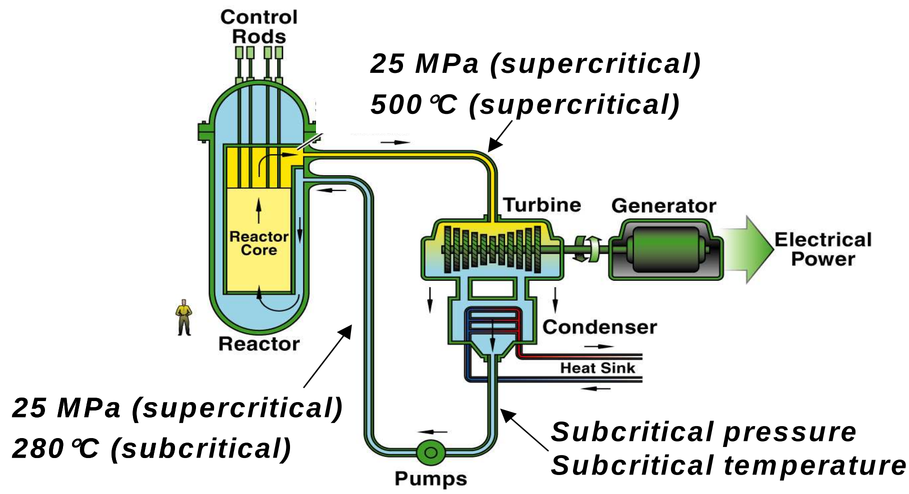
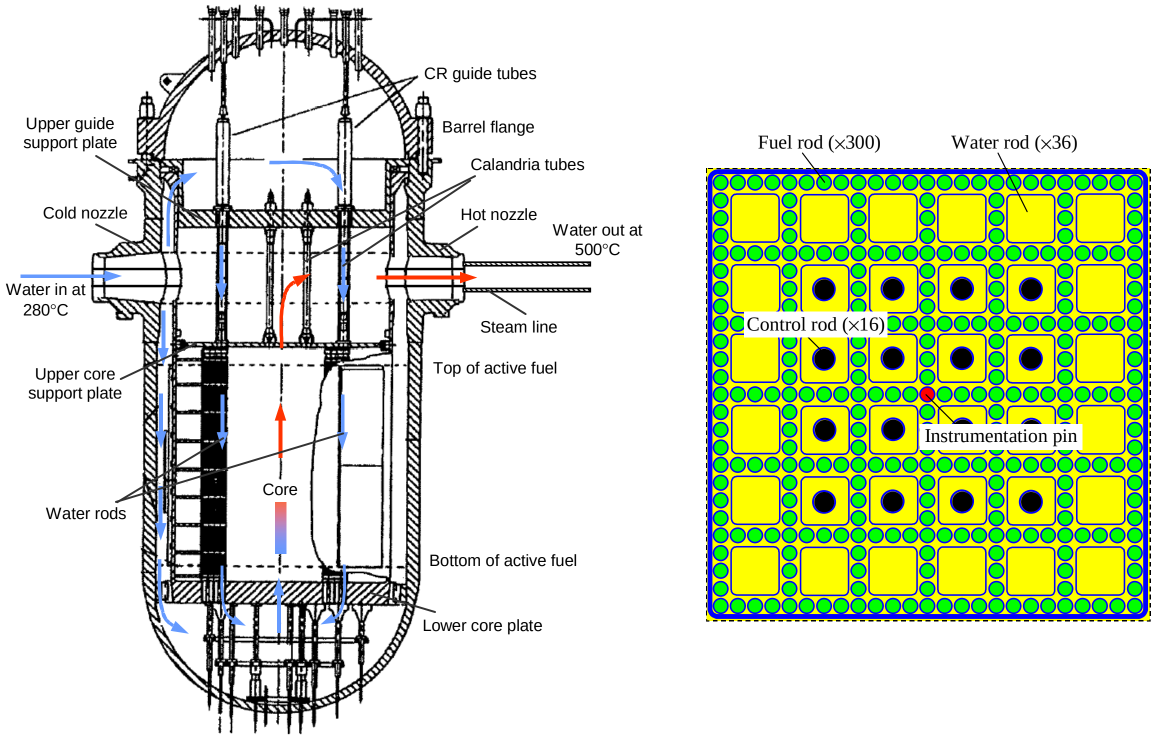


Figure 3.2. Schematics of US pressure-vessel SCWR (courtesy of Professor J. Buongiorno (MIT) (Buongiorno and MacDonald 2003)).



(a) **(b)**

Figure 3.3. Cross-sectional view of US SCWR pressure vessel (a) and reference US SCWR fuel assembly with water rods (b) (courtesy of Professor J. Buongiorno (MIT) (Buongiorno and MacDonald 2003)).

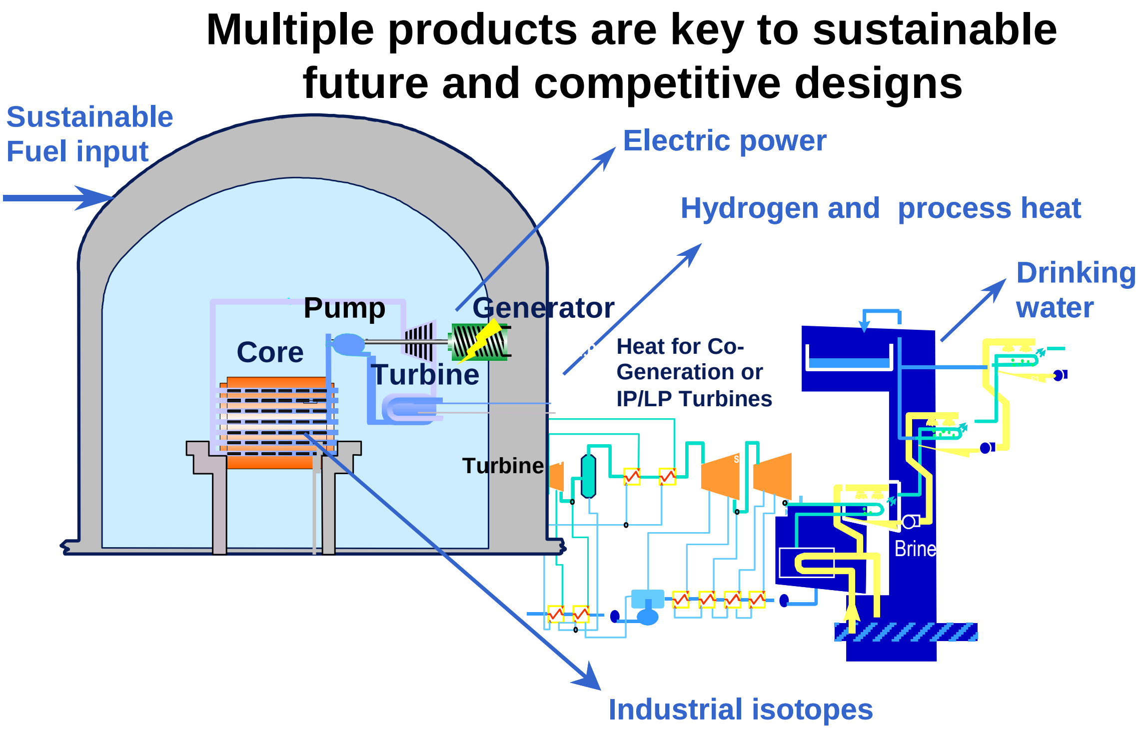


Figure 3.4. General scheme of pressure-channel SCW CANDU reactor (courtesy of Dr. R. Duffey (AECL)): IP – intermediate-pressure turbine and LP – low-pressure turbine.

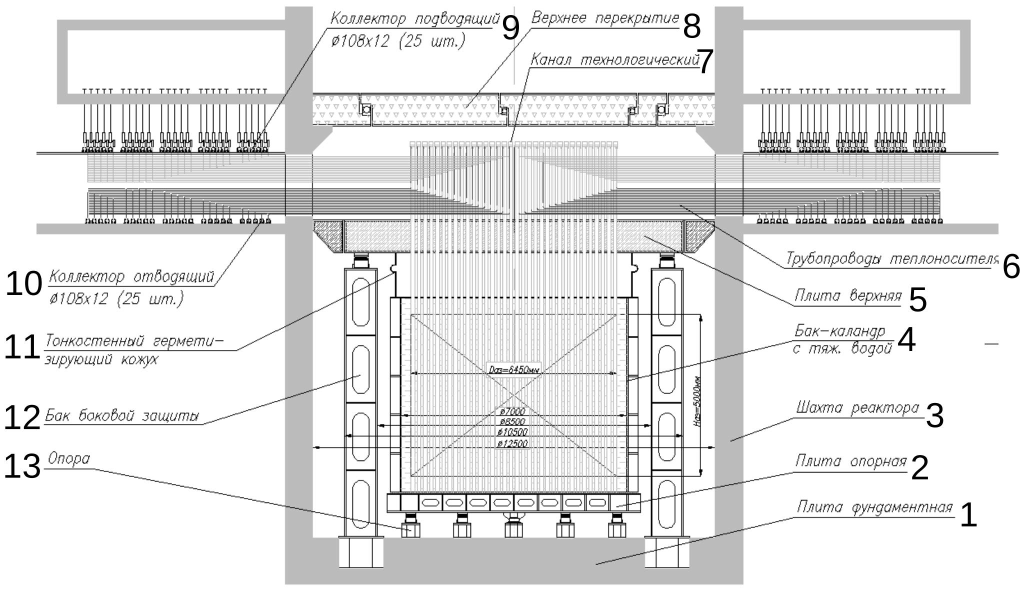


Figure 3.5. Layout of RDIPE pressure-channel nuclear reactor (Duffey et al. 2006; Gabaraev et al. 2005): 1) & 2) foundation and bearing plates, respectively; 3) reactor shaft; 4) calandria tank; 5) top plate; 6) coolant pipes; 7) technological channel; 8) top cover; 9) & 10) inlet and outlet collectors, respectively; 11) thin-wall sealing casing; 12) lateral shielding tank; and 13) supports.

The system features together set the fuel design, the channel power, the core lattice pitch, the enrichment, and the flow circuit parameters, where the coolant is usually water. A thermal neutron spectrum is used with either a solid moderator using graphite or zirconium hydride, or a liquid using heavy water moderator.

To reduce the severe axial flux tilt due to the large density decrease as the coolant is heated, the core flow path can be a re-entrant in the vessel option, coming down unheated first and then turning into an upflow; or interlaced or re-entrant in channels with flow in opposite directions. Both options allow the chance to reduce pressure boundary temperatures, by partly insulating the pressure-retaining vessel of the channel wall using the first pass of the unheated flow. Typical outlet temperatures are expected to be up to near 600ºC to match supercritical turbine inlet needs. There is also the option of a superheat pass (return flow) to further raise outlet temperatures if needed (for example, for hydrogen production).

The limit on supercritical water outlet temperature is effectively set by the fuel cladding, since the peak clad temperature will be some 20% higher than the average, and the corrosion rates much higher. Hence, the thermal limits depend on the wall heat transfer, and estimates of the peak values have been made to establish the margins and clad lifetime expected before refuelling.

Moreover, one of the unique features of the SCWRs is the very low coolant mass-flow rates that are required through the reactor core because of the high thermal capacity. Preliminary calculations showed that the rate could be about *five to eight times less* than in modern PWRs, significantly reducing the pumping power and costs. This improvement is due to the considerable increase in the specific enthalpy at supercritical conditions, of about 2000 kJ/kg.

Therefore, tightly packed cylindrical fuel bundles, which are more acceptable in SCWRs than in other types of reactors, can be used. These tight bundles have a significant pressure drop, which in turn can enhance the hydraulic stability of the flow. Since the supercritical water is a single-phase “gas”, then the cladding surfaces can and should be finned or ridged to enhance turbulence levels to increase the HTC. This is already done for Advanced Gas-cooled Reactors (AGRs) today, and hence will increase the heat transfer and reduce peak cladding temperatures in normal operation.

To optimize thermal efficiency and capital cost, there are also options for the thermal cycles (Spinks et al. 2002; Bushby et al. 2000a,b; Oka et al. 1996), with either a direct cycle into a supercritical turbine or an indirect using a heat exchanger. Analyses by Spinks et al. (2002) show a cost reduction of 15% or more for the direct-cycle option, without use of the process heat.

One advantage of separating the moderator and coolant in the pressure-channel design is that the moderator can act as a backup heat sink in the event when emergency core cooling is not available. The advanced fuel-channel design (Khartabil et al. 2005) combined with a passive-moderator cooling system (Khartabil 1998) results in a design where severe core damage is practically eliminated. In this design, the passive-moderator loop operates continuously to remove heat deposited in the moderator during normal operation. The moderator heat during normal operation is comparable to decay heat following reactor shutdown; therefore, the moderator can also be used to remove decay heat following postulated accidents.

In summary, the use of supercritical water in nuclear reactors will, according to the US DOE (Roadmap) Generation IV[4] Nuclear Energy Systems Report (2001):

- Significantly increase thermal efficiency from 33 – 35% up to 40 – 45%;
- Eliminate steam generators, steam separators, steam dryers and re-circulation pumps;
- Allow the production of hydrogen due to high coolant outlet temperatures;
- Decrease reactor coolant pumping power;
- Reduce frictional losses;
- Lower containment loadings; and
- Eliminate dryout.

The latest concepts of SCWRs are summarized in Table 3.2. Figure 3.2 shows the schematic of the US pressurized-vessel SCWR, Figure 3.4 – the general concept of the pressurized-channel SCW CANDU nuclear reactor and Figure 3.5 – the general scheme of the RDIPE pressurized-channel SCWR.

Specific features of SCWRs (see Table 3.2) being developed:

- in Canada are listed in Duffey et al. 2006, 2005, 2003; Duffey and Pioro 2006, 2005a,b, 2004; Khartabil et al. 2005; Torgerson et al. 2005; Spinks et al. 2002; Bushby et al. 2000a,b; Khartabil 1998;
- in Europe are described in Hofmeister et al. (2005), Mori et al. (2005), Waata et al. (2005), Marsault et al. (2004); Starflinger et al. (2004), Aksan et al. (2003), Bittermann et al. (2003a,b), Cheng et al. (2003), Rimpault et al. (2003) and Squarer et al (2003a,b);
- in Japan can be found in the book by Oka et al. (2010) and papers by Kamei et al. (2005), Kitou and Ishii (2005); Ookawa et al. (2005), Yamada and Oka (2005), Yang et al., (2005), Yamaji et al. (2005a-d, 2004, 2003a-d), Yi et al. (2005a,b; 2004a-c, 2003), Yoo et al. (2005), Ishiwatari et al. (2004, 2003a-e, 2002, 2001), Oka and Yamada (2004), Tanabe et al. (2004), Kataoka et al. (2003), Koshizuka et al. (2003), Oka et al. (2003a-d, 2002, 2000, 1996, 1995a,b, 1994a,b, 1993, 1992a,b), Shioiri et al. (2003), Cheng et al. (2002), Oka (2003, 2002, 2000), Kitoh et al. (2001, 1999), Nakatsuka et al. (2001, 2000, 1998), Koshizuka and Oka (2000, 1998), Mukohara et al. (2000a,b, 1999), Oka and Koshizuka (2000, 1998, 1993), Lee et al. (1999, 1998), Kitoh et al. (1998), Dobashi et al. (1998a,b, 1997), Jevremovich et al. (1996, 1994, 1993a,b), Okano et al. (1996a,b, 1994), Oka and Kataoka (1992) and Kataoka and Oka (1991);
- in Korea – in Joo et al. (2005) and Bae et al. (2004);
- in Russia – in Gabaraev et al. 2005, 2004, 2003a,b; Kuznetsov and Gabaraev 2004; and
- in the USA – Buongiorno et al. (2006, 2003), MacDonald et al. (2005), Modro (2005), Yang and Zavaljevski (2005), Zhao et al. (2005), Fischer et al. (2004), Buongiorno (2004, 2003), Buongiorno and MacDonald (2003a,b,c), Davis et al. (2003).

Typical values of the HTC and wall temperatures at SCWR operating conditions are presented in the nesxt section.

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[4] On progress of the Generation IV nuclear energy systems, see Sagayama (2005).

Table 3.2. Modern concepts of nuclear reactors cooled with supercritical water.

Parameters	Unit	SCW CANDU	HPLWR	SCLWR-H	SCFBR-H	SCWR	B-500 SKDI	ChUWR	ChUWFR	KP-SKD
Reference	–	Bushby et al. 2000; Khartabil et al. 2005	Squarer et al. 2003	Yamaji et al. 2004	Oka, Koshizuka 2000	Bae et al. 2004; Bae 2004	Silin et al. 1993	Kuznetsov 2004 (project from 80s)	Gabaraev et al. 2003a,b; 2004	Gabaraev et al. 2005
Country	–	Canada	EU/Japan	Japan		Korea	Russia	Russia	Russia	Russia
Organization	–	AECL	EU / U of Tokyo	University of Tokyo		KAERI / Seoul NU	Kurchatov Institute	RDIPE (НИКИЭТ)		
Reactor type	–	PT	RPV	RPV	RPV	RPV	RPV	PT	PT	PT
Spectrum	–	Thermal	Thermal	Thermal	Fast	Thermal	Thermal	Thermal	Fast	Thermal
Power thermal	**MW**	2540	2188	2740	3893	3846	1350	2730	2800	1960
electrical	**MW**	1140	1000	1217	1728	1700	515	1200	1200	850
linear max/ave	**kW/m**		39/24	39/18	39	39/19		38/27		69/34.5
Thermal eff.	**%**	45	44	44.4	44.4	44	38.1	44	43 (48)	42
Pressure	**MPa**	25	25	25	25	25	23.5	24.5	25	25
T_{in} coolant	**ºC**	350	280	280	280	280	355	270	400	270
T_{out} coolant	**ºC**	625	500	530	526	508	380	545	550	545
Flow rate	**kg/s**	1320	1160	1342	1694	1862	2675	1020		922
Core height	**m**		4.2	4.2	3.2	3.6	4.2	6	3.5	5
Diameter	**m**	~4		3.68	3.28	3.8	2.61	11.8	11.4	6.45
Fuel	–	UO_2/Th	UO_2 or MOX	UO_2	MOX	UO_2	UO_2	UCG	MOX	UO_2
Enrichment	**% wt.**	4	<6%	~6.1		5.8	3.5	4.4		6
Cladding material	–	Ni alloy	SS	Ni alloy	Ni alloy	SS	Zr alloy / SS	SS	SS	SS
# of FA		300	121	121	419	157	121	1693	1585	653
# of FR in FA		43	216/252	300		284	252	10	18	18
D_{rod}/δ_w	**mm/mm**	11.5 and	8	10.2/0.63	12.8	9.5/0.635	9.1 (Zr), 8.5	12/1	12.8	10/1
Pitch	**mm**	13.5	9.5			11.5	(SS)			
T_{max} cladding	**ºC**	<850	620	650	620	620	425	630	650	700
Moderator	–	D_2O	H_2O	H_2O	H_2O	ZrH_2	H_2O	Graphite	–	D_2O

Explanations to the table:

Concepts appear according to the alphabetical order of the country of origin.

ChUWR –Channel-type Uranium-graphite Water Reactor with annular-type elements cooled from inside; ChUWFR – Channel-type Uranium-graphite Water

Fast Reactor; FA – fuel assembly; FR – fuel rod; HPLWR – High Performance Light Water Reactor; KP-SKD – Channel Reactor of Supercritical Pressure (in Russian abbreviations); PT – Pressure Tube (reactor); PVWR – Pressure-Vessel Water reactor; RPV – Reactor Pressure Vessel; SCFBR-H – SuperCritical Fast Breeder Reactor with High temperature; SCLWR-H – SuperCritical Light Water Reactor with High temperature; Seoul NU – Seoul National University; SKDI – SuperCritical Pressure Integral (reactor) (in Russian abbreviations); TBD – To Be Determined; UCG – Uranium Carbide Grit; U of Tokyo – University of Tokyo; VNIIAM – All-Union Scientific-Research Institute of Atomic Machine Building (in Russian abbreviations); WWPR-SCP – Water-Water Power Reactor ("VVER" in Russian abbreviations) of SuperCritical Pressure.

Continuation of Table 3.2.

Parameters	Unit	PVWR	WWPR-SCP	SCWR-US
Reference	–	Filippov et al. 2003	Baranaev et al. 2004	Buongiorno, MacDonald 2003
Country	–	Russia	Russia	USA
Organization	–	VNIIAM / Kurchatov Institute	IPPE (ФЭИ)	INEEL
Reactor type	–	RPV	RPV	RPV
spectrum	–	Thermal	Fast	Thermal
Power thermal	**MW**	3500	3830	3575
electrical	**MW**	1500	1700	1600
linear max/ave	**kW/m**		35/15.8	39/19.2
Thermal eff.	**%**	43	44	44.8
Pressure	**MPa**	25	25	25
T_{in} **coolant**	**ºC**	280	280	280
T_{out} **coolant**	**ºC**	550–610	530	500
Flow rate	**kg/s**	1600	1860	1843
Core height **diameter**	**m** **m**	3.5 2.92	4.05 3.38	4.87 3.91
Fuel	–	UO_2	MOX	UO_2 95%
Enrichment	**% wt.**			5
Cladding material	–		Ni alloy	TBD
# of FA		37	241	145
# of FR in FA			252	300
D_{rod}/δ_w **Pitch**	**mm/mm** **mm**	Sphere 1.8 mm	10.7/0.55 12	10.2/0.63 11.2
T_{max} **cladding**	**ºC**	630–730	630	
Moderator	–	H_2O	$ZrH_{1.7}$	H_2O

3.3 SUPERCRITICAL WATER-COOLED CANDU NUCLEAR-REACTOR CONCEPT

The SCW CANDU nuclear reactor is a pressurized-channel type reactor (Torgerson et al. 2005; Spinks et al. 2002). The general concept of an SCW CANDU reactor is shown in Figures 3.4 and 3.6. Supercritical water (dense fluid) at a temperature of about 350ºC (inlet temperature is below the pseudocritical temperature of 384.9ºC) from a circulation pump enters the reactor core and heats up caused by the heat of fission to 625ºC (outlet temperature is above the pseudocritical temperature of 384.9ºC) at a pressure of about 25 MPa, which is above the critical pressure of 22.1 MPa. After that, supercritical water is directed to a turbine to perform useful work and returns back through the cooler to the circulation pump. Due to high operating parameters, the coolant in the second circuit may be used for a heat supply or be directed to intermediate or low-pressure turbines.

High pressures and temperatures inside the reactor core require a new design of the fuel channel (Duffey et al. 2003).

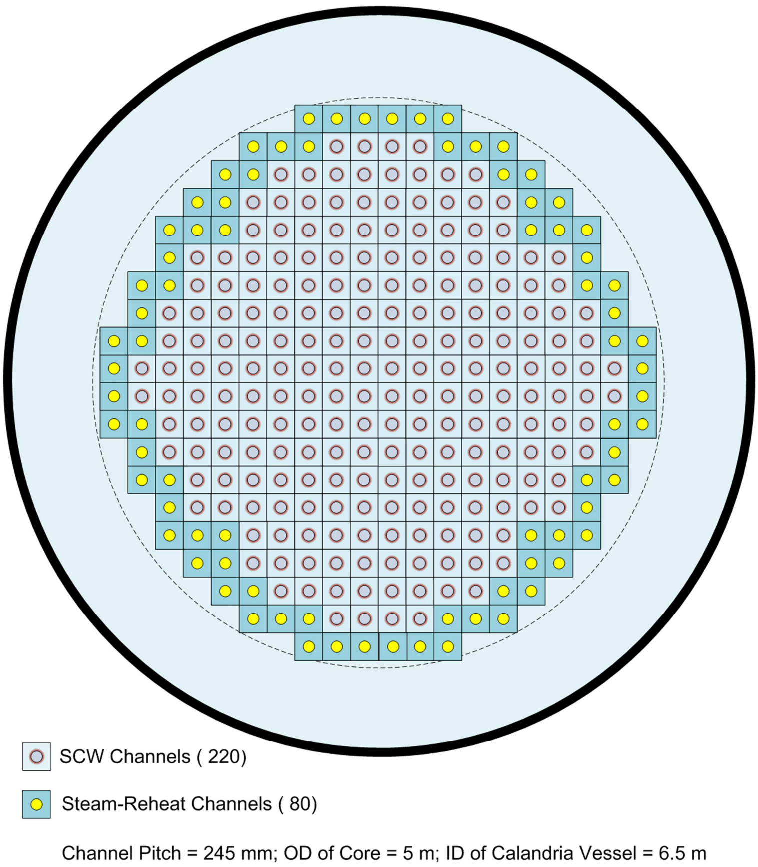

Figure 3.6. Generic channel layout of a 1200-MW$_{el}$ PT SCWR.

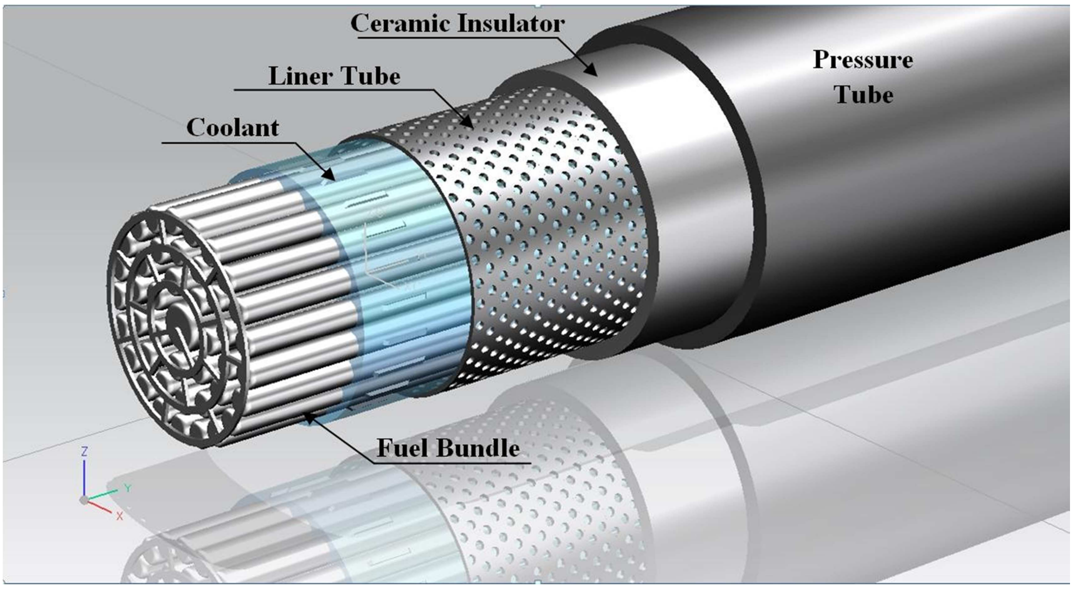

Figure 3.7. 3-D view of high efficiency fuel channel (based on AECL design).

The insulated pressure tube (Figure 3.7) is a key technology that is needed to make use of supercritical water in channel-type reactors feasible (however, other options such as a solid moderator using graphite or zirconium hydride are possible). In this design, the pressure tube is insulated from the coolant by using an internal layer of low-neutron absorbing material. Furthermore, an internally insulated pressure tube operates at much lower temperatures (close to the moderator temperature) than in current reactors, which means that any increase in pressure-tube thickness, if any, is negligible.

Like the commercialization of High-Temperature Reactors (HTRs), Very High-Temperature Reactors (VHTRs) and SCWRs themselves, direct application of heat from HTRs to produce hydrogen is not an immediate prospect. In the near term, electrolysis can gradually supplement first-generation production by Steam-Methane-Reforming (SMR) process ($CH_4 + 2\cdot H_2O = 4\cdot H_2 + CO_2$) with the electricity produced in low-cost Generation III+ reactors such as the ACR™ (Advanced CANDU Reactor) at other than periods of peak electrical demand. Economic assessments show this is competitive with the SMR process for large-scale, industrial production of hydrogen as well as for dispersed, smaller-scale production.

Since so-called high-temperature electrolysis using solid oxide fuel cells (SOFCs) seems to have advantages in efficiency at higher temperatures, at say around 750°C and above, a possibility was examined of adding steam superheat channels to the SCWR concept to give even higher outlet temperatures.

This is relatively simple in principle, and has been demonstrated in practice at the Russian Beloyarsk NPP, named by I.V. Kurchatov (Grigor'yants et al. 1979; Baturov et al. 1978; Samoilov et al. 1976; Aleshchenkov et al. 1971; Dollezhal' et al. 1974, 1971, 1958). In these 170 – 500 MW_e power reactors of the Beloyarsk NPP, the superheat channels were supplied as a second pass with the exit steam from the first pass through the reactor core, to give an average outlet temperature of about 500°C at a pressure of about 7 to 8 MPa. Operation of these reactors with these superheat channels was reported as entirely satisfactory, once some initial manufacturing issues had been resolved.

The chosen reactor exit temperature can be increased by either extending the channel length, or having one or more additional passes through the core. Superheat channels are then located at the periphery of the reactor core and have about 1.5 times lower heat flux compared to the average heat flux. For the SCWR superheat version of the SCW CANDU reactor, the following option was examined: the use of steam discharged from the HP turbine outline, reheated and then introduced at 6 MPa and 395°C into the superheat channels (Figure 3.8).

Since these superheat channels are now at a much lower (highly subcritical) pressure near standard steam conditions, re-entrant flow or stainless steel can be used for the pressure tubes for these superheat channels. In addition, a degree of the superheat can be chosen and varied together with the mass-flow fraction that is superheated, as the steam demand may vary.

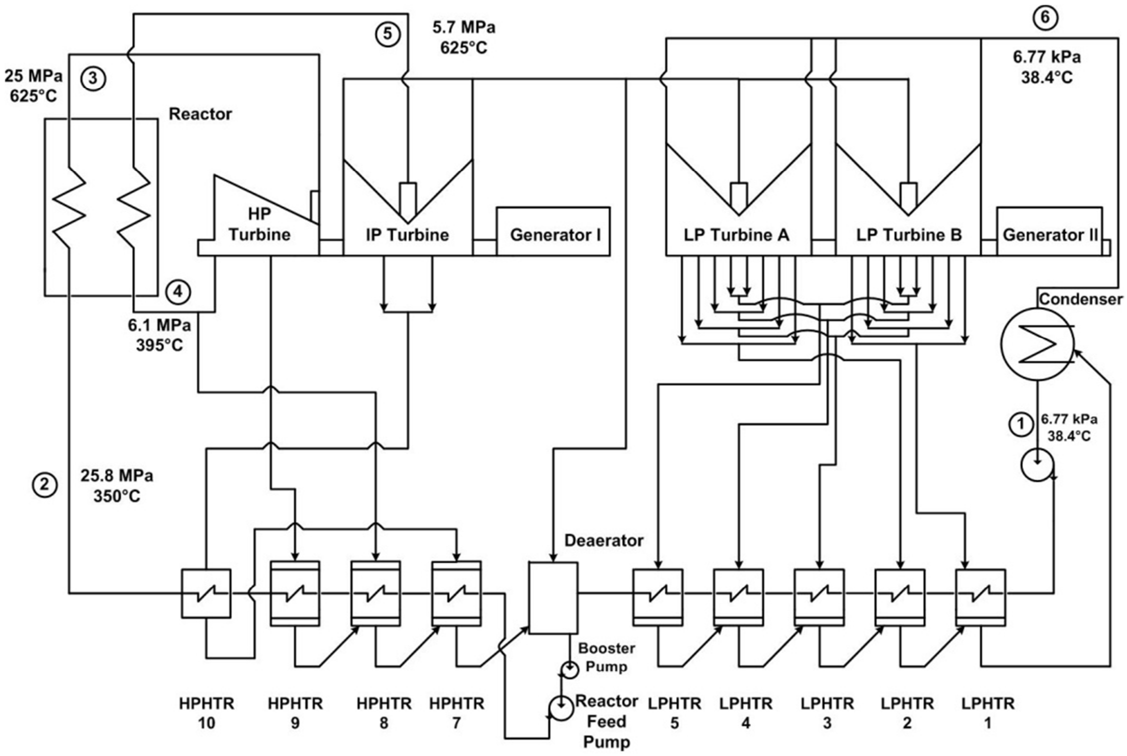

Figure 3.8: Single-reheat cycle for SCW NPP.

3.3. SOME DESIGN FEATURES OF RDIPE PRESSURE-CHANNEL SCWR

The current concept of a direct-flow pressure-channel reactor operating at supercritical pressures (KP-SKD) with vertical channels (Gabaraev et al. 2005) (Figure 3.5 and Table 3.1) possesses all advantages of current pressure-channel reactors. This reactor is a thermal-spectrum type, because of the special features of the channel design eliminating the closely-spaced lattice of fuel elements in the core. Therefore, it cannot be tailored to a fast neutron spectrum.

It has been shown (Gabaraev et al. 2003) that a fast pressure-channel reactor with a gaseous medium instead of the moderator does not possess an efficient fuel cycle because of the extremely high neutron leakage and low fuel-breeding ratio. Consequently, the proposed concept aims at thermal reactors also with a heavy-water moderator, which gives an acceptable neutron balance and deep fuel burn-up. The required temperature 80 – 90°C is easily achieved for such moderator because of the possibility of circulation and external cooling. A graphite moderator is not considered due to the difficulty of achieving the required cooling of the structure at supercritical operating conditions.

Neutron-physical characteristics and economic indicators of KP-SKD have been investigated for two types of fuel elements and channels:

- A channel containing a pressure-carrying zirconium channel tube (placed in a calandria zirconium tube), a steel screen casing, and RBMK-type fuel assemblies; fuel – uranium dioxide pellets or cermet – micro-particles of uranium dioxide in a metallic matrix; fuel element coating – heat-resistant steel (Figure 3.9); And
- A channel containing an assembly of tubular (annular) AMB-type fuel elements, likewise placed in a calandria tube; fuel – cermet; the outer coating and a central-pressure carrying fuel-element tube are made of heat-resistant steel (Figure 3.10).

The first type of channel design was developed with nuclear superheating of steam (Dollezhal' and Emel'yanov 1980). Here, the channel tube is protected from heating by a casing (in contrast to the Canadian concept, which uses a layer of thermal insulation). This design has been successfully tested as a steam-superheating channel in the No. 2 unit at the Beloyarskaya NPP.

In the first design (Figure 3.9), coolant with density of 0.78 g/cm^3 and temperature of 275°C is fed into the channel from below into a gap between the steel casing and the channel tube and rises upward, cooling the tube. In the top of the channel, the coolant flow with a temperature of 360°C and a density of 0.61 g/cm^3 turns into the space between the fuel elements, and is heated, on average, up to 550°C at the core exit. The maximum wall temperature is about 620°C.

In the second design (Figure 3.10), the coolant flows into the channel from above into the downflow central tube of each fuel element and the downflow (interior) tube of the central tube of the channel. At the bottom of the channel coolant with temperature 380°C and density 0.43 g/cm^3 turns and enters the ring-shaped gap of the rising branch, cooling directly the interior wall of the fuel element and having the same water parameters at the exit – 550°C and density of 0.08 g/cm^3. And the maximum wall temperature is about 600°C.

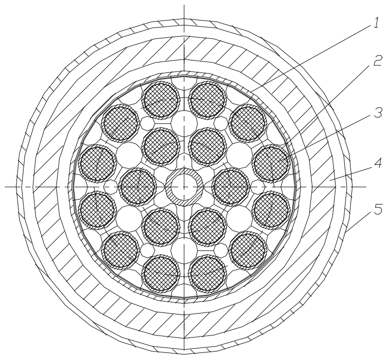

Figure 3.9. Cross section of fuel channel with 18-rod fuel elements (Duffey et al. 2006; Gabaraev et al. 2005): 1) thin-wall 60 × 1 mm separating casing; 2) 10 × 1 mm fuel element cladding; 3) fuel element; 4) 78 × 6 mm channel tube; and 5) 90 × 3 mm calandria tube.

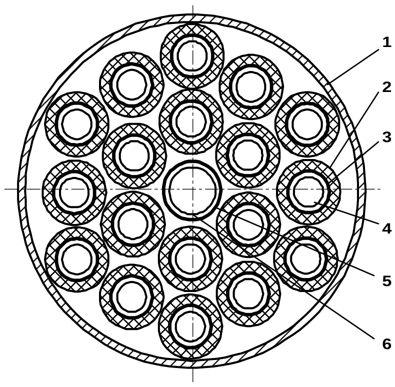

Figure 3.10. Cross section of fuel channel with 18-annular fuel elements (Duffey et al. 2006; Gabaraev et al. 2005): l) 107 × 3 mm calandria tube; 2) 19 × 0.3 mm outer cladding of a fuel element; 3) 12 × 1.2 mm pressure-carrying fuel-element tube; 4) 70 × 0.2 mm separating tube; 5) 20 × 2 mm pressure-carrying central tube of a fuel assembly; and 6) 13 × 0.3 mm separating tube.

To implement this technology, the fuel elements differ from those with the conventional ceramic fuel as follows.

The prototype of the second type of channel is a nominal steam-superheating channel used in the reactors in the first phase of the Beloyarskaya NPP (Samoilov et al. 1982). This design was found to be reliable for prolonged operation with temperatures close to those of KP-SKD. It was possible to increase the steam temperature at the exit from a group of such channels up to 560 – 565°C and to operate a channel successfully up to average burn-up of 43 – 44 MW·days/kg.

3.4. Heat-Transfer Optimization

It is known that the HTC from a fuel element to a gaseous coolant (supercritical water is considered physically as a dense gas) is lower than in subcritical water-cooled nuclear reactors

(Hewitt and Collier 2000) (for details, see Table 3.3) for the same velocity. Hence the fuel centerline temperature will be higher in a SCWR than in a subcritical water-cooled nuclear reactor.

Table 3.3. Typical values of HTC (normal heat-transfer regime) for reactor coolants within operating ranges.

Reference	Reactor coolant	Heat-transfer cooling conditions	Typical geometry	HTC range (kW/m^2K)
Hewitt and Collier 2000	Subcritical water	Forced convection	Fuel bundles	30
		Flow boiling		60
	Subcritical CO_2	Forced convection		1
Yamagata et al. 1972	SCW*	Forced convection: G = 1,120 kg/m^2s	Inside circular tube (10 mm ID)	10–15
Dyadyakin and Popov 1977 (correlation)	SCW	Forced convection: G = 860 kg/m^2s	7-element helically finned bundle model (correlation used for D_{hy}≈8 mm)	4
Pioro and Khartabil 2005	Supercritical CO_2	Forced convection: G = 900 kg/m^2s G = 2000 kg/m^2s	Inside circular tube (8 mm ID)	2–3 3–4

Using simple logic, the wall temperature would be too high after allowance for hot spots and flow distribution uncertainties. However, if high temperature cladding and enhanced heat-transfer surfaces are used in bundles, then the wall temperature is well within safety limits. The objective is simply to increase the supercritical water turbulence level and inter-channel mixing in the bundles. Evidence for the efficiency of bundles is available. It is important to implement these special design features directed to decreasing the fuel centreline temperature and hence the centreline fuel temperature within values allowed by safety limits. These features can be:

- Manufacturing fins on the external surface of fuel-element cladding (Hewitt and Collier 2000, Dyadyakin and Popov 1977). In Magnox gas-cooled nuclear reactors, fuel elements equipped with the herringbone pattern of fins with splitters (Hewitt and Collier 2000) were used. This design feature works as a heat-transfer enhancing device by mixing the gas and as a developed heat-transfer surface (finned surface); hence, the total heat transfer rate was increased by up to 5 – 6 times compared to that of a plain surface.
- Manufacturing ribs on the external surface of fuel element cladding (Figure 3.11) (Hewitt and Collier 2000). In AGRs, fuel elements are machined to produce rectangular ribs of a relatively small height. This design feature works mainly as a heat-transfer enhancing device by mixing of the gas. The HTC was increased by up to 2.5 times compared to that of a plain surface. And
- Using hollow fuel pellets installed inside annular-type elements with internal or internal and external cooling (Figure 3.12) (Dement'ev 1990; Dollezhal' et al. 1971; Aleshchenkov et al. 1971; Kornbichler 1964; Spalaris et al. 1961).

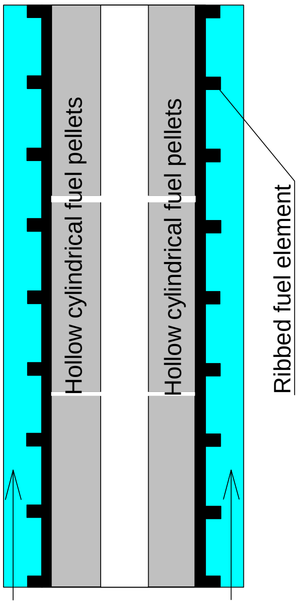

Figure 3.11. AGR ribbed fuel element (Hewitt and Collier 2000).

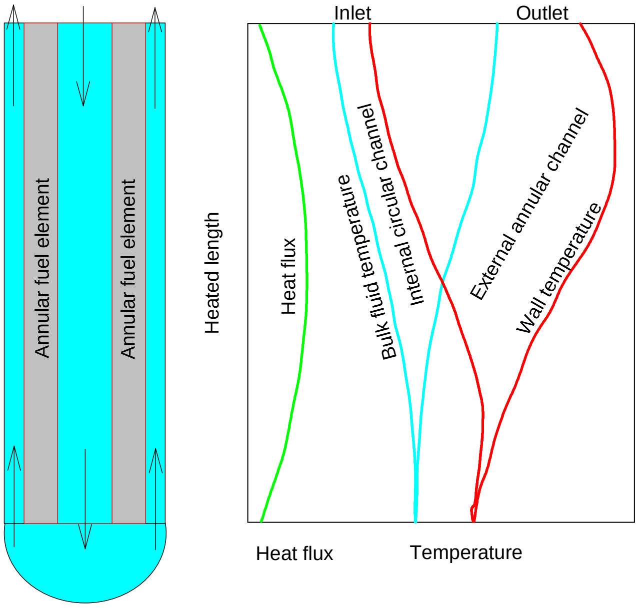

Figure 3.12. Closed-end annular-type fuel element with internal and external cooling (Dement'ev 1990): Scheme and heat flux/temperature profiles along heated length.

Therefore, in spite of some technical difficulties, there are definitely proven ways to overcome them. The supercritical water and other supercritical fluid literature show that enhancements of 2 to 5 times are possible using ribs, grids and "turbulizers" (for details on the heat-transfer enhancement at supercritical pressures, see Chapter 9). Therefore the supercritical water bundle HTC is expected to be between ~ 4 – 20 kW/m^{2}K (see Table 3.4).

Table 3.4. Comparison of values of thermophysical properties of water* and values of HTC for conditions of SCW CANDU reactor, CANDU-6 reactor and PWR.

Parameter	Unit	SCW CANDU		Typical CANDU-6		PWR	
Pressure	MPa	25**		10.5		15	
Location	–	Inlet	Outlet	Inlet	Outlet	Inlet	Outlet
Temperature	ºC	350	625	265	310	290	325
Increase in temperature from inlet to outlet	ºC	275		45		35	
Density	kg/m^3	625.5	67.58	782.9	692.4	745.4	664.9
Enthalpy	kJ/kg	1624	3567	1159	1401	1285	1486
Increase in enthalpy from inlet to outlet	kJ/kg	1943		242		201	
	kJ/kg·K	7.06		5.38		5.74	
Specific heat	J/kg·K	6978	2880	4956	6038	5257	6460
Expansivity	1/K	$5.17 \cdot 10^{-3}$	$1.74 \cdot 10^{-3}$	$2.09 \cdot 10^{-3}$	$3.71 \cdot 10^{-3}$	$2.54 \cdot 10^{-3}$	$4.36 \cdot 10^{-3}$
Thermal conductivity	W/m·K	0.481	0.107	0.611	0.530	0.580	0.508
Dynamic viscosity	Pa·s	$7.28 \cdot 10^{-5}$	$3.55 \cdot 10^{-5}$	$10.12 \cdot 10^{-5}$	$8.24 \cdot 10^{-5}$	$9.23 \cdot 10^{-5}$	$7.81 \cdot 10^{-5}$
Kinematic viscosity	m^2/s	$11.63 \cdot 10^{-8}$	$52.47 \cdot 10^{-8}$	$12.93 \cdot 10^{-8}$	$11.90 \cdot 10^{-8}$	$12.38 \cdot 10^{-8}$	$11.75 \cdot 10^{-8}$
Diffusivity	m^2/s	$11.02 \cdot 10^{-8}$	$54.72 \cdot 10^{-8}$	$15.75 \cdot 10^{-8}$	$12.68 \cdot 10^{-8}$	$14.80 \cdot 10^{-8}$	$11.83 \cdot 10^{-8}$
Surface tension	N/m	–	–	$22.5 \cdot 10^{-3}$	0.0121	$16.7 \cdot 10^{-3}$	$8.77 \cdot 10^{-3}$
Prandtl number	–	1.06	0.96	0.82	0.94	0.84	0.99
Reynolds number ($\times 10^6$) at G***=860 kg/m^{2}s and D_{hy}=8 mm	–	0.946	1.940	0.680	0.835	0.745	0.881
Nusselt number**** ($=0.023 \cdot \mathbf{Re}^{0.8} \cdot \mathbf{Pr}^{0.4}$) (3.1)	–	1418	2425	985	1225	1068	1308
HTC	W/m^{2}K	8527	3228	7522	8114	7744	8303

* All thermophysical properties of water were calculated according to NIST (2002).

** Pseudocritical temperature at pressure of 25 MPa is 384.9ºC.

*** This value of mass flux corresponds to SCW CANDU reactor operating conditions. Mass flux values in subcritical pressure nuclear reactors are much higher; therefore, values of Reynolds number, Nusselt number and HTC will be also much higher in subcritical pressure reactors.

**** Nusselt number is calculated according to Equation (3.1) (Dittus and Boelter 1930) for forced convective heat transfer in a circular tube as a first estimate only.

ACKNOWLEDGEMENTS

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SYMBOLS AND ABBREVIATIONS

A	area, m^2
c_p	specific heat at constant pressure, J/kg K
D	inside diameter, m
G	mass flux, kg/m^{2}s; $\left(\dfrac{m}{A_{fl}}\right)$
g	gravitational acceleration, m/s^2
H	specific enthalpy, J/kg
h	heat transfer coefficient, W/m^{2}K
k	thermal conductivity, W/m K
L	heated length, m
m	mass-flow rate, kg/s; $(\rho\, V)$
P, p	pressure, MPa
Q	power or heat-transfer rate, W
q	heat flux, W/m^2; $\left(\dfrac{Q}{A_h}\right)$
T	temperature, K
t	temperature, ºC

Greek Letters

α thermal diffusivity, m^2/s; $\left(\dfrac{k}{c_p\ \rho}\right)$

μ dynamic viscosity, Pa s

ρ density, kg/m^3

υ kinematic viscosity, m^2/s

Non-dimensional Numbers

Nu Nusselt number; $\left(\dfrac{h\ D}{k}\right)$

Pr Prandtl number; $\left(\dfrac{\mu\ c_p}{k}\right)=\left(\dfrac{\upsilon}{\alpha}\right)$

Re Reynolds number; $\left(\dfrac{G\ D}{\mu}\right)$

Subscripts or superscripts

ave average

cr critical

e electrical

h heated

in inlet

int internal

iso isothermal

ℓ liquid or local

m molar

pc pseudocritical

Abbreviations and acronyms widely used in the text and list of references

ACR	Advanced CANDU Reactor
AECL	Atomic Energy of Canada Limited (Canada)
AGR	Advanced Gas-cooled Reactor
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium nuclear reactor
DHT	Deteriorated Heat Transfer
DOE	Department Of Energy (USA)
HP	High Pressure
HT	Heat Transfer
HTC	Heat Transfer Coefficient
HTR	High Temperature Reactor
IAEA	International Atomic Energy Agency (Vienna, Austria)
ICONE	International Conference On Nuclear Engineering
IHT	Improved Heat Transfer
IP	Intermediate-Pressure (turbine)
KP-SKD	Channel Reactor of Supercritical Pressure (in Russian abbreviations)
LP	Low Pressure (turbine)
MIT	Massachusetts Institute of Technology (Cambridge, MA, USA)
MOX	Mixed Oxide (nuclear fuel)
NIST	National Institute of Standards and Technology (USA)
NPP	Nuclear Power Plant
PT	Pressure Tube
PWR	Pressurized Water Reactor

RBMK	Reactor of Large Capacity Channel type (in Russian abbreviations)
R&D	Research and Development
RDIPE	Research and Development Institute of Power Engineering (Moscow, Russia) (NIKIET in Russian abbreviations)
SCW	SuperCritical Water
SCWR	SuperCritical Water-cooled Reactor
SMR	Steam-Methane-Reforming (process)
SOFC	Solid Oxide Fuel Cell
UOIT	University of Ontario Institute of Technology
US or USA	United States of America
USSR	Union of Soviet Socialist Republics
VHTR	Very High-Temperature Reactor

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