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Pioro I. and Peiman W.
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 2. SUPERCRITICAL WATER-COOLED REACTOR CONCEPT

Анотація

У останній другій частині статті розглядаються найновітніші принципові, конструктивні та технологічні схеми атомних реакторів IV покоління надкритичних водоохолоджувальних реакторів, надаються перспективні та прогнозні параметри технічних характеристик таких енергетичних установок.

Abstract

Currently, there are a number of Generation IV SuperCritical Water-cooled nuclear Reactor (SCWR) concepts under development worldwide. The main objectives for developing and utilizing SCWRs are to: 1) Increase gross thermal efficiency of current Nuclear Power Plants (NPPs) from 30–35 % to approximately 45–50 %, and 2) Decrease capital and operational costs and, in doing so, decrease electrical-energy costs.

SuperCritical Water (SCW) NPPs will have much higher operating parameters compared to current NPPs (i. e., steam pressures of about 25 MPa and steam outlet temperatures up to 625°C). Additionally, SCWRs will have a simplified flow circuit in which steam generators, steam dryers, steam separators, etc. will be eliminated. Furthermore, SCWRs operating at higher temperatures can facilitate an economical co-generation of hydrogen through thermochemical cycles (particularly, the copper-chlorine cycle) or direct high-temperature electrolysis.

To decrease significantly the development costs of an SCW NPP, to increase its reliability, and to achieve similar high thermal efficiencies as the advanced fossil steam cycles, it should be determined whether SCW NPPs can be designed with a steam-cycle arrangement that closely matches that of mature SuperCritical (SC) fossil-fired thermal power plants (including their SC-turbine technology). The state-of-the-art SC-steam cycles at fossil-fired power plants are designed with a singlesteam-reheat and regenerative feedwater heating. Due to this, they can reach gross thermal steam-cycle efficiencies of up to 52 %.

This paper presents and discusses basic ideas on SCWRs as one of the most promising Generation IV concept.

Introduction

Accounting that the vast majority of modern nuclear reactors are water-cooled reactors we consider a SuperCritical Water-cooled Reactor (SCWR) concept as the most viable option for further development. Concepts of nuclear reactors cooled with water at supercritical pressures were mainly studied in Russia and the USA as early as the 1950s and 1960s (Pioro and Duffey, 2007). After a 30-year break, the idea of developing nuclear reactors cooled with supercritical water became attractive again as the ultimate development path for water cooling. Many countries (Canada, China, Germany, Japan, Korea, Russia, USA and others) have started to work in this direction. However, none of these concepts is expected to be implemented in practice before 2020–2025.

The main objectives of using supercritical water in nuclear reactors are: 1) Increase thermal efficiency of modern Nuclear Power Plants (NPPs) from 30–35 % to about 45–50 %, 2) Decrease capital and operational costs and hence, decrease electrical-energy costs, and 3) Possibility for co-generation of hydrogen. For instance, the copper-chlorine cycle requires steam at temperatures between 500 and 530 °C (Naterer et al., 2010, 2009), which is within the operating range of SCWRs. These systems work when supercritical water from a reactor flows through a heat exchanger and transfers heat to a low-pressure steam, which becomes a superheated steam. This superheated steam is transferred to an adjacent hydrogen plant at a lower pressure.

SCW NPPs will have much higher operating parameters compared to modern NPPs (a pressure of about 25 MPa and outlet temperature up to 625 °C), and a simplified flow circuit, in which steam generators, steam dryers, steam separators, etc., can be eliminated (Pioro, 2011; 2009).

The design of SCW nuclear reactors (Pioro and Duffey, 2007) is seen as a natural and ultimate evolution of today's conventional modern water-cooled reactors. Development of SCWRs is based on the following three proven technologies: 1) modern Pressurized Water Reactors (PWRs), which operate at pressures of 15–16 MPa (see Fig. 1), i. e., quite high pressures; 2) Boiling Water Reactors (BWRs), which are a once-through or direct-cycle design, i. e., steam from a nuclear reactor is forwarded directly into a turbine; and 3) modern supercritical turbines with pressures about 23.5–38 MPa and inlet temperatures up to 625°C, which operate successfully at coal-fired thermal power plants for more than 50 years. In addition, some experimental reactors used nuclear steam reheat with outlet steam temperatures well beyond the critical temperature (up to 550°C), but at pressures below the critical pressure (3–7 MPa), to increase the gross thermal efficiency of NPP (for details, see Figs. 2 and 3, and Tables 1 and 2) (Saltanov and Pioro, 2011).

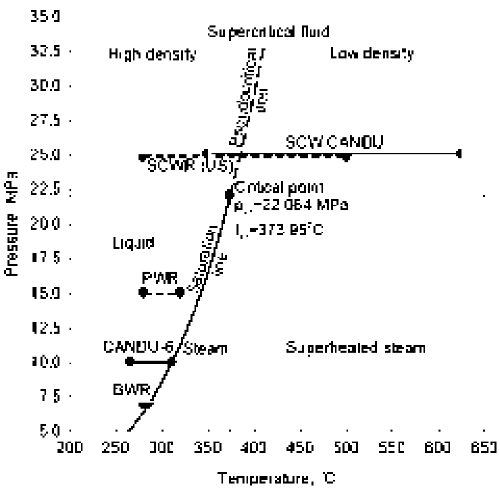


Figure 1. Pressure-Temperature diagram of operating conditions of various water-cooled nuclear reactors

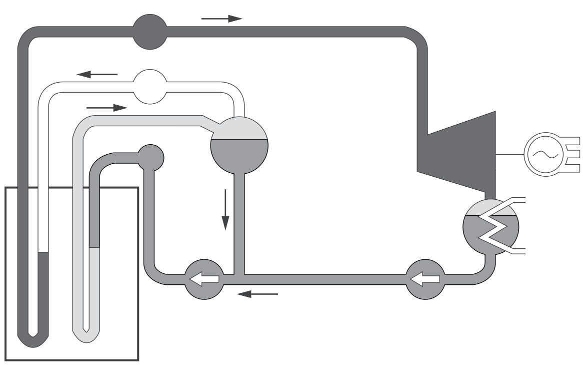


Figure 2. Beloyarsk NPP (Russia) reactor schematic: Unit 2 with direct steam cycle (courtesy of Dr. Yurmanov, NIKIET, Russia):

Reheated steam; Saturated steam; Water-steam mixture; Water

Table 1

Main parameters of Beloyarsk NPP reactors (Saltanov and Pioro, 2011)

Parameters	Unit 1 (730 EChs & 268 SRChs)	Unit 2 (732 EChs & 266 SRChs)
Electrical power, MW_{el}	100	200
Number of K-100-90-type turbines	1	2
Inlet-steam pressure, MPa	8.5	7.3
Inlet-steam temperature, °C	500	501
Gross thermal efficiency, %	36.5	36.6
Uranium load, t	67	50
Uranium enrichment, %	1.8	3.0
Square lattice pitch, mm	200	200
Core dimensions, m: Diameter	7.2	7.2
Height	6	6

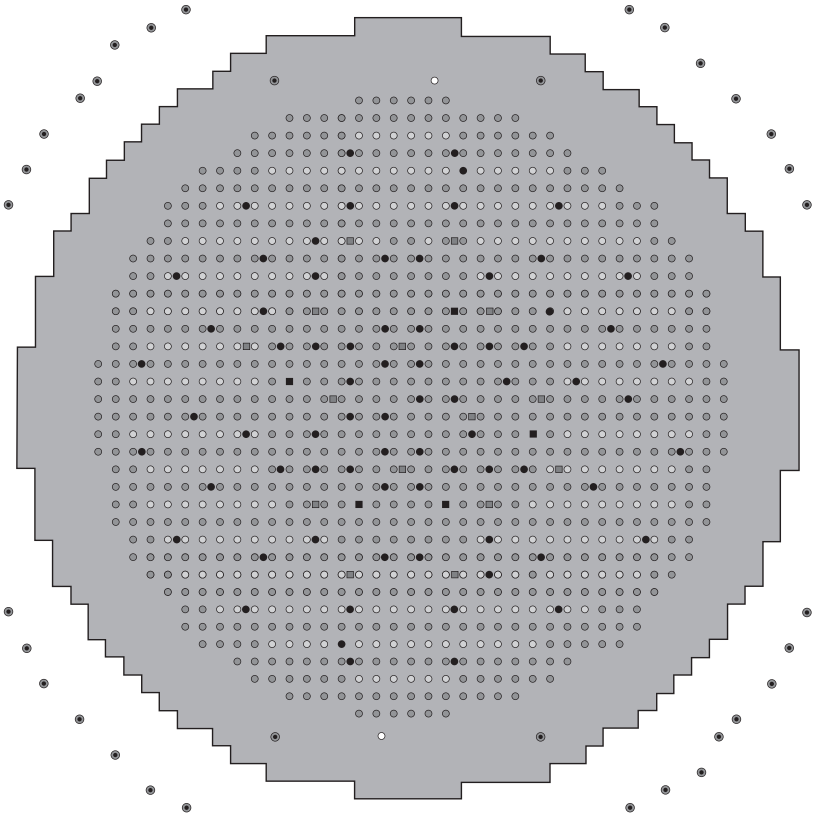
EChs — Evaporative Channels; SRChs — Steam-Reheat Channels

Table 2

Average parameters of Beloyarsk NPP Unit 1 before and after installation of Steam-Reheat Channels (SRChs) (Saltanov and Pioro, 2011)

Parameters	Before SRChs installation	After SRChs installation
Electrical power, MW_{el}	60–70	100–105
Steam inlet pressure, MPa	5.9–6.3	7.8–8.3
Steam inlet temperature, °C	395–405	490–505
Exhaust steam pressure, kPa	9–11	3.4–4.0
Water mass flowrate (1^{st} loop), kg/h	1400	2300–2400
Pressure in steam separators, MPa	9.3–9.8	11.8–12.7
Gross thermal efficiency, %	29–32	35–36
Electrical power for internal needs, %	10–12	7–9

In general, SCWRs can be classified based on a pressure boundary, neutron spectrum or moderator. In terms of the pressure boundary, SCWRs are classified into two categories, a) Pressure Vessel (PV) SCWRs, and b) Pressure Tube (PT) or Pressure Channel (PCh) SCWRs (Oka et al., 2010; Pioro and Duffey, 2007). The PV SCWR requires a pressure vessel with a wall thickness of about 50 cm (Pioro and Duffey, 2007) in order to withstand high pressures. The vast majority of conventional PWRs and BWRs are examples of PV reactors. Figure 4 shows a schematic diagram of a PV SCWR. Table 3 lists general operating parameters of modern PV SCWR concepts. On the other hand, the core of a PT SCWR consists of distributed pressure channels, with a thickness of about 10 mm, which might be oriented vertically or horizontally, analogous to RBMK and CANDU reactors, respectively. For instance, SCW CANDU reactor (Fig. 5) consists of 300 horizontal fuel channels with coolant inlet and outlet temperatures of 350 and 625 °C at a pressure of 25 MPa (Pioro and Duffey, 2007). It should be noted that a vertical core option (Fig. 6) has not been ruled out; both horizontal and vertical cores are being studied by the Atomic Energy of Canada Limited (AECL). Table 4 provides information about modern concepts of PT SCWR.



- Water-Steam Channels-730
- Operating Steam-reheat Channels-268
- Channels for Compensating Rods-78
- Shut-Down Rods-16
- Regulating Rods-6
- Channels for Measuring-2
- Channels for measuring-4 + 30-Channels for lonization Chambers

Figure 3. Beloyarsk NPP Unit 1 channels layout

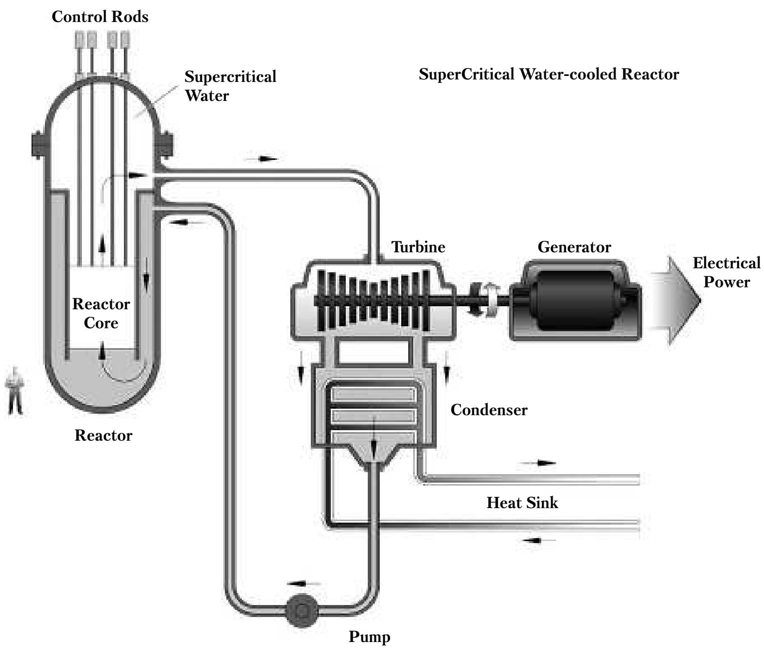


Figure 4. Schematic diagram of PV SCWR (US DOE, 2002)

Table 3

Modern concepts of PV SCWRs (Pioro and Duffey, 2007)

Parameters	Unit	PV SCWR Concepts		
Country	–	**Russia**		**USA**
Spectrum	–	**Thermal**	**Fast**	**Thermal**
Power electrical	MW	1500	1700	1600
Thermal efficiency	%	34	44	45
Pressure	MPa	25	25	25
Coolant temperature	°C	280–550	280–530	280–500
Flow rate	kg/s	1600	1860	1840
Core height/ diameter	m/m	3,5/2,9	4,1/3,4	4,9/3,9
Fuel	–	UO_2	MOX	UO_2
Enrichment	$\%_{wt}$	–	–	5
Maximum cladding temperature	°C	630	630	–
Moderator	–	H_2O	–	H_2O

Table 4

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

Parameters	Unit	PT SCWR concepts			
Country	–	**Canada (Fig. 5)**	**Russia (NIKIET)**		
Spectrum	–	**Thermal**	**Thermal**	**Fast**	**Thermal**
Power electrical	MW	1220	1200	1200	800
Thermal efficiency	%	48	44	43	42
Pressure	MPa	25	24,5	25	25
Coolant temperature	°C	350–625	270–545	400–550	270–545
Flow rate	kg/s	1320	1020	–	922
Core height/ diameter	m/m	/7	6/12	3,5/11	5/6,5
Fuel	–	UO_2/Th	UCG	MOX	UO_2
Enrichment	$\%_{wt}$	4	4,4	–	6
Maximum cladding temperature	°C	850	630	650	700
Moderator	–	D_2O	Graphite	–	D_2O

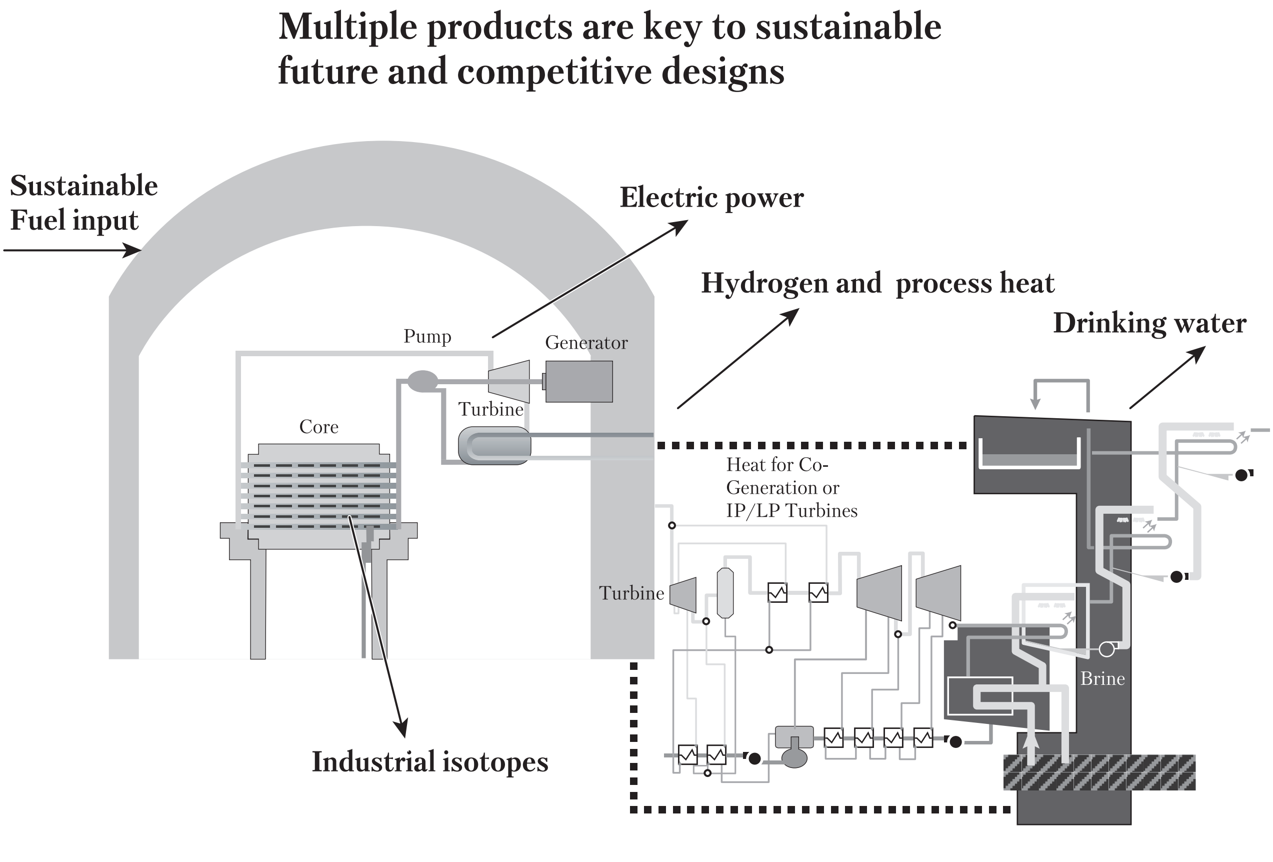


Figure 5. General scheme of pressure-channel SCW CANDU reactor:
IP — intermediate-pressure turbine and LP — low-pressure turbine (courtesy of Dr. Duffey, AECL)

Figure 6. Vertical core-configuration option (courtesy of AECL)

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

One of these options is to use a liquid moderator such as heavy-water. One of the advantages of using a liquid moderator in PT SCWRs is that the moderator acts as a passive heat sink in the event of a Loss Of Coolant Accident (LOCA). A liquid moderator provides an additional safety feature[1], which enhances the safety of operation. On the other hand, one disadvantage of liquid moderators is an increased heat loss from the fuel channels to the liquid moderator, especially at SCWR conditions.

The second option is to use a solid moderator. Currently, in RBMK reactors and some other types of reactors such as Magnox, AGR, and HTR, graphite is used as a moderator. However, graphite may catch fire at high temperatures under some conditions when exposed to water or oxygen. Other materials such as beryllium oxide and zirconium hydride may be used as solid moderators. In this case, heat losses can be reduced significantly. On the contrary, the solid moderators do not act as a passive-safety feature.

High operating temperatures in SCWRs lead to high fuel centerline temperatures. Currently, UO_2 has been used in Light Water Reactors (LWRs) and Pressurized Heavy-Water Reactors (PHWRs). However, the uranium dioxide fuel has a lower thermal conductivity (Fig. 7 and Table 5), which results in high fuel centerline temperatures. Therefore, alternative fuels with high thermal-conductivities such as Uranium Dioxide plus Silicon Carbide (UO_2–SiC), Uranium Dioxide composed of Graphite fibbers (UO_2–C), Uranium Dioxide plus Beryllium Oxide (UO_2–BeO), Uranium Dicarbide UC_2, Uranium Monocarbide (UC) and Uranium Mononitride (UN) (Fig. 7 and Table 5) might be used.

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

Table 5

Properties of Uranium Dioxide, Uranium Mononitride, Uranium Monocarbide, and Uranium Dicarbide at 0.1 MPa and 298 K)

Property	**UO_2**	**UC**	**UC_2**	**UN**
Molecular Mass, amu	270.3	250.04	262.05	252.03
Theoretical density, kg/m^3	10 960	13 630	11 680	14 300
Melting Point, °C	2847 ± 30	2507 2532	2375 2562[3]	2850 ± 30
Heat Capacity J/kg K	235	203	233	190
Heat of Vaporization, kJ/kg	1530	2120	1975 ± 20	1144 3325
Thermal Conductivity, W/m K	8.68	21.24	11.57	14.58
Linear Expansion Coefficient, 1/K	$9.75 \cdot 10^{-6}$	$10.1 \cdot 10^{-6}$	$(18.1 \cdot 10^{-6})$	$7.52 \cdot 10^{-6}$

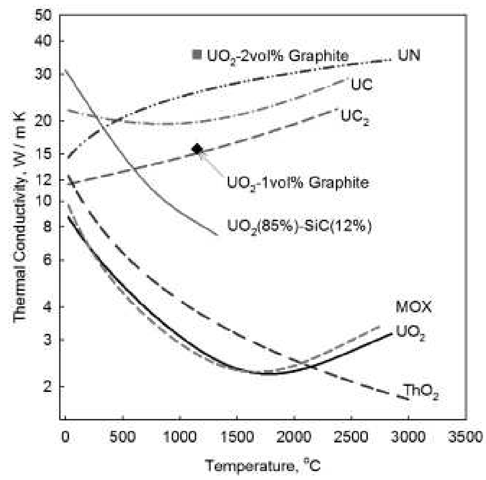
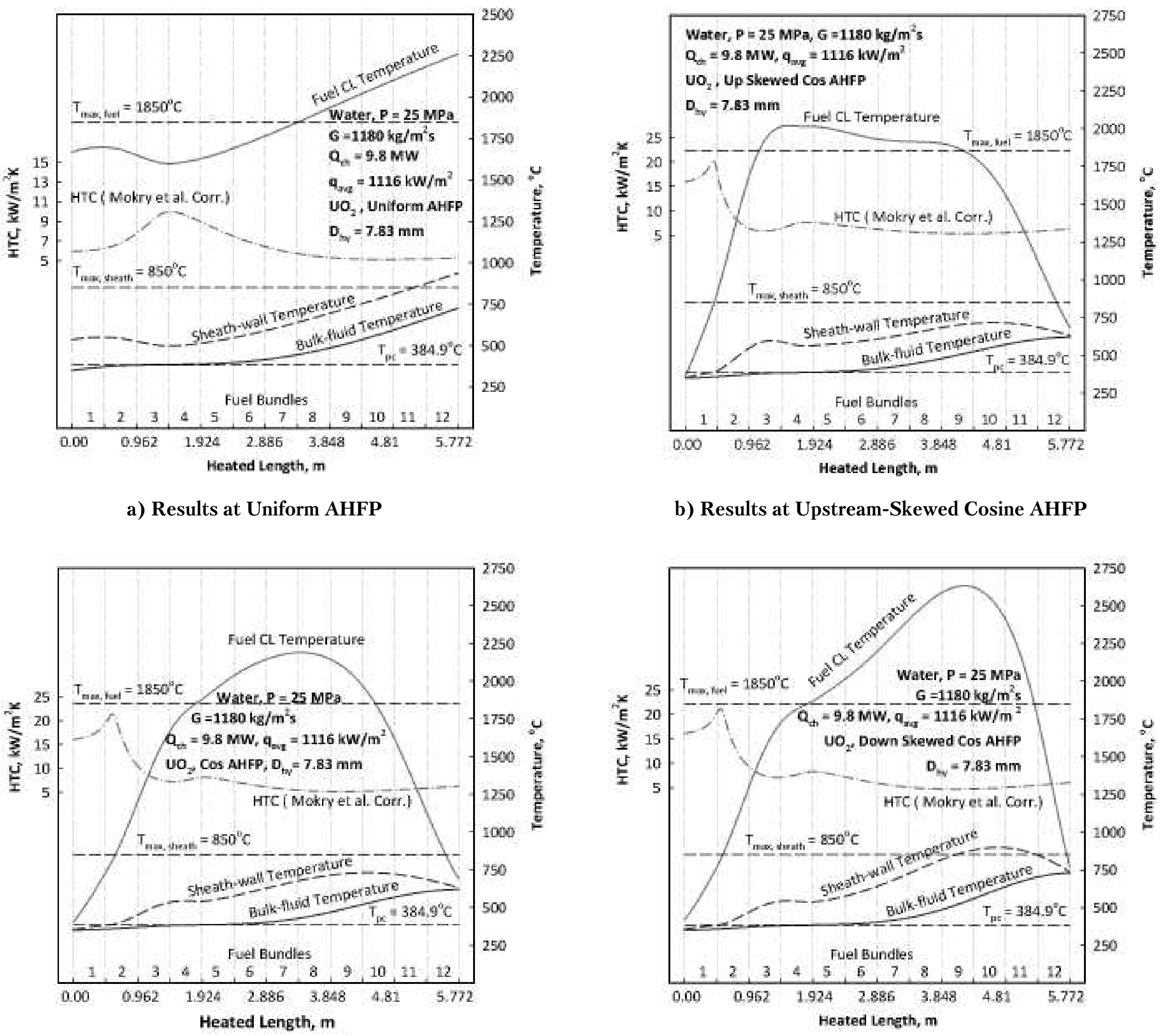


Figure 7. Thermal Conductivity of UO_2, UN, UC, and UC_2, UO_2 plus Graphite-Fiber Fuels as a Function of Temperature

[1] Currently, such option is used in CANDU-6 reactors.



c) Results at Cosine AHFP

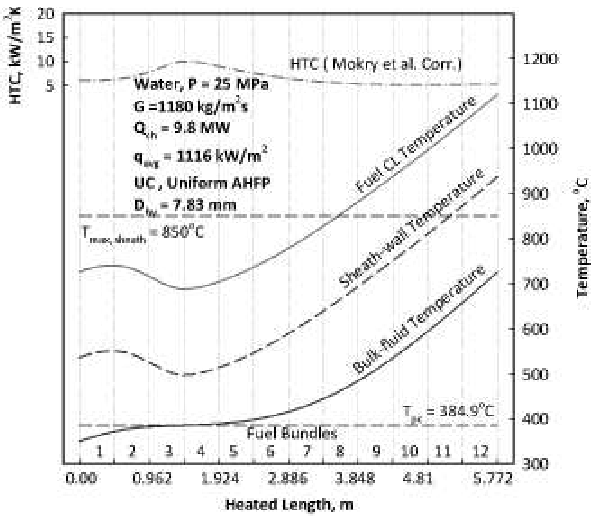
d) Results at Downstream-Skewed Cosine AHFP

Figure 8. Temperature and HTC profiles for UO2 Fuel at Maximum Channel Power with (a) Uniform, (b) Upstream-Skewed Cosine, (c) Cosine, and (d) Downstream-Skewed Cosine AHFPs; Mokry et al. correlation is heat-transfer correlation for vertical bare tubes cooled with supercritical water (Pioro, 2011)

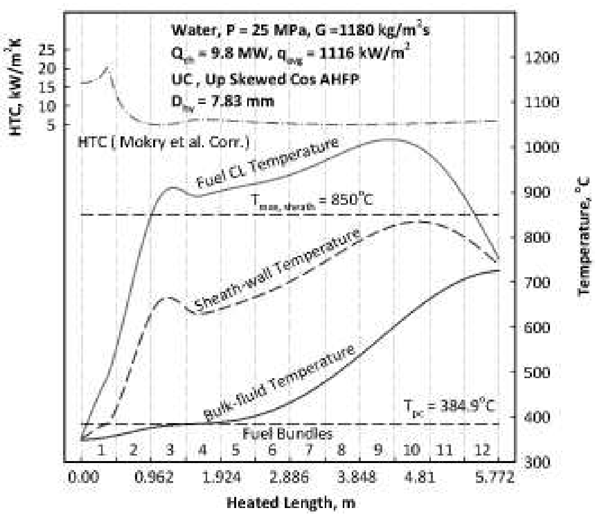
SCWRs Design Considerations

Pressure-vessel SCWRs. The pressure-vessel SCWR design (see Fig. 4 and Table 3) is being developed in China, European Union (EU), Japan and some other countries. This type of reactor, which is based on proven technologies in PWRs and BWRs, uses a traditional high-pressure circuit layout. However, due to significantly reduced flow rates (at supercritical conditions flow rates can be up to 8 times less than those in current reactors at subcritical pressures), high outlet temperatures and some other parameters significant fuel-sheath temperature non-uniformities may appear, which in turn can lead to sheath damage. Another challenge associated with pressure-vessel SCWRs is manufacturing of pressure vessel due to quite large wall thickness. Also, in pressure-vessel reactors nuclear steam reheat at subcritical pressures is not practical, eliminating the possibility for an additional increase in thermal efficiency. More information on thermal and fast pressure-vessel SCWRs can be found in the latest book by Oka et al. (2010).

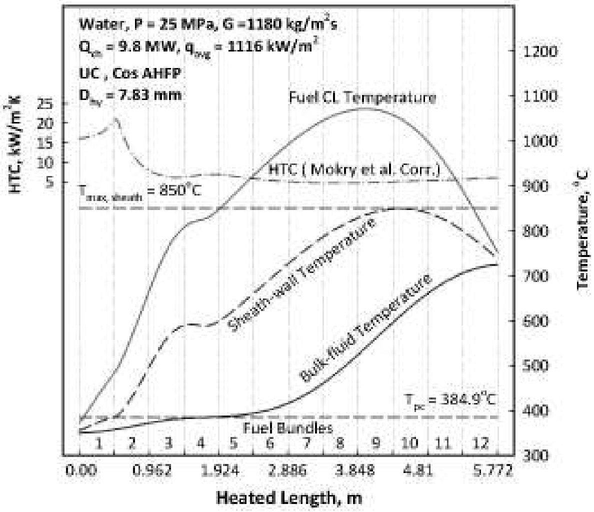
Pressure-channel SCWRs. The pressure-channel SCWR designs (see Figs. 5 and 6 and Table 4) are developed in Canada and in Russia (Pioro and Duffey, 2007). Figure 5 shows the maximum possible outcome from SCWRs. Within those two main classes, pressure-channel reactors are more flexible to flow, flux and density changes than pressure-vessel reactors. In addition, a nuclear steam reheat can be implemented inside a pressure-channel SCWR based on the experience obtained during an oper-



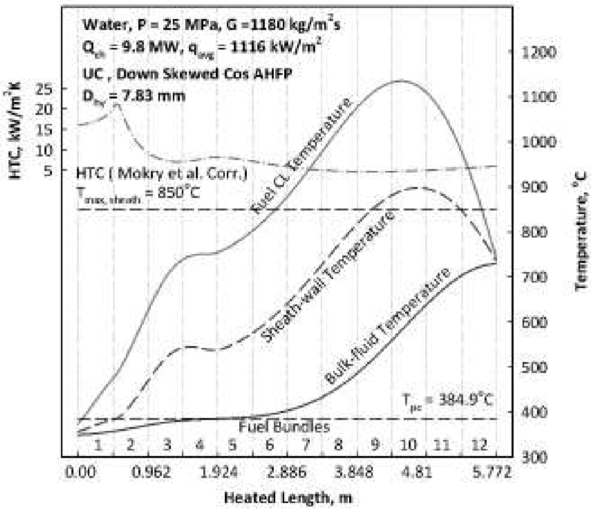
a) Results at Uniform AHFP



b) Results at Upstream-Skewed Cosine AHFP



c) Results at Cosine AHFP



d) Results at Downstream-Skewed Cosine AHFP

Figure 9. Temperature and HTC profiles for UC Fuel at Maximum Channel Power with (a) Uniform, (b) Upstream-Skewed Cosine, (c) Cosine, and (d) Downstream-Skewed Cosine AHFPs

ation of several experimental pressure-channel BWRs in 60-s and 70-s (see Figs. 2 and 3 and Tables 1 and 2), which makes it completely suitable to modern supercritical direct single steam-reheat-cycle turbines. All these make it possible to use the experimentally confirmed, better solutions developed for these reactors. One of them is channel-specific flow-rate adjustments or regulations. Also, a pressure tube at such pressures will have a wall thickness of about 7–9 mm compared to about 400–500 mm for a pressure vessel. Therefore, a design whose basic element is a channel, which carries a high pressure, has an inherent advantage of greater safety than large vessel structures at supercritical pressures.

In general, pressure-channel SCWRs can be with vertical (Fig. 5) or horizontal (Fig. 6) fuel channels. Horizontal orientation has significant benefits if an on-line refuelling is considered. However, at supercritical pressures the implementation of on-line refuelling is an extremely challenging task. As such, it might be abandoned. In this case, the vertical orientation can be a better option. Figure 10 shows a possible fuel-channel layout of a generic 1200-MW_{el} pressure-channel SCWR.

From moderator point of view, pressure-channel SCWRs can be with a liquid moderator (heavy water) or with a solid moderator. After the Chernobyl NPP disaster (Ukraine, year 1986), it seems that graphite as a moderator will not be used in any water-cooled reactors. However, other solid moderators may be used, for example, Beryllium, Beryllium Oxide, ZH_2 etc.

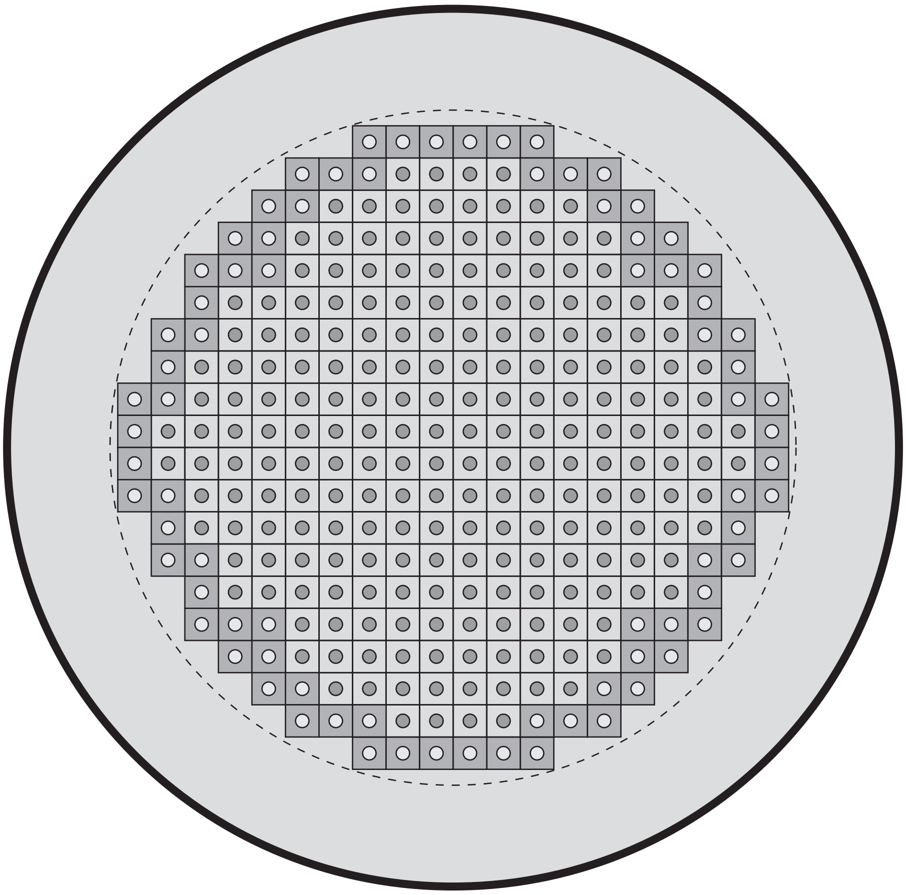


Figure 10. Fuel-channel layout of generic 1200-MW_{el} pressure-channel SCWR

With a solid moderator the fuel-channel design can be simplified, because heat losses from the hot pressure tube are minimal. In this case, for example, Re-Entrant Channels (RECs) without thermal insulation can be used (Fig. 11). Due to lower inlet temperatures (about 300°C) a pressure tube can be manufactured from Zirconium alloys, but a flow tube — from stainless steels or Inconels (currently, these materials are considered only as candidate materials for application at SCWR conditions).

A liquid moderator has a unique feature as an extra safety system during emergency fuel-channel cooling. The moderator in CANDU reactors acts as a backup heat sink in the unlikely event of loss of coolant combined with loss of emergency core cooling. The moderator cooling system removes heat deposited in the moderator during normal operation. The moderator cooling system can also remove decay heat in certain postulated accident scenarios. In the SCWR design, the moderator operates slightly subcooled, which makes it possible to use a flashing-driven passive loop to remove the moderator heat (see Fig. 12).

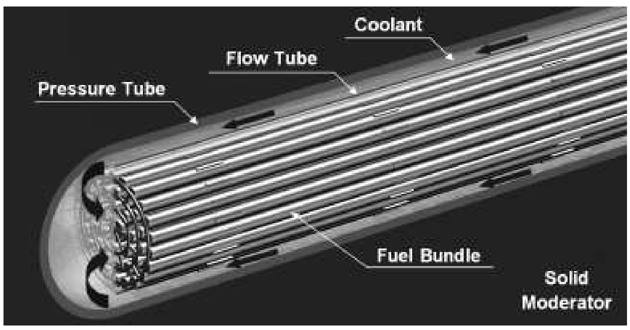


Figure 11. Re-Entrant Channel (REC) for SCWR with solid moderator

Fuel channels in SCWR with a liquid moderator will have more complicated designs (see Figs. 13 15) to prevent high heat losses from the "hot" pressure tube to the low-temperature moderator. A challenging task in these designs is a thermal insulation, which would resist high thermal gradients without developing significant cracks.

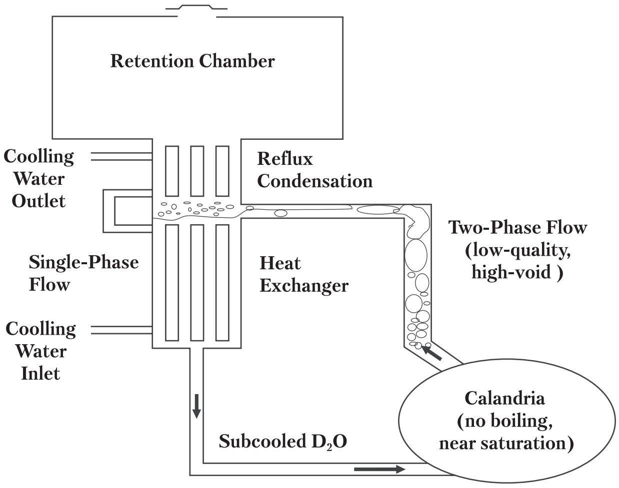


Figure 12. Passive moderator-cooling concept "walk away safety" with no core melting (courtesy of Dr. H. Khartabil, AECL)

Possible Thermodynamic Cycles for SCWRs

In general, the following thermodynamic cycles can be used in SCW NPPs (Pioro, 2011, 2009):

1. Direct single-reheat regenerative thermodynamic cycle (Rankine cycle) (see Fig. 16), which is a basic cycle for the vast majority of modern supercritical coal-fired thermal power plants.
2. In-direct single-reheat regenerative thermodynamic cycle (see Fig. 17).
3. Direct no-reheat regenerative thermodynamic cycle (see Fig. 18).
4. In-direct no-reheat regenerative thermodynamic cycle (see Fig. 19).
5. Dual regenerative thermodynamic cycles (see Figs. 20 and 21).

In the direct cycle, supercritical "steam" from an SCWR is fed directly to a supercritical turbine. This concept eliminates the need for complex and expensive equipment such as steam generators (heat exchangers). From a thermodynamic perspective, this allows for high steam pressures and temperatures, and results in the highest cycle thermal efficiency for the given parameters. The direct single-reheat cycle with current supercritical "steam" parameters will have the gross thermal efficiency of about 52% and no-reheat cycle – about 51%. However, the direct single-reheat cycle is easier to implement in pressure-channel SCWRs and might be impossible to implement in pressure-vessel SCWRs. The direct no-reheat cycle can be implemented in both types of SCWRs.

The single-reheat cycle is widely used in thermal power industry, but we have not found any information on thermal power plants operating on the no-reheat cycle. The major technical challenge for the no-reheat cycle is relatively high moisture content at the outlet of the LP turbine (about 19%). However, the moisture can be reduced by implementing contoured channels in the inner casing for draining the water and moisture removal stages.

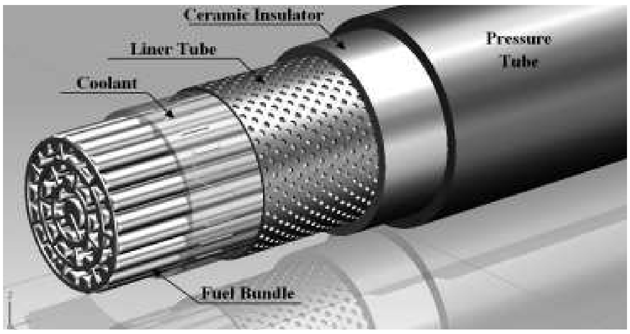


Figure 13. High-Efficiency Channel (HEC) with ceramic insert (AECL design) (drawing prepared by W. Peiman, UOIT)

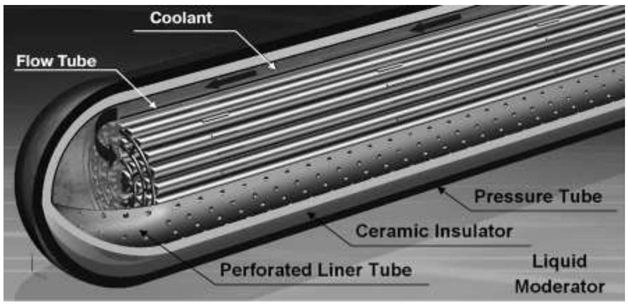


Figure 15. Re-Entrant Channel (REC) with ceramic insulator for SCWR with liquid moderator (drawing prepared by W. Peiman, UOIT)

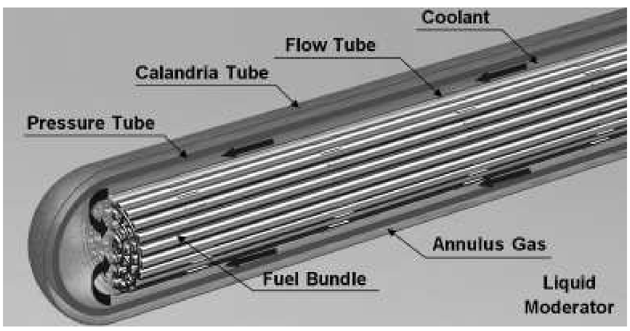
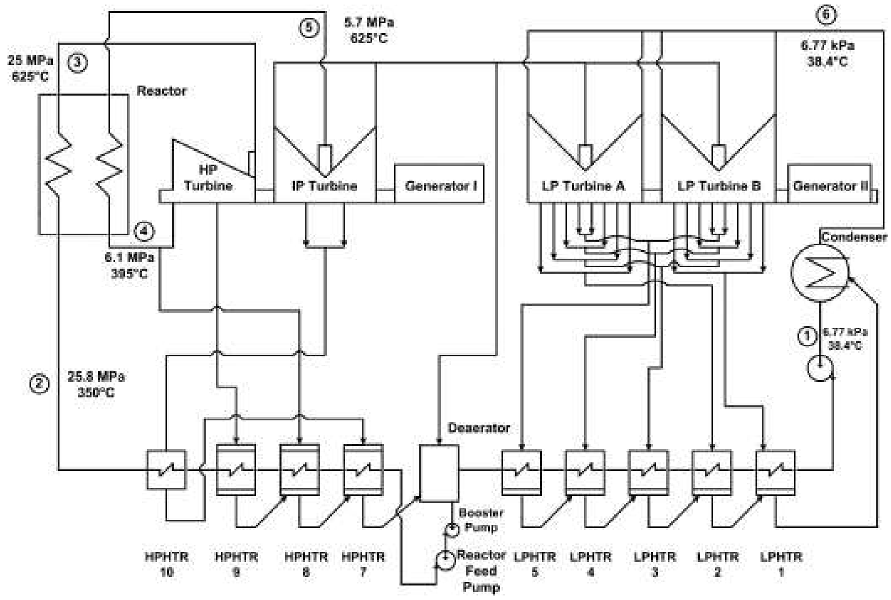
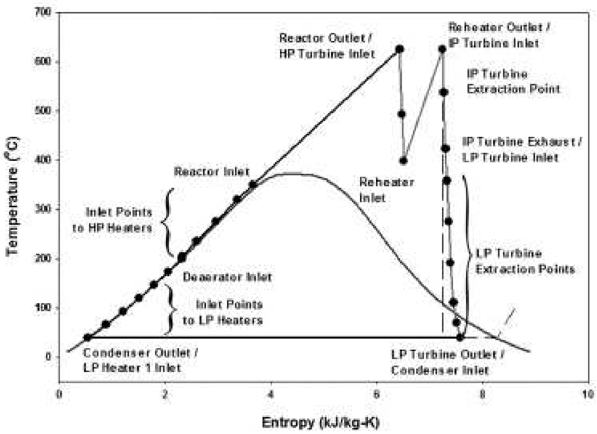


Figure 14. Re-Entrant Channel (REC) with annulus gas as thermal insulation for SCWR with liquid moderator (drawing prepared by W. Peiman, UOIT)

The indirect and dual cycles utilize heat exchangers (steam generators) to transfer heat from the reactor coolant to a turbine. The indirect cycle has a safety benefit of containing potential radioactive particles inside the primary heat-transport system. Also, this cycle arrangement prevents deposition of various substances from the reactor coolant on turbine blades. However, the heat-transfer process through heat exchangers reduces the maximum temperature in the secondary-loop coolant at least by 25–75 °C, thus lowering the efficiency of the cycle. Also, heat exchangers (steam generators) can be quite large units with about 200 thousand square meters of heat transfer surfaces.



(a)



(b)

Figure 16. Direct single-reheat regenerative thermodynamic cycle for 1200-MWel pressure-channel SCW NPP:
(a) Schematic and (b) Temperature-Entropy diagram

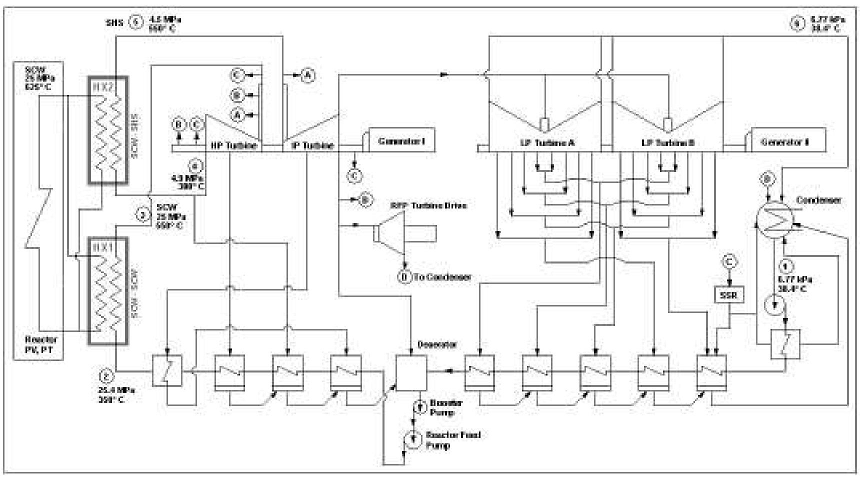
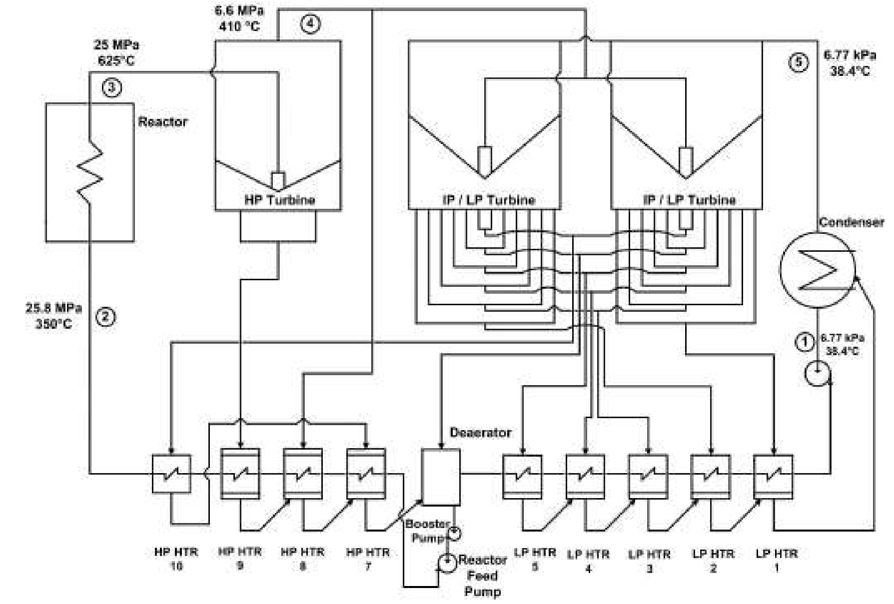
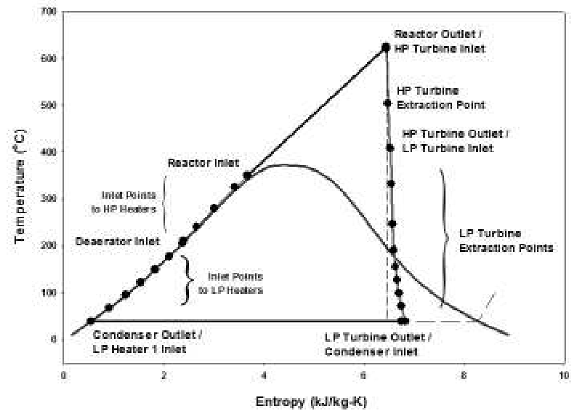


Figure 17. Schematic of in-direct single-reheat regenerative thermodynamic cycle for 1200-MW_{el} pressure-vessel or pressure-channel SCW NPP (drawing prepared by H. Thind, UOIT)



(a)



(b)

Figure 18. Direct no-reheat regenerative thermodynamic cycle for 1200-MW$_{el}$ pressure-vessel or pressure-channel SCW NPP: (a) Schematic and (b) Temperature-Entropy diagram

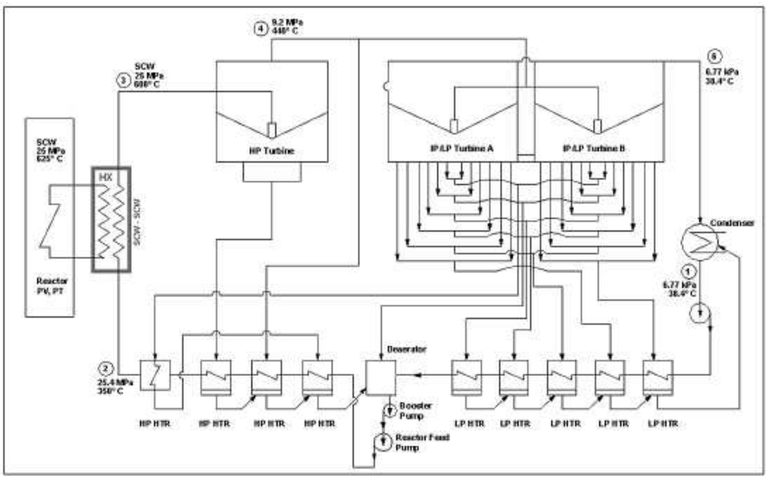


Figure 19. Schematic of in-direct no-reheat regenerative thermodynamic cycle for 1200-MW$_{el}$ pressure-vessel or pressure-channel SCW NPP

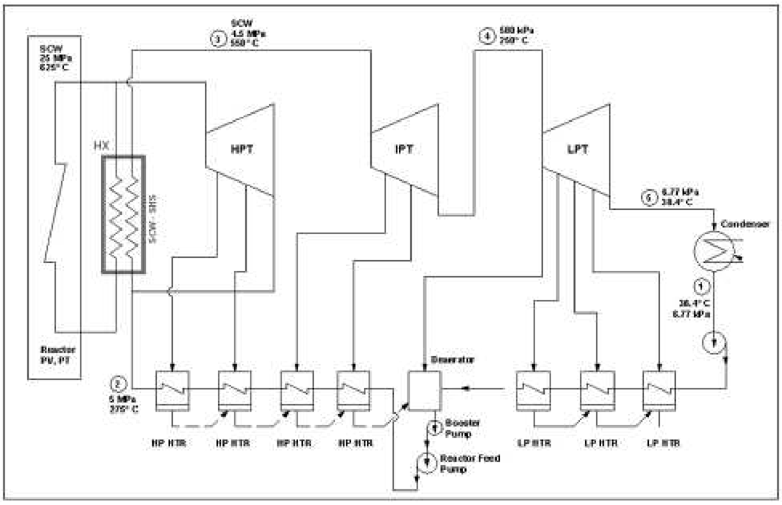


Figure 20. Schematic of dual no-reheat primary (SCW) loop and single-reheat secondary (superheated steam) loop regenerative thermodynamic cycle for 1200-MW$_{el}$ pressure-vessel or pressure-channel SCW NPP

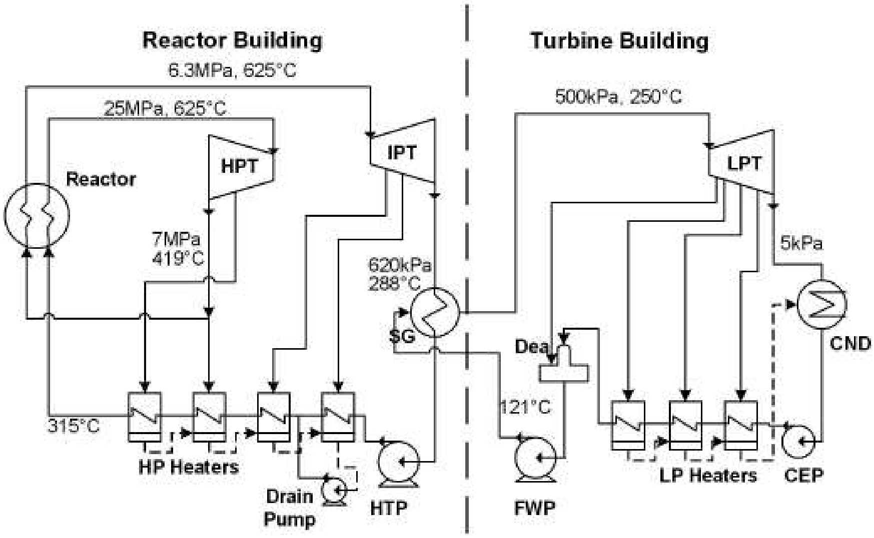


Figure 21. Schematic of dual single-reheat regenerative thermodynamic cycle for 1200-MW$_{el}$ pressure-channel SCW NPP: High-pressure units located in Reactor Building for increased safety

Conclusions

SuperCritical Water-cooled nuclear Reactor (SCWR) is one of six Generation IV concepts developed worldwide. Their design is seen as a natural and ultimate evolution of today's conventional modern water-cooled reactors. Development of SCWRs is based on the following three proven technologies: 1) modern PWRs, which operate at pressures of 15–16 MPa; 2) BWRs, which operate with a once-through or direct cycle; and 3) modern supercritical turbines with pressures about 25 MPa and inlet temperatures up to 625 °C, which operate successfully at coal-fired thermal power plants for more than 50 years. Therefore, this option is one of the most viable Generation IV concepts.

Nomenclature

A area, m^2

D_{hy} hydraulic-equivalent diameter, m; $\left(\dfrac{4\,A_{fl}}{P_{wetted}}\right)$

G mass flux, kg/m^{2}s; $\left(\dfrac{m}{A_{fl}}\right)$

m mass-flow rate, kg/s; (ρV)

P, p pressure, Pa

q	heat flux, W/m^2; $\left(\frac{Q}{A_h}\right)$
T, t	temperature, °C
V	volumetric flowrate, m^3/s

Greek letters

ρ	density, kg/m^3

Subscripts

ave	average
cr	critical
el	elctrical
fl	flow
h	heated
max	maximum
pc	pseudocritical
th	thermal
wt	weight

Abbreviations

AECL	Atomic Energy of Canada Limited
AGR	Advanced Gas-cooled Reactor
AHFP	Axial Heat Flux Profile
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
CEP	Condensate Extraction Pump
CND	CoNDenser
Dea	Deaerator
DOE	Department Of Energy
ECh	Evaporating Channel
FWP	Feedwater Pump
HEC	High Efficiency Channel
HP	High Pressure
HPT	High Pressure Turbine
HTC	Heat Transfer Coefficient
HTP	Heat-Transport Pump
HTR	HeaTeR or High Temperature Reactor
IP	Intermediate Pressure
IPT	Intermediate Pressure Turbine
LP	Low Pressure
LPT	Low Pressure Turbine
LWR	Light-Water Reactor
MOX	Mixed Oxide
MSR	Moisture Separator Reheater
NIKIET	Research and Development Institute of Power Engineering (Moscow, Russia)
NPP	Nuclear Power Plant
PCh	Pressure Channel
PHWR	Pressurized Heavy-Water Reactor
PT	Pressure Tube
PV	Pressure Vessel
PWR	Pressurized Water Reactor
RBMK	Reactor of Large Capacity Channel type (in Russian abbreviations)
REC	Re-Entrant Channel
RFP	Reactor Feedwater Pump
SC	SuperCritical
SCW	SuperCritical Water
SCWR	SuperCritical Water Reactor
SG	Steam Generator
SHS	SuperHeated Steam
SRCh	Steam-Reheat Channel
SRH	Steam ReHeat
UOIT	University of Ontario Institute of Technology
USA	United States of America

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