SCW Pressure-Channel Nuclear Reactor
Some Design Features*


**School of Energy Systems and Nuclear Science
University of Ontario Institute of Technology
2000 Simcoe Str. North, Oshawa ON L1H 7K4 Canada
E-mail: Igor.Pioro@uoit.ca

Abstract
Concepts of nuclear reactors cooled with water at supercritical pressures were studied as early as the 1950s and 1960s in the USA and Russia. After a 30-year break, the idea of developing nuclear reactors cooled with SuperCritical Water (SCW) became attractive again as the ultimate development path for water cooling. The main objectives of using SCW in nuclear reactors are: 1) to increase the thermal efficiency of modern Nuclear Power Plants (NPPs) from 30 – 35% to about 45 – 48%, and 2) to decrease capital and operational costs and hence decrease electrical energy costs (~$1000 US/kW or even less). SCW NPPs will have much higher operating parameters compared to modern NPPs (pressure about 25 MPa and outlet temperature up to 625ºC), and a simplified flow circuit, in which steam generators, steam dryers, steam separators, etc., can be eliminated. Also, higher SCW temperatures allow direct thermo-chemical production of hydrogen at low cost, due to increased reaction rates. Pressure-tube or pressure-channel SCW nuclear reactor concepts are being developed in Canada and Russia for some time. Some design features of the Canadian concept related to fuel channels are discussed in this paper. The main conclusion is that the development of SCW pressure-tube nuclear reactors is feasible and significant benefits can be expected over other thermal-energy systems.

Key words: Supercritical, Nuclear Reactor, Pressure Tube, Thermophysical Properties

1. Introduction

Prior to a general discussion on SuperCritical Water-cooled nuclear Reactors (SCWRs), it is important to define special terms and expressions used at these conditions. Therefore, general definitions of selected terms and expressions related to critical and supercritical pressures are listed below. For better understanding of these terms and expressions a thermodynamic diagram is shown in Fig. 1.

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 the point where the distinction between the liquid and gas (or vapor) phases disappears, i.e., both phases have the same temperature, pressure and volume. The critical point is characterized by the phase state parameters $T_{cr}$, $P_{cr}$ and $V_{cr}$, which have unique values for each pure substance.

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

*Supercritical steam* is actually supercritical water because at supercritical pressures the medium is considered as single-phase fluid. 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.

In general, water-cooled nuclear reactors operating at subcritical pressures (7 – 16 MPa) started to be developed at the end of 1940s and have been used for the last fifty years. At the same time, i.e., as early as the 1950s, concepts of SCWRs were studied in the USA and Russia. Typical operating conditions of SCWRs, PWRs, CANDU-6 reactors and BWRs are shown in Fig. 2 for reference purposes. Now several countries (Canada, European Union, Japan, Korea, Russia, USA and others) have started to work in that direction, building on the successful international deployment of supercritical and ultra-supercritical “steam” generators and turbines. However, none of these newer SCWR concepts is expected to be implemented before 2020 – 2025.

The main objectives of using SuperCritical Water (SCW) in nuclear reactors (1)-(11) are: 1) using the increased temperatures and perhaps using direct-cycle turbines to increase the thermal efficiency of modern NPPs from 30 - 35% to 45 – 48% for electricity production, or more than 50% with co-generation of heat, and 2) using a more compact design that results in decreased capital and operational costs and hence in decreased capital and electrical energy costs (~$1000 US/kW or even less).

The development challenges are to minimize the higher costs of materials needed at the higher temperatures, and to enhance safety and performance margins despite the increased pressures, while retaining the economic advantage.

SCW NPPs (8) will have much higher operating parameters compared to modern NPPs (pressure about 25 MPa and outlet temperature up to 625°C) (Fig. 2), and can use a simplified direct-cycle flow circuit in which steam generators, steam dryers, steam separators, etc., are eliminated. Importantly, mass-flow rates are drastically reduced for the same thermal output. Also, if superheating of steam (2)(8) is adopted at lower pressures, even higher temperatures (limited only by materials’ corrosion rates) could allow direct thermo-chemical production of hydrogen, due to increased reaction rates, which can be utilized in fuel cells, hydrogen vehicles and as a part of chemical processing or hydrocarbon upgrading.

The SCWR concepts (8) follow two main types: the use of either: (a) a large reactor pressure vessel (PV) (3) with a wall thickness of about 0.5 m to contain the reactor core (fuelled) heat source, analogous to conventional Light Water Reactors (LWRs); or (b) distributed pressure tubes (PTs) or channels analogous to conventional Heavy Water Reactors (HWRs) (1)(2)(4)-(10).
Within those two main classes (PV and PT) \(^{(6)}\), pressure-tube reactors are designed to be more flexible to flow, flux and density changes than the PV reactors. This makes it possible to use the experimentally confirmed, better solutions developed for these reactors. The main ones are fuel re-loadings and channel-specific flow-rate adjustments or regulations. A design whose basic element is a channel or tube, which carries a high pressure, has an inherent advantage of greater safety than large vessel structures at supercritical pressures.
Nomenclature

\( A_f \): flow area, \( m^2 \)

\( c_{pb} \): bulk-fluid specific heat at constant pressure, J/kg K

\( \overline{c_p} \): average specific heat within range of \((T_w - T_b)\); \( \left( \frac{H_w - H_b}{T_w - T_b} \right) \), J/kg K

\( D \): inside diameter, m

\( D_{hy} \): hydraulic-equivalent diameter, m

\( D_{rod} \): rod diameter, m

\( G \): mass flux, \( \frac{kg}{m^2 s} \);

\( H_b \): bulk-fluid enthalpy, J/kg

\( H_{pc} \): pseudocritical bulk-fluid enthalpy, J/kg

\( H_w \): fluid enthalpy at wall temperature, J/kg

\( h \): heat transfer coefficient, W/m\(^2\)K

\( k_b \): bulk-fluid thermal conductivity, W/m K

\( k_w \): fluid thermal conductivity at wall temperature, W/m K

\( m \): mass-flow rate, kg/s

\( Nu \): Nusselt number; \( \left( \frac{h D_{hy}}{k_b} \right) \)

\( P \): pressure, MPa

\( P_{cr} \): critical pressure, MPa

\( P_{pc} \): pseudocritical pressure, MPa

\( Pr \): Prandtl number; \( \left( \frac{\mu_b c_{pb}}{k_b} \right) \)

\( \overline{Pr} \): average Prandtl number within the range of \((T_w - T_b)\); \( \left( \frac{\mu_b \overline{c_p}}{k_b} \right) \)

\( Q_{channel} \): channel power, W

\( q_{ave} \): average heat flux, W/m\(^2\)

\( Re \): Reynolds number; \( \left( \frac{G D}{\mu_b} \right) \)

\( T_{cr} \): critical temperature, °C

\( T_{in} \): inlet bulk-fluid temperature, °C

\( T_{max} \): maximum temperature, °C

\( T_{out} \): outlet bulk-fluid temperature, °C

\( T_{pc} \): pseudocritical temperature, °C

\( T_w \): wall temperature, °C

\( V_{cr} \): critical specific volume, m\(^3\)/kg

Greek symbols

\( \delta_w \): wall thickness, mm

\( \mu_b \): bulk-fluid dynamic viscosity, Pa s

\( \mu_w \): fluid dynamic viscosity at wall temperature, Pa s

\( \rho_b \): bulk-fluid density, kg/m\(^3\)

\( \rho_w \): fluid density at wall temperature, kg/m\(^3\)
2. Basis for Using SCW in Nuclear Reactors

The design of SCWRs is seen as the logical, natural and ultimate evolution of today’s conventional water-cooled nuclear reactors but using thermodynamic cycles, materials and turbines, which are already proven in SCW fossil steam generators. Reviews of the SCWR concepts have been given in (5)-(8)-(11).

It is known that the heat-transfer coefficient (HTC) from a fuel element to a gaseous coolant (SCW is considered as a dense-gas substance) is lower than that in conventional water-cooled nuclear reactors for the same mass flux (2) (8) (10). Hence the fuel centerline temperature might be higher in a SCWR than in a subcritical water-cooled nuclear reactor.

There are relatively very few published experimental data on SCW heat transfer in bundles (8)(12)-(14). Therefore, a focus of this paper is on preliminary heat-transfer calculations based on proven heat-transfer correlations obtained with SCW in vertical bare tubes.

3. Thermophysical Properties of Water near Critical and Pseudocritical Points

Heat transfer at critical and supercritical pressures is influenced by significant changes in thermophysical properties. For many fluids that are used at supercritical conditions, their physical and thermophysical properties are well established. This is especially important for a creation of generalized correlations in non-dimensional form, which allows experimental data for several fluids to be combined into one set. The most significant thermophysical property variations occur near the critical and pseudocritical points.

Thermophysical properties of water at different pressures and temperatures, including the supercritical region, can be calculated using the NIST software (15).
The specific heat of water (as well as of other fluids) has a maximum value in the critical point. The exact temperature that corresponds to the specific heat peak at pressures above the critical pressure is known as the pseudocritical temperature.

General trends of various properties near the critical and pseudocritical points are being illustrated on a basis of those of water as seen in Fig. 3, because water is considered as a primary choice of coolant in SCWRs, and because these trends are similar for many fluids.

In general, all thermophysical properties undergo significant changes near the critical and pseudocritical points. Near the critical point, these changes are dramatic.

In the vicinity of pseudocritical points, with an increase in pressure, these changes become less pronounced. It can also be seen in Fig. 3 that properties such as density (the same apply to dynamic viscosity) undergo a significant drop (near the critical point this drop is almost vertical) within a narrow temperature range (~50ºC), while properties such as specific enthalpy (the same applies to kinematic viscosity) undergo a sharp increase. Thermal conductivity and Prandtl number (the same applies to volume expansivity and specific heat) have a peak near the critical and pseudocritical points. The magnitude of these peaks decreases very quickly with an increase in pressure (for example, see thermal conductivity profile at 25 MPa).

4. Pressure-Tube SCWRs

Pressure-channel or pressure-tube SCWR concepts developed in Russia and Canada (see Table 1) avoid using a thick-wall vessel, and allow, in principle, five key features for safety and performance:

1. 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.
2. Use of multi-pass reactor flows, so that reheat and superheat are possible while still keeping the pressure channel cool. Thus the system can produce process heat on
demand.
3. Use of optimized fuel bundles, so the fuel cycle, cladding, flow and heat transfer in each channel can be adjusted almost independently, thus ensuring stable and predictable performance.
4. Reactor size (and thermal power) can be adjusted by simply changing the number of channels, from 300 MWₑ to 1400 MWₑ depending on the customer site, financing and product mix application.
5. Negative void, power and temperature coefficients of reactivity, which are dependable from the choice of lattice and enrichment, and flat power and temperature profiles due to interlacing the flow directions, ensure passive inherently safe operational and performance characteristics.

These features together set the fuel design, the channel power, the core lattice pitch, the enrichment, and the flow circuit parameters. The coolant of choice is water. A thermal neutron spectrum is produced with either a liquid heavy-water moderator (primary choice) or a solid moderator using graphite or zirconium hydride.

Therefore, two complementary concepts of several SCW PT reactors are listed in Table 1: one from Canada (AECL) and one from Russia (RDIPE). Also, a Temperature-Entropy diagram of the future SCWR NPP is shown in Fig. 4 for reference purposes.

5. Canadian SCWR Concept

The SCW CANDU nuclear reactor concept (see Fig. 5) is a PT-type reactor with a thermal spectrum, using a D₂O moderator and a bi-directional (one pass) flow core. The direct cycle was downselected using a thermo-economic analysis that considered the offsetting influence of cost reductions and thermal efficiency gains. Major parameters

![Fig. 4 Temperature-Entropy diagram of future SCWR nuclear power plant (currently coal-fired supercritical-pressure thermal power plants work based on similar diagram) (data for this diagram are courtesy of U. Zirn, Hitachi America, Ltd.).](image-url)
of the SCW CANDU reactor are listed in Table 1. The reactor works as described below. Supercritical water at a temperature of 350°C (i.e., below the pseudocritical temperature of 384.9°C) and at a supercritical pressure of 25 MPa, from a circulation pump enters the reactor core and heats up by the heat of fission to 625°C. After that, it is directed to a supercritical turbine to perform useful work and returns back through the cooler to the circulation pump.

Table 1 Modern concepts of pressure-tube nuclear reactors cooled with SCW

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Unit</th>
<th>SCW CANDU®¹</th>
<th>KP-SKD</th>
</tr>
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<tbody>
<tr>
<td>Reference</td>
<td>–</td>
<td>(4)</td>
<td>(5)</td>
</tr>
<tr>
<td>Country</td>
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<td>Canada</td>
<td>Russia</td>
</tr>
<tr>
<td>Organization</td>
<td>–</td>
<td>AECL</td>
<td>RDIPE</td>
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<td>electrical</td>
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<tr>
<td>Thermal efficiency</td>
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<td>Pressure</td>
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<td>25</td>
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<tr>
<td>(T_{in}) coolant</td>
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<tr>
<td>(T_{out}) coolant</td>
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<td>diameter</td>
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<tr>
<td>Enrichment</td>
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<td>6</td>
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<tr>
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<td>St. st.</td>
</tr>
<tr>
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</tr>
<tr>
<td># of fuel rods in bundle</td>
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<td>18</td>
</tr>
<tr>
<td>(D_\text{rod} / \delta_w)</td>
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<td>11.5 and 13.5*</td>
<td>10/1</td>
</tr>
<tr>
<td>(T_{max}) cladding</td>
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<td>&lt;850</td>
<td>700</td>
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<tr>
<td>Moderator</td>
<td>–</td>
<td>D₂O</td>
<td>D₂O</td>
</tr>
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</table>

* For 43-element bundle.

In general, the SCW CANDU reactor (¹) can produce electrical energy as the main product, and process heat, industrial isotopes, drinking water and even hydrogen as supplementary products. Safety and other systems seem to be similar to those implemented in modern nuclear reactors.

High pressures and temperatures inside the reactor core require a new design of the fuel channel (⁷). Current CANDU reactors use a channel design that consists of a pressure channel that is insulated from the cool heavy-water moderator by an annulus gap filled with inert gas and a calandria tube (see Fig. 6).

Using supercritical water as a coolant requires a major design change to allow continued use of the low-neutron absorbing material currently used in CANDU reactor fuel channels (¹). The insulated pressure channel (see Fig. 7) is a key technology that is needed to make use of supercritical water in CANDU reactors feasible (however, other options such as a solid moderator using graphite or zirconium hydride are not ruled out).

¹ CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL).
Multiple products are key to sustainable future and competitive designs

In this design, the pressure channel \(^{(1)}\) is insulated from the coolant by using an internal layer of low-neutron absorbing material. This simplifies the fuel-channel design by eliminating the calandria tube and the insulating annulus gap. Furthermore, an internally insulated pressure channel 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.

The exhaust of the high-pressure (HP) turbine can be used for process heat or electricity production in a low-pressure (LP) turbine. More recent analyses and concepts \(^{(2)}\) also consider the addition of multi-pass steam-superheating channels at lower pressure.

6. Preliminary Results of Heat-Transfer Calculations

Heat-transfer calculations were performed based on the SCW CANDU reactor data listed in Table 1 and NIST properties software \(^{(15)}\). Also, cross-section average thermalhydraulics parameters were used in calculations. Results obtained at the uniform
axial heat-flux distribution are shown in Figs. 8–12: (a) Temperature and thermophysical properties profiles along the heated length of fuel channel are shown in Figs. 8–10; and (b) Temperature and HTC profiles along the heated length of fuel channel are shown in Figs. 11 and 12.

Fig. 7 Details of Insulated Fuel-Channel Concept (Cross-Sectional View) (based on figure from (7))

Fig. 8Bulk-Fluid Temperature and Selected Thermophysical Properties \((^{(1)})\) Profiles along Heated Length of Fuel Channel
Fig. 9 Bulk-Fluid Temperature and Selected Thermophysical Properties (15) Profiles along Heated Length of Fuel Channel

Fig. 10 Bulk-Fluid Temperature and Selected Thermophysical Properties (15) Profiles along Heated Length of Fuel Channel
In general, at subcritical pressures forced-convection single-phase heat transfer to fluids can be described with the conventional Dittus-Boelter correlation \(^{(17)}\), as introduced by McAdams \(^{(19)}\) (for details, see Winterton \(^{(20)}\)):
However, at critical and supercritical pressures the Dittus-Boelter correlation overpredicts significantly HTC within the critical or pseudocritical regions (8) (see Fig. 11), where all thermophysical properties undergo large variations within a narrow temperature range (see Figs. 8-10). A peak or “hump” in the HTC predicted by the Dittus-Boelter correlation is due to a peak or “hump” in Prandtl number (see Fig. 8), which in turn is due to a significant peak in the specific heat near the critical and pseudocritical points (see Fig. 9). Therefore, other correlations have to be used at these conditions.

In general, many of these correlations are based on the conventional Dittus-Boelter-type correlation (see Eq. 1) in which the regular specific heat is replaced with the averaged specific heat within a range of \((T_w - T_b)\): 

\[
\overline{C_p} = \left( \frac{H_w - H_b}{T_w - T_b} \right) \text{J/kg K} \tag{1}
\]

Also, additional terms, such as \(\left( \frac{k_b}{k_w} \right)^k; \left( \frac{\mu_b}{\mu_w} \right)^m; \left( \frac{\rho_b}{\rho_w} \right)^n\); etc., can be implemented into correlations to account for significant variations in thermophysical properties within a cross section due to a non-uniform temperature profile.

One of such HTC correlations is the Bishop et al. correlation (18) (see Fig. 12):

\[
\begin{align*}
\text{Nu}_x &= 0.069 \text{Re}_x^{0.9} \text{Pr}_x^{0.66} \left( \frac{\rho_w}{\rho_b} \right)_x^{0.43} \left( 1 + 2.4 \frac{D}{x} \right); \\
\text{Nu}_D &= 0.0023 \text{Re}_D^{0.8} \text{Pr}_D^{0.4}
\end{align*}
\tag{2}
\]

where \(x\) is the axial location along the heated length. The last term in Eq. (2) is responsible for the entrance effect. However, the entrance effect has not been accounted for in the present calculations, because usually fuel bundles have high level of turbulization from the inlet due to various appendages (endplates, spacers, bearing pads, etc.).

This correlation was obtained with supercritical water flowing upward inside tubes and annuli within the following range of operating parameters: pressure 22.8 – 27.6 MPa, bulk-fluid temperature 282 – 527ºC, mass flux 651 – 3662 kg/m²s and heat flux 0.31 – 3.46 MW/m²; which is close to that of SCWRs. The Bishop et al. correlation has an accuracy of ±15%, and, therefore, is considered to be a reliable correlation.

It should be noted that usually generalized correlations, which contain fluid properties at a wall temperature, require iterations to be solved, because there are two unknowns: 1) HTC and 2) the corresponding wall temperature. Therefore, the initial wall-temperature value at which fluid properties will be estimated should be “guessed” to start iterations.

The Bishop et al. correlation was obtained in bare tubes. Therefore, these heat-transfer calculations can be considered as a conservative approach (in general, heat transfer in fuel bundles will be enhanced with various types of appendages).

Variations in the HTC along the heated length of the fuel channel, i.e., an HTC decreasing beyond the pseudocritical point, are thought to be due to a special behavior of supercritical fluid below the pseudocritical point (high-density fluid, i.e., “liquid”) and beyond (low-density fluid, i.e., a near-perfect “gas”).

The heat-transfer calculations (Figs. 11 and 12) showed that even using the conventional HTC correlations obtained in water flowing in vertical bare tubes, the maximum centerline fuel temperature for the larger diameter fuel element (OD 13.5 mm) will be within the reasonable limits for current nuclear reactors, i.e., about 1950ºC.

This paper considers only the preliminary steady-state HTC calculations based on the bare-tube correlations. Therefore, heat transfer in fuel bundles at steady-state, transient and accident conditions including regimes of deteriorated heat transfer must be considered in future investigations.
7. Conclusion

The preliminary calculations showed that development of SCW pressure-tube nuclear reactors is feasible and significant benefits can be expected. However, there is still a long way towards the final SCWR design.

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