HEAT LOSS ANALYSIS OF A RE-ENTRANT FUEL CHANNEL FOR SCWR TYPE REACTORS

J. Samuel, G.D. Harvel, and I. Pioro

Faculty of Energy Systems and Nuclear Science
University of Ontario Institute of Technology
2000 Simcoe St. North, Oshawa, Ontario, L1H 7K4 Canada
jeffrey.samuel@uoit.ca; glenn.harvel@uoit.ca; igor.pioro@uoit.ca

Abstract

One of the fuel-channel design concepts for a Pressure Tube (PT) SuperCritical Water-cooled Reactor (SCWR) currently under development is called the re-entrant fuel channel. The re-entrant fuel-channel design consists of two tubes, the inner tube (flow tube) and the pressure tube. The fuel bundles are placed in the inner tube. The flow and pressure tubes form an annulus through which flows the primary coolant. The coolant flows through the annulus receiving heat from the inner tube from one end of the channel to the other. At the far end, the flow will reverse direction and enter the inner tube, and hence the fuel bundle-string. The results from a numerical analysis performed using MATLAB indicate that the total heat loss from the re-entrant channel will be too high when no insulating barrier is included in the design. Thus, the design of the re-entrant fuel channel needs an insulating barrier for normal operating conditions.

1. Introduction

There are a number of new concepts for nuclear reactors being developed worldwide as part of the Generation IV collaboration project. One such concept is a SuperCritical Water-cooled Reactor (SCWR), which will have a thermal efficiency of about 50% [1]. SCWRs will use SuperCritical Water (SCW) as a coolant and operate at higher temperatures and pressures compared to those of current water-cooled reactors. While current PWRs operate at a coolant pressure within 10 – 16 MPa, SCWRs will operate at about 25 MPa. The coolant would thus pass through a pseudocritical region somewhere along the channel [2].

Two types of SuperCritical Water-cooled Reactor (SCWR) concepts are a large Pressure Vessel (PV) and a Pressure Tube (PT) reactors. The current fuel-channel reference design for a PT reactor consists of a bundle, ceramic layer and pressure tube [3]. The outer surface of the pressure tube is in contact with the moderator, while a perforated liner protects the ceramic layer from the bundles, through which flows the primary coolant. While such a design may work, there are concerns with the construction, assembly and maintenance of the fuel channel. Alternative fuel-channel design concepts are under development to address these concerns and to explore other opportunities [3], [4].

A key element to consider in designing the fuel channel is the minimization of heat loss to the moderator at normal operating conditions. In this paper, heat losses to a heavy-water moderator are evaluated for the Re-Entrant Channel (REC).

2. Design

A new fuel-channel-design concept under consideration consists of two tubes; the inner tube (flow tube) and the outer tube (pressure tube). Fuel bundles are installed in the inner tube. The coolant first flows through the annulus gap between the pressure tube and the inner tube from one end of the channel to the other before reversing direction and flowing through the inner tube. Thus, the fuel channel effectively becomes a double-pipe heat exchanger in which the annulus is actually a preheater. This configuration is known as the REC. The inner tube is referred to as the hot side, and the annulus is referred to as the cold side of the fuel channel. Figure 1 shows the proposed fuel-channel arrangement in a calandria-type vessel. Note that the design could also be used in a vertical configuration [5].

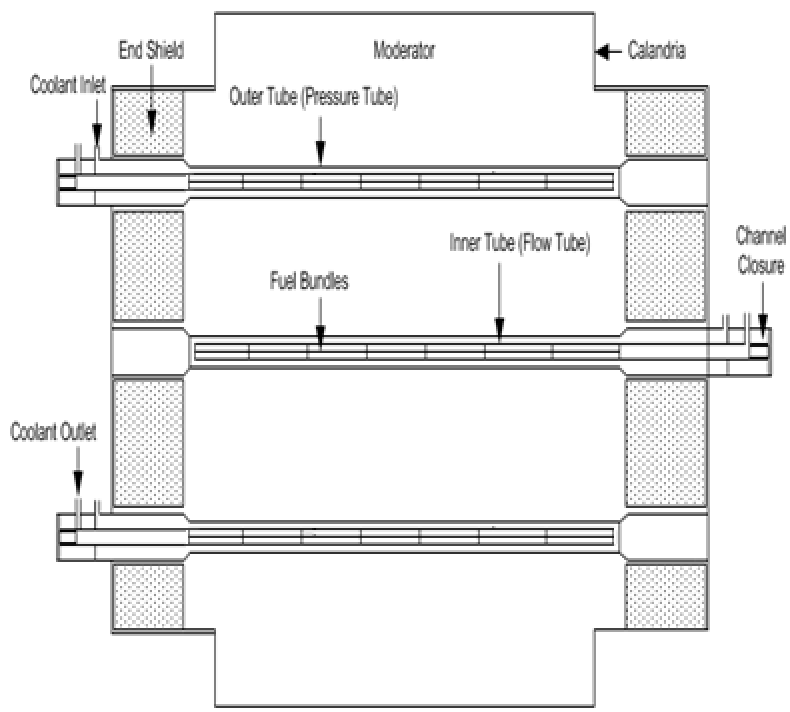

Figure 1: Re-entrant fuel channel in calandria vessel.

The REC is shown in Figure 2. For this design, a reference channel length of 5.772 m was chosen.

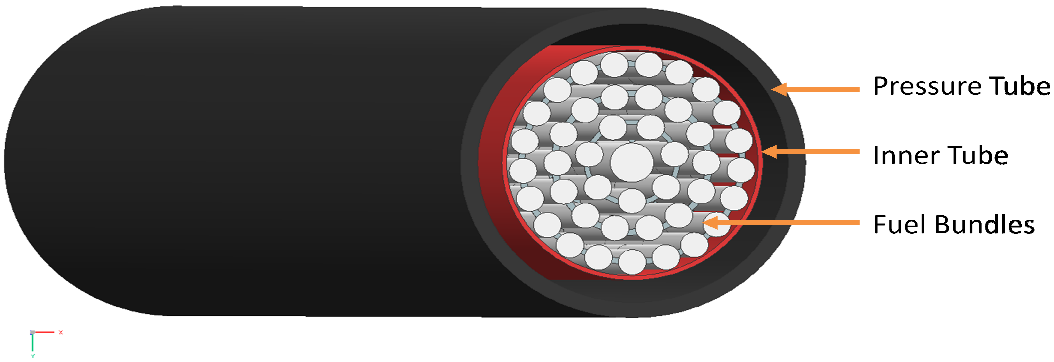

Figure 2a: Re-entrant fuel channel.

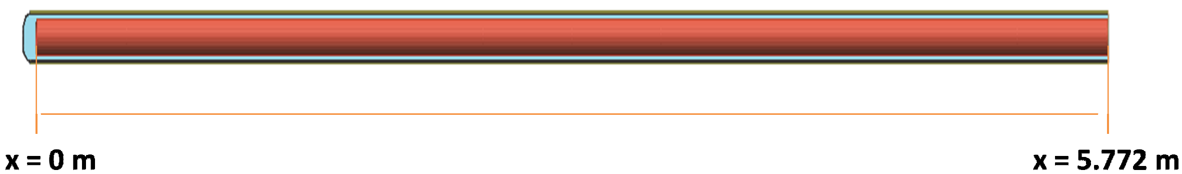

Figure 2b: Side view of re-entrant fuel channel showing heated length.

Figures 3a and 3b show the front and rear ends of the REC, respectively. The REC is only refueled from one end [4]. The reference flow-tube inner and outer diameters are 103.5 mm and 107.5 mm, respectively, while the reference pressure tube inner and outer diameters are 127.9 mm and 149.9 mm respectively. Stainless Steel-304 has been chosen as a reference material of construction of the flow and pressure tubes for this analysis.

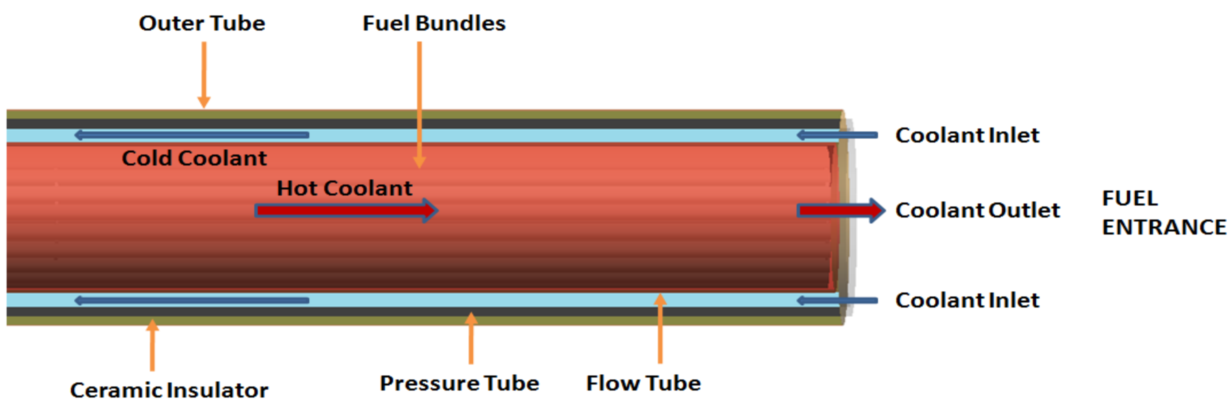

Figure 3a: Front end of re-entrant fuel channel.

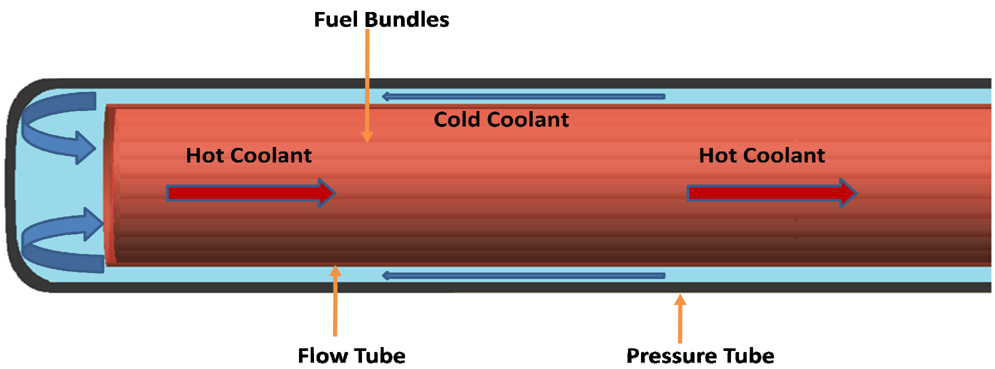

Figure 3b: Rear end of re-entrant fuel channel.

3. Heat-Transfer Model

The reference reactor model used in the heat-transfer analysis is a 1200-MW_e SCWR with 300 fuel channels. The inlet temperature of the coolant is 350°C, and the outlet temperature of the coolant varies according to the thermal power per channel with the maximum value of 8.5 MW_{th}. The pressure drop along the channel is not accounted for due to low mass fluxes compared to current water-cooled reactors. The pressure was assumed to be 25 MPa for determination of fluid properties in this analysis. The mass-flow rate of the coolant is chosen as 4.37 kg/s for typical operating conditions. For consistency with earlier work, Variant-18 bundles are used for the heat-transfer analysis [4]. The moderator was assumed to have a bulk-fluid temperature of 80°C and a pressure of 200 kPa. Moderator properties were calculated at 100°C for the heat-loss analysis to be more representative of fluid conditions at the wall.

A cross-sectional 1-D averaged numerical model was developed using MATLAB in which the fuel channel was divided into 121 nodes; 60 for the cold side, 60 for the hot side and one for the re-entrant mixing node, which is the region where the coolant from the annulus changes direction and flows in the inner tube. As the inlet temperature and pressure are known, the enthalpy is easily obtained using NIST REFPROP software. The enthalpies for the other nodes are obtained via heat balance using Equation (1).

$$H_{node} = H_{previous\ node} + \frac{\dot{Q}}{\dot{m}} \quad (1)$$

where H_i is the enthalpy of the node, $\dot{Q}$ is the heat transfer rate and m is the mass flow rate. The difference between the nodal power and the heat loss to the coolant is used to find $\dot{Q}$. Heat is transferred from the fuel to the hot coolant and from the flow tube to the cold coolant via forced convection, while heat is transferred via conduction in the flow and pressure tubes. As the enthalpy and pressure for each node is known, the bulk-fluid temperatures and thermophysical properties for the fluid are obtained using NIST REFPROP software. As the properties of the fluid in each node are constant, the energy equation can be written as follows [6]:

$$u\frac{\partial T}{\partial x}+v\frac{\partial T}{\partial y}=\alpha\frac{\partial^2 T}{\partial y^2} \quad (2)$$

where u and v are the velocities, and α is the thermal diffusivity. Equation (2) can be written in a non-dimensionalized form that can be used to derive an equation for the Nusselt number, which represents the enhancement of heat transfer through a fluid layer as a result of convection relative to conduction. The Mokry et al. correlation (Equation (3)) is used to calculate the Nusselt number for the coolant at each node [7].

$$\mathbf{Nu}_x = 0.0061\,\mathbf{Re}_x^{0.904}\,\overline{\mathbf{Pr}}_x^{0.684}\,\left(\frac{\rho_w}{\rho_b}\right)_x^{0.564} \quad (3)$$

where $\overline{\mathbf{Pr}}_x$ is defined as the average Prandtl number and the subscript x is the axial location along the heated length. The Mokry et al. correlation has been developed for vertical bare tubes, however, it is used in this analysis as a conservative approach for fuel bundles.

As the wall temperature of the outer sheath is much higher than the coolant within the hot side, the averaged specific heat and averaged Prandtl number are used to calculate the flow parameters of the hot coolant as they account for the difference in temperature. The averaged specific heat and the averaged Prandtl number for each node in the hot side can be calculated using the following equations:

$$\bar{\boldsymbol{C}}_{\boldsymbol{p}} = \frac{H_w - H_b}{T_w - T_b} \quad (4)$$

$$\overline{\mathbf{Pr}} = \frac{\mu_{hs}\cdot\bar{C}_p}{k_{hs}} \quad (5)$$

Heat transfer coefficients for the hot and cold coolant are calculated from the Nusselt number using Equation (5).

$$HTC = \frac{Nu_x\cdot k}{D_{hy}} \quad (6)$$

The thermal conductivity of SS-304 for the flow tube and the pressure tube is calculated using Equation (6) [8].

$$k = 2.0\,\times 10^{-6}T^2 + 0.0134T + 10.689 \quad (7)$$

Heat loss from the channel to the moderator is assumed to be via free convection. The Nusselt number for free convection is calculated using the Churchill and Chu correlation as shown below [9].

$$Nu^{free} = \left\{0.60 + \frac{0.386{Ra_D}^{1/6}}{\left[1+\left(\frac{0.559}{Pr}\right)^{9/16}\right]^{8/27}}\right\}^2 \quad (8)$$

where the Rayleigh number is calculated using Equation (8) [10].

$$Ra_D = \frac{g\beta\left(T_{o\,(PT)} - T_{mod}\right)D^3}{\nu\alpha} \tag{9}$$

where $T_{o\,(PT)}$ is the temperature of the outer surface of the pressure tube and T_{mod} is the moderator temperature.

Once the temperature of the coolant at each node is known, and the total thermal resistances of the fuel-channel components are calculated, the heat loss to the moderator can be calculated using:

$$q = \frac{T_{os} - T_{mod}}{R_{total}}$$

$$q = \frac{T_{os} - T_{mod}}{R_{hot\,side} + R_{flow\,tube} + R_{cold\,side} + R_{pressure\,tube} + R_{moderator}}$$

$$q = \frac{T_{os} - T_{mod}}{\left(\frac{1}{hA}\right)_{hs} + \left(\frac{\ln\left(\frac{r_o}{r_i}\right)}{2\pi Lk}\right)_{FT} + \left(\frac{1}{hA}\right)_{cs} + \left(\frac{\ln\left(\frac{r_o}{r_i}\right)}{2\pi Lk}\right)_{PT} + \left(\frac{1}{hA}\right)_{mod}} \tag{10}$$

where T_{os} is the outer sheath temperature, and R_{total} is the sum of the thermal resistances of the hot side, flow tube, cold side, pressure tube and moderator [10]. The above equation can also be used to calculate the outer sheath temperature and the inner and outer surface temperatures of the flow and pressure tubes if the relevant terms are considered.

4. Analysis and Discussion

Heat-transfer analysis was performed in MATLAB with fluid properties transferred from NIST REFPROP. The fuel-channel power, the thickness of the flow tube and the material of construction of the flow tube were varied for this analysis, while heat flux was considered to be uniform. The coolant enters the cold side of the double-pipe fuel channel at *x* = 5.772 m, and enters the hot side at *x* = 0 m.

Figure 4 shows temperature profiles of the coolant, the inner and outer surfaces of the flow and pressure tubes and the temperature profile of the outer sheath along the heated length of the channel. The temperature of the coolant on the cold side increases approximately linearly as expected as the heat source is the outer wall of the flow tube. The coolant temperature on the hot side increases slowly at first and then dramatically after approximately 2 m of fuelled length. The main reason for this, as will be seen in later figures, is the transition through the pseudocritical point. The outer-sheath temperature increases at first, decreases and then increases again. This behaviour is expected for a uniform heat flux and constant mass flux as there is a significant improvement in heat transfer as the coolant passes through the pseudocritical point [7]. The outer-sheath temperature is below the sheath melting temperature limit of 850°C.

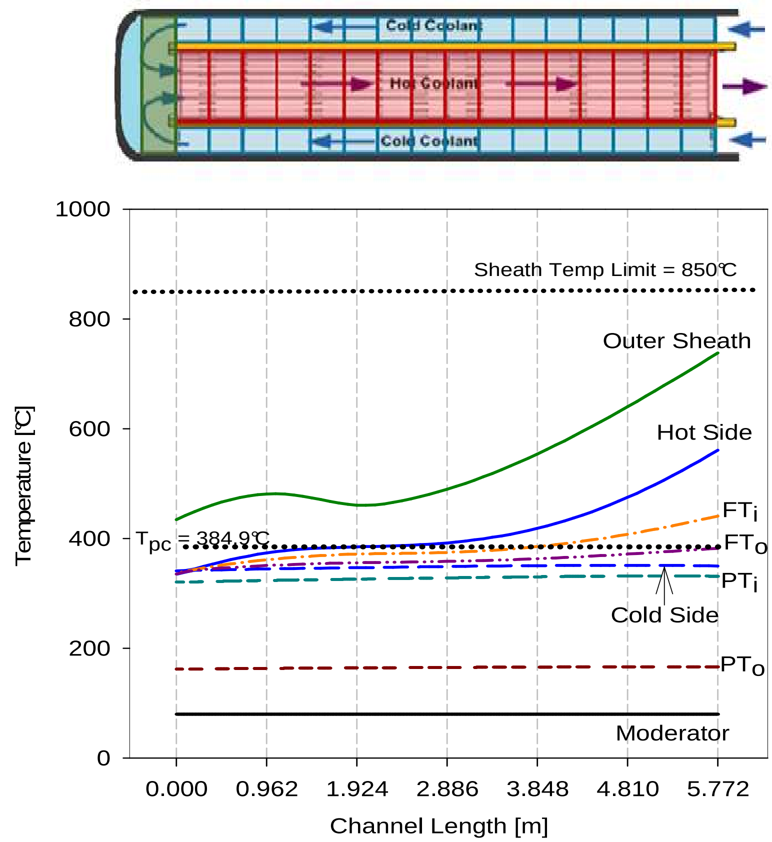

Figure 4: Temperature profile along channel length for channel power of 8.5 MW_{th}.

The location of the pseudocritical point in the channel can be identified by the peak in the specific heat profile for the coolant as seen in Figure 5. Thermophysical properties, such as thermal conductivity, and specific heat drastically change within the pseudocritical region as seen in Figure 5. The Prandtl number, which is based on thermophysical properties, also changes drastically within the pseudocritical region. These changes result in the variation in slopes of the coolant and outer sheath temperature profiles at these locations in Figure 4. The thermophysical properties are calculated using the bulk-fluid temperature along the channel.

The average Prandtl number and average specific heat that are required for Equations (4) and (5) were calculated using bulk-fluid and wall temperatures and their profiles can be seen in Figure 6. The jump in these profiles at the entrance of the hot side can be accounted for by the fact that as the coolant enters the fuelled region, there is a rapid increase in wall temperature of the fuel sheath. The heat transfer coefficient profile which is calculated using Equation (6) is also shown in Figure 6.

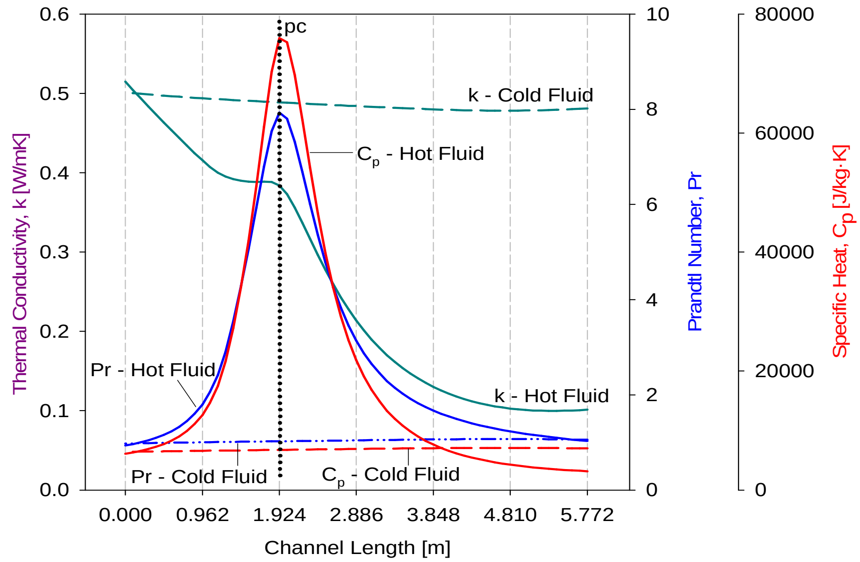

Figure 5: Specific heat, thermal conductivity and Prandtl number profiles along fuel-channel for channel power of 8.5 MW.

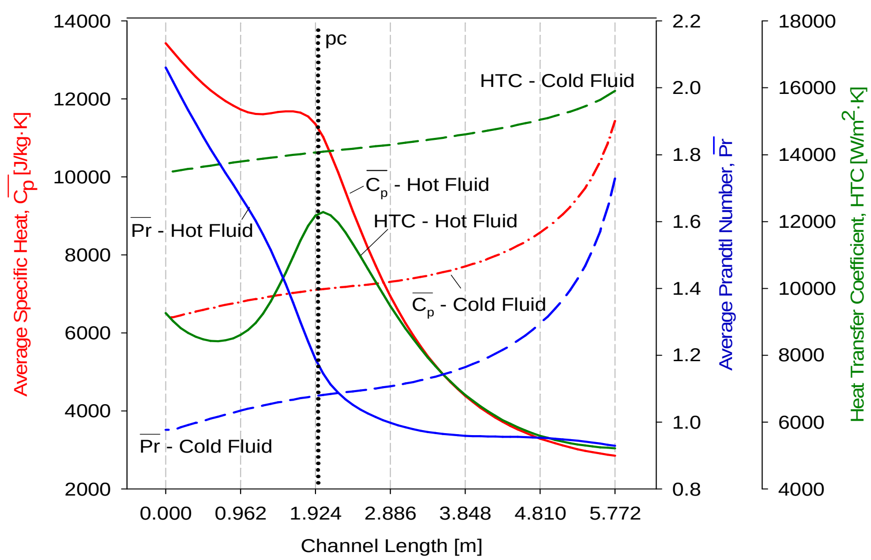

Figure 6: Average specific heat, average Prandtl number and HTC profiles along fuel-channel for channel power of 8.5 MW.

The temperature gradients across the radial section of the reference REC at x = 0 m, x = 1.924 m and x = 5.772 m are shown in Figure 7, where x = 1.924 m is the location of the pseudocritical point. The dotted line indicates the expected temperature profile in the liquid. As the total heat loss is proportional to the change in temperature, the figure indicates that the heat loss across the pressure tube to the moderator will be high. It is also important to note that the figure indicates boiling in the moderator, which is undesirable.

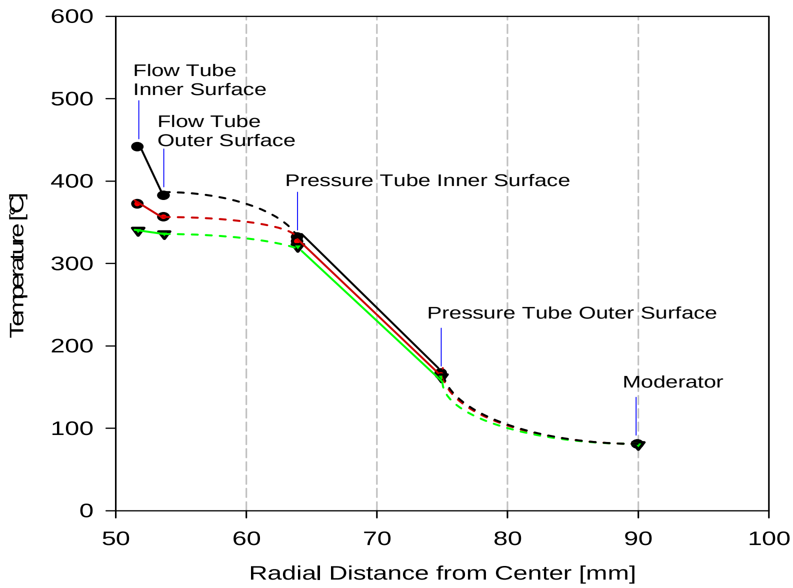

Figure 7: Temperature gradients along radial distance from center for re-entrant channel.

The total heat loss from the cold side of the reference REC to the moderator was approximately 677 kW per fuel-channel and the corresponding heat loss of 300 fuel-channels is approximately 203 MW. Thus, the total heat loss per channel is approximately 8% the total power. Table 1 compares the heat loss between the reference REC, the reference High Efficiency Channel (HEC) and the present CANDU-6 fuel channel [11]. The heat loss in the REC is approximately 7 times the heat loss in the HEC channel, when no insulation barrier is included. While a higher heat loss is expected, this does not occur in part as the cold annulus acts as a preheater recovering some of the heat.

Table 1: Heat Loss Comparison

	Total Heat Loss Per Channel (kW)	Total Heat Loss in Reactor (MW)
Re-Entrant Channel without insulation	677	203 MW (300 channels)
Reference HEC SCWR Fuel-Channel Concept	104	31 MW (300 channels)
CANDU-6 Fuel Channel	11.32	4.3 MW (380 channels)

One option to reduce heat loss in the REC is to increase the thickness of the flow and pressure tubes. However, previous work has shown that heat transfer is not significantly affected by the thickness of the inner flow tube [4]. The total heat loss from the cold side of the REC channel to the moderator for varying pressure tube thicknesses is shown in Figure 8. The heat loss does not decrease significantly as the pressure tube thickness is increased. For a pressure-tube thickness of 16 mm, the heat loss is approximately 5 times the heat loss of the reference HEC fuel channel. Thicker pressure tubes are not suitable from a design perspective as it will lead to larger calandria vessels and are not suitable from a neutronics perspective where less thickness is preferred.

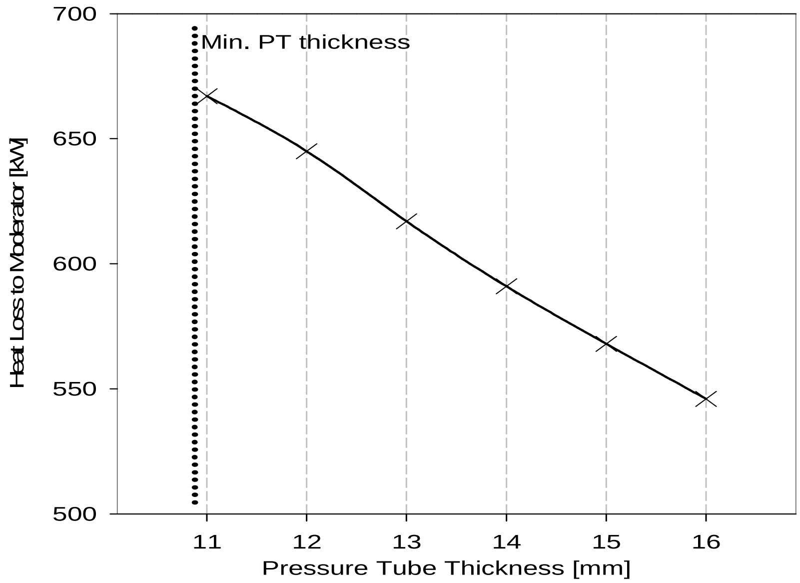

Figure 8: Heat loss to the moderator when pressure tube thickness is varied.

An alternative option to reduce heat loss is to use a different material of construction for the pressure tube. A sensitivity analysis for the material of construction of the pressure tube was performed by varying the reference thermal conductivity. Figure 9 shows the difference in temperature across the pressure tube when thermal conductivity is varied. The total heat loss to the moderator when the thermal conductivity is changed by a factor of 0.5 is 322 kW per channel, which is still 3 times the reference HEC.

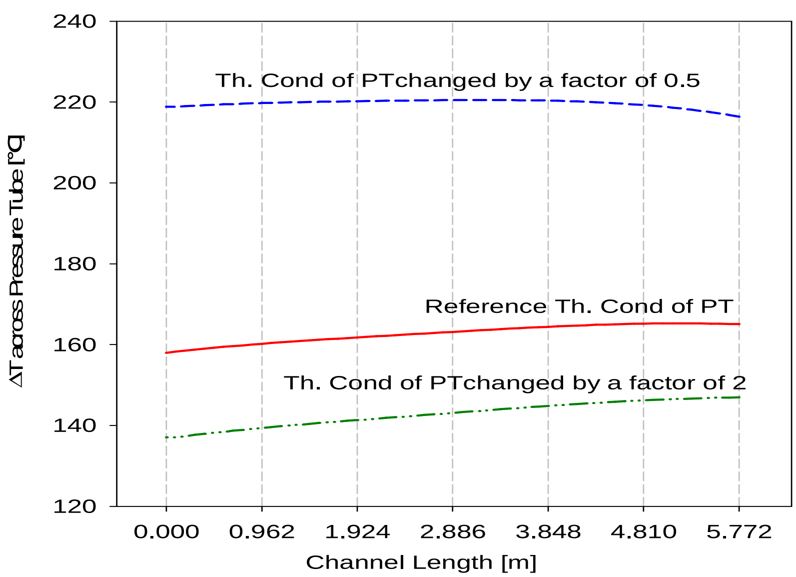

Figure 9: Change in temperature across pressure tube when thermal conductivity is varied.

5. Concluding Remarks

A preliminary heat-transfer analysis was performed for the re-entrant channel for PT-Type SCWRs.

The temperature profiles of the coolant, outer sheath, and inner and outer surfaces of the flow tube and the pressure tube were estimated and the total heat loss to the moderator was calculated. The temperature profiles of the coolant in the hot side and of the outer sheath change considerably within the pseudocritical region. The pseudocritical point for the reference re-entrant channel is within the hot side of the fuel channel, approximately near bundle #5.

The total heat loss to moderator is much higher compared to the reference HEC and current CANDU-6 fuel-channels [11]. Placing the ceramic used in the HEC outside the pressure tube in the REC should considerably reduce heat loss.

6. Acknowledgements

Financial support from the NSERC/NRCan/AECL Generation IV Energy Technologies Program and NSERC Discovery Grants is gratefully acknowledged.

7. Nomenclature

$\bar{c}_p$	average specific heat, J/kg·K
D	diameter, m
G	mass flux, kg/m^{2}s
H	enthalpy, J/kg
HTC	heat transfer coefficient, W/m^{2}K
k	thermal conductivity, W/m·K
L	length, m
$\dot{m}$	mass-flow rate, kg/s
r	radius, m
R	thermal resistance, °C/W
$\dot{Q}$	total heat transfer rate, W
q	heat flux, W/m^2
T	temperature, °C
u	fluid velocity in x-direction, m/s
v	fluid velocity in y-direction, m/s

Greek Letters

α	thermal diffusivity, m^2/s
β	expansion coefficient, 1/K
μ	dynamic viscosity, Pa·s
ν	viscosity, Pa.s
ρ	density, kg/m^3

Acronyms

Subscripts and Superscripts

b	bulk
cs	cold side
free	free convection
hs	hot side
hy	hydraulic
i	inner
mod	moderator
node	node number
o	outer
os	outer sheath
pc	pseudocritical
w	wall
x	axial length along fuel channel

Dimensionless numbers

$\overline{\mathbf{Cp}}$	average specific heat
Nu	Nusselt number ($HTC \cdot D_{hy} / k$)
Pr	Prandtl number ($\mu \cdot c_p / k$)
$\overline{\mathbf{Pr}}$	average Prandtl number
Ra	Rayleigh number
Re	Reynolds number ($G \cdot D_{hy} / \mu$)

CANDU	CANada Deuterium Uranium
FT	Flow Tube
PT	Pressure Tube
HEC	High Efficiency Channel
HTC	Heat Transfer Coefficient
MATLAB	MATrix LABoratory
NIST	National Institute of Standards and Technology
REFPROP	REFerence fluid thermodynamic and transport PROPerties
SCWR	SuperCritical Water Reactor

8. References

[1] Sun, P., Jiang, J., "Thermal-Hydraulic Modeling of CANDU-SCWR and Linear Dynamic Model Development", Proceedings of the International Conference on Nuclear Engineering (ICONE-18), Xi'an, China, 2010, May 17-21.

[2] Pioro, I.L. and Duffey, R.B., "Heat Transfer and Hydraulic Resistance at Supercritical Pressures in Power Engineering Applications", ASME Press, New York, NY, USA, 2007.

[3] Chow, C. K., & Khartabil, H. F., "Conceptual Fuel Channel Designs for CANDU – SCWR", Nuclear Engineering and Technology , 2008, pp. 1−8.

[4] Samuel, J., Harvel, G., & Pioro, I. "Design Concept and Heat Transfer Analysis for a Double Pipe Channel for SCWR Type Reactors", Proceedings of the International Conference on Nuclear Engineering (ICONE-18), Xi'an, China, 2010, May 17-21.

[5] Buongiorno, J., and MacDonald, P. E., "Supercritical water reactor (SCWR). Progress report for the FY-03 Generation-IV R&D activities for the development of the SCWR in the U.S.", Report for INEEL/EXT-03-01210, Idaho, USA, 2003.

[6] Cengel, Y., "Heat and Mass Transfer", McGraw Hill, USA, 2006.

[7] Mokry, S., Gospodinov, Y., Pioro, I.L., "Supercritical Water Heat-Transfer Correlation for Vertical Bare Tubes", Proceedings of the International Conference on Nuclear Engineering (ICONE-17), Brussels, Belgium, 2009, July 12-16.

[8] Sweet, J. N., Roth, E. P., & Moss, M., "Thermal Conductivity of Inconel 718 and 304 Stainless Steel". International Journal of Thermophysics, 1987.

[9] Churchill, S.W., & H. H. S. Chu, "Correlating Equations for Laminar and Turbulent Free Convection from a Horizontal Cylinder", International Journal of Heat Mass Transfer. Vol 18, 1975, p. 1049.

[10] Incropera, F. P., Dewitt, D. P., Bergman, T. L., & Lavine, A. S., "Fundamentals of Heat and Mass Transfer", John Wiley & Sons, USA, 2006.

[11] Peiman, W., Saltanov, E., Gabriel, K., Pioro, I.L., "Heat-Loss Calculations in a SCWR Fuel-Channel", Proceedings of the International Conference on Nuclear Engineering (ICONE-18), Xi'an, China, 2010, May 17-21.