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ANALYSIS OF LOW THERMAL-CONDUCTIVITY FUELS IN A 64-ELEMENT SCWR BUNDLE

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Abstract

The objective of this study is to examine the possibility of using nuclear fuels with low thermal-conductivities (UO_2, ThO_2, and MOX) in order to compare fuel options for use in a SuperCritical Water-cooled Reactor (SCWR). The bundle being analyzed will be a 64-element fuel bundle containing 63 heated elements (9.13-mm OD) and one central unheated element (20-mm OD). When UO_2 is utilized as a fuel in a generic 43-element bundle (11.5-mm OD), the fuel centerline temperature may exceed the industry accepted limit of 1850°C for certain Axial Heat Flux Profiles (AHFPs). The 64-element fuel bundle showed promising results at SCWR conditions with the use of low thermal-conductivity fuels with various AHFPs.

1. Introduction

As part of the Generation-IV International Forum (GIF), the SCWR-concept is under development worldwide and will operate with elevated operating parameters compared to current Nuclear Power Plants (NPPs). For example, the outlet pressure of an SCWR is 25 MPa, compared to that of the current CANDU reactors at approximately 10 MPa, with outlet temperatures of 625°C compared to 310°C, respectively. As a result of such high parameters, SCWRs will use SuperCritical Water (SCW) as a coolant, which is a fluid at pressures and temperatures higher than that of the critical pressure and critical temperature of water (See Figure 1) [1]. Supercritical fluids have unique heat transfer characteristics, such as dramatic changes in the fluid density, specific heat, thermal conductivity, and viscosity, which can be seen in Figure 2 [2,3]. The most significant changes occur within a region of ±25°C from the pseudocritical point (a point at a pressure and temperature above the critical point corresponding to the maximum value of specific heat for a particular pressure [1]) of 384.9°C at 25 MPa. Phenomena such as critical heat flux does not occur because there is no distinctive phase change when crossing from a high-density to a low-density fluid. However, a deteriorated heat-transfer regime may be present at supercritical conditions [1]. The National Standards Institute of Technology (NIST) Reference Fluid Properties (REFPROP) software [4] was used to calculate various thermophysical properties of water. Profiles of selected thermophysical properties of SCW within the pseudocritical range are shown in Figure 2.

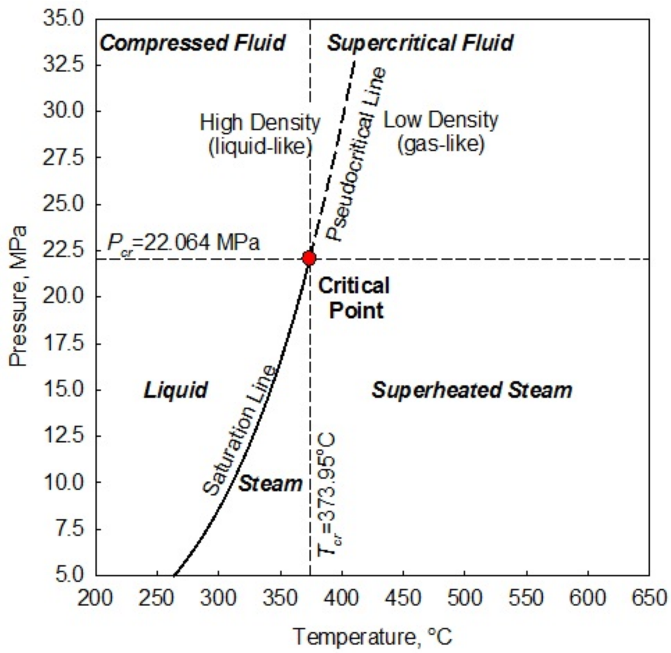


Figure 1: Pressure-Temperature Diagram for Water in the Critical Region [5].

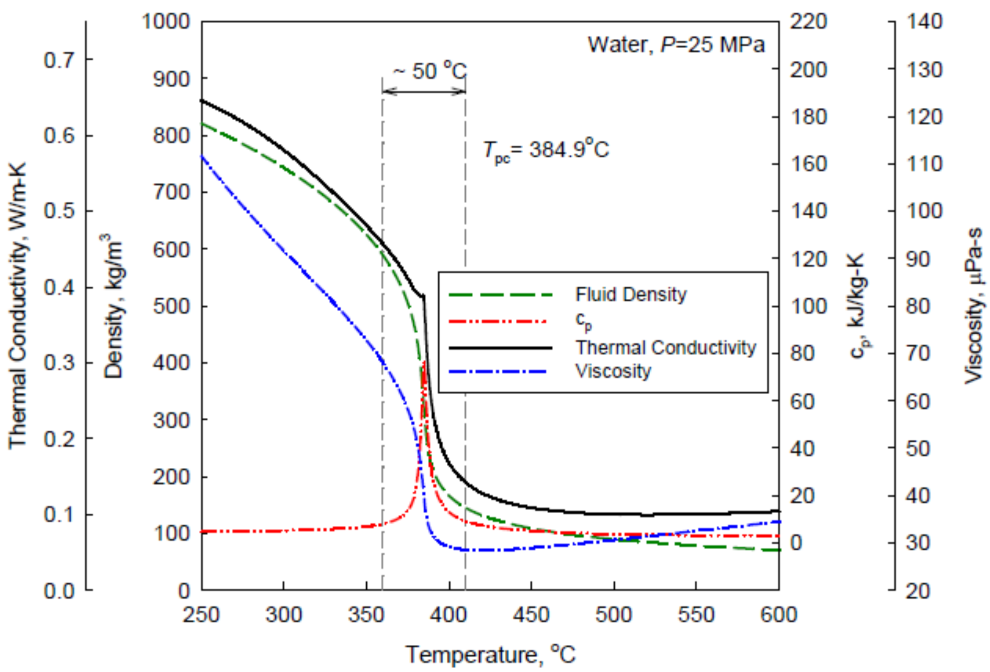


Figure 2: Selected Properties of Supercritical Water at Pseudocritical Point [5].

One of the main objectives for developing and utilizing SCWRs is that SCW NPPs offer an increased thermal efficiency of approximately 10 – 20% to current NPPs [1]. The SCWR concepts include pressure-vessel (PV) and pressure-tube (PT) options, which are currently under development worldwide. Canada is currently working on the development of a PT-reactor concept – SCW CANDU reactor and Russia on the VVER – SCP concept (see Table 1) [1]. The VVER – SCP concept is an advancement of the Pressurized Water Reactor (PWR) – type which operates at supercritical temperatures and pressures.

Table 1: Major Parameters of SCW CANDU (Canada) and SCW VVER-SCP (Russia) Nuclear-Reactor Concepts [1].

Parameters	SCW CANDU	VVER-SCP
Reactor Type	PT	RPV
Reactor Spectrum	Thermal	Fast
Thermal Power, MW	2540	3830
Electric Power, MW	1220	1700
Thermal Efficiency, %	48	44
Pressure, MPa	25	25
Inlet Temperature, °C	350	280
Outlet Temperature, °C	625	530
Flowrate, kg/s	1320	1860
Number of Fuel Channels	300	241
Number of Element per Fuel Bundle	43	252
Length of Fuel Bundle String, m	6	4

The proposed SCW CANDU nuclear reactor is a PT-type core consisting of 300 fuel channels. The fuel-channel design is internally insulated by a ceramic insulator to enable the PT to operate at temperatures close to that of the moderator (~80°C) [1]. The fuel channel also incorporates a perforated metal liner. Within the fuel channel assembly, 12 fuel bundles will reside at about 0.5 m in length each and will be cooled with SCW. Figure 3 is used to illustrate the fuel assembly as described, including a 64-element fuel bundle used for thermalhydraulic analysis. The 64-element bundle design is the most recent bundle development and thus has been chosen for analysis. Previous studies have shown that the Variant-20 fuel bundle design, comprised of 42 fueled heated elements (11.5-mm OD) and one central unheated element (20-mm OD), may exceed the fuel centerline temperature industry accepted limit of 1850°C for the UO_2 fuel. In order to lower the fuel centerline temperature, the following design options may be considered: 1) to use annular fuel elements, 2) to use hollow fuel pellets, 3) to increase the number of fuel elements per fuel bundle, or 4) to use fuel material with a higher thermal conductivity than that of currently used UO_2. The current paper will only consider the third option mentioned.

The objective of this paper is to examine the possibility of using fuels with low thermal conductivities, MOX fuel, UO_2 fuel, and ThO_2 fuel, for application in the most recent PT fuel-bundle design (i.e., the 64-el.), and to determine which options are feasible.

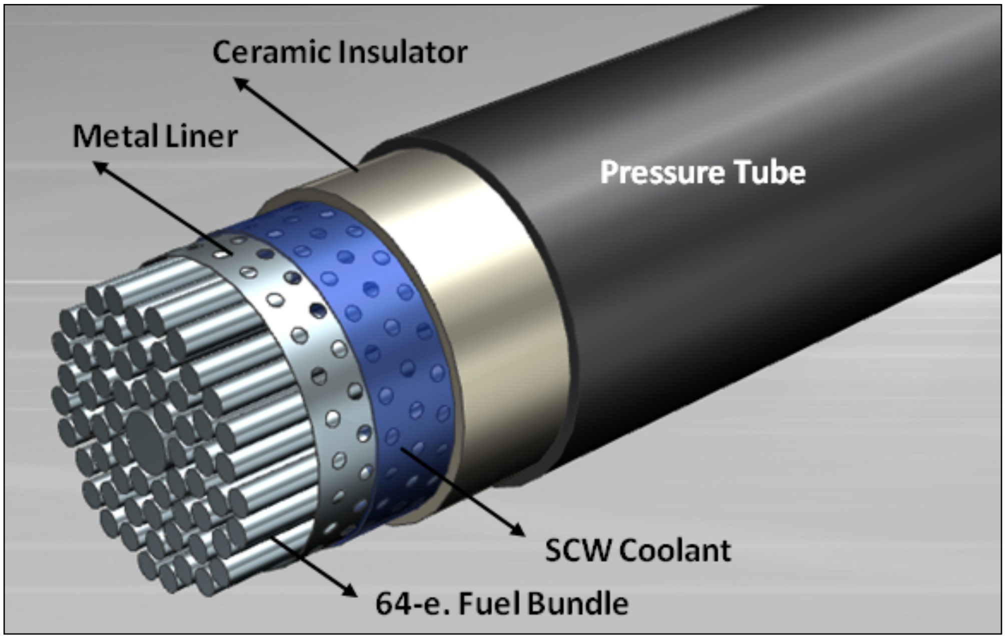


Figure 3: SCWR PT-type Fuel Channel Shown with a 64-Element Fuel Bundle.

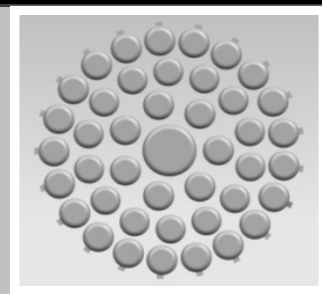
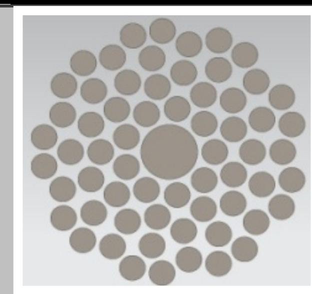
2. Background

SuperCritical Water-cooled Reactors, as mentioned above, are one of the six nuclear-reactor concepts currently being developed under the Generation-IV program. Also under the Generation-IV program are: Gas-cooled Fast Reactors (GFRs), Lead-cooled Fast Reactors (LFRs), Molten Salt-cooled Reactors (MSRs), Sodium-cooled Fast Reactors (SFRs), and Very High-Temperature gas-cooled Reactors (VHTRs). The main advantage of SCWRs is an increase in the thermal efficiency from 30-34% (current level of NPPs) to 45-50% for SCW NPPs. In general, several fuel-bundle designs can be considered for Pressure-Channel (PCh) SCWRs. Some of these bundles are: the 37-element, 43-element CANFLEX, 43-element Variant-18, 43-element Variant-20, and a 64-element design based off of the Variant designs. The fuel bundle being analysed throughout this paper will be the 64-element fuel bundle, described in detail in the section to follow.

2.1 Fuel Bundle Used for Analysis

The chosen fuel bundle for analysis is the 64-element design, based off of the 43-element Variant-20 design. It is comprised of 63 heated fuelled elements, and the central element is assumed to be unheated with a 20-mm outer diameter (OD). The 63 heated elements have a diameter of 9.13-mm, and the bundle has a hydraulic-equivalent diameter of 7.24-mm. Table 2 compares the previously analysed Variant-20 fuel bundle design to that of the 64-element design.

Table 2: SCWR-Type Fuel Bundles.

Fuel Bundle	Variant-20	64-Element
Cross-Section		
Total Number of Elements	43	64
Central Element Diameter, mm	20	20
Outer Element Diameter, mm	11.5	9.13
Fuel Bundle Diameter, mm	103.45	103.45
Fuel Bundle Length, mm	500	500
Hydraulic-Equivalent Diameter, mm	7.83	7.24
Mass Flux, kg/m^{2}s	1772	1101
Heat Flux, kW/m^2	970	815
Mass Flow Rate, kg/s	4.4	4.4
Average Channel Power, MW	8.5	8.5

2.1.1 Burnable Neutron Absorber

The 64-element fuel bundle is proposed to have a central unheated element which will contain a type of neutron absorbing material in the form of burnable poison. All of the fuel bundle designs mentioned within this paper that incorporate a central unheated element, are proposed to have a burnable neutron absorber (poison) within the central element. The central unheated element, with a larger diameter than the outer elements, is proposed to be filled with a neutronic poison in order to manage the flux values within the bundle; in attempts to lower maximum power per bundle. Having this feature within each fuel bundle will also reduce the void reactivity [6].

3. Nuclear Fuels Analysed

The nuclear fuels used for analysis within this paper are the Mixed Oxide (MOX) fuel, Uranium Dioxide (UO_2) fuel and Thoria or Thorium Dioxide (ThO_2) fuel. All three fuels have low thermal conductivities which decrease with increasing temperature as shown in Figure 4.

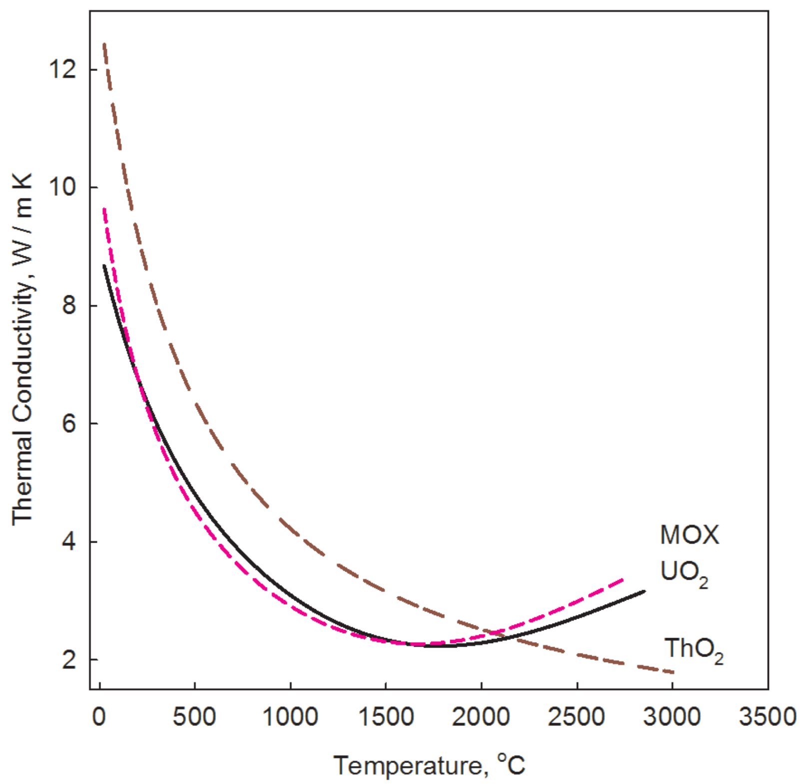


Figure 4: Thermal Conductivities of Selected Nuclear Fuels [7].

With respect to the most commonly used UO_2 fuel, MOX and ThO_2 fuels have similar properties, as shown in Table 3. All three fuels have melting temperatures above 2700°C and thermal conductivities below 12 W/m·K. Specific properties related to each of the fuels are shown in Table 3. The higher melting point of ThO_2 gives reason to believe that an increased safety margin between the normal operating point and fuel melting point, will act as an added safety feature. The increasing thermal conductivity from MOX fuel, to UO_2 fuel, and then to ThO_2 fuel also gives reason to believe that the use of ThO_2 fuel will result in lower fuel centerline temperatures.

Table 3: Basic Properties of Selected Nuclear Fuels at 0.1 MPa and 298K [7].

Property	Unit	MOX[1]	UO_2	ThO_2
Melting Point	°C	2750	2847±30	3227±150
Thermal Conductivity	W/m·K	7.82[2]	8.68	9.7
Theoretical Density	kg/m^3	11,074	10,960	10,000
Heat Capacity	J/kg·K	240	235	235

Heat of Vaporization	kJ/kg	1498	1530	–
Linear Expansion Coefficient	1/K	9.43E-6	9.75E-6	8.9E-6[3]

[1]MOX – ($U_{0.8}$Pu0.2)O_2, where 0.8 and 0.2 are the molar parts of UO_2 and PuO_2.
[2]at 95% density.
[3]at 1000°C [8].

The thermal conductivity for each fuel is required for the analysis within this paper. As illustrated in Figure 4, the thermal conductivities of each fuel are a function of temperature and therefore are determined using the following equations. For 95% Theoretical Density (TD) MOX fuel, Eq. (1) was used, and is valid for temperatures within the range of 700 – 3100 K [9].

$$k_{MOX}(T,x)=\frac{1}{A(x)+C(x)\,10^{-3}T}+\frac{6400}{(10^{-3}T)^{5/2}}\exp\left(\frac{-16.35}{10^{-3}T}\right) \quad (1)$$

Where *T* is temperature, and *x* is a function of oxygen to heavy metal ratio ($x=2-O/M$) and

$$A(x)=2.58x+0.035 \quad (2)$$

$$C(x)=-0.715x+0.286 \quad (3)$$

A(x) has the units of m·K/W, and *C(x)* has the units of m/W.

The thermal conductivity of 95% TD UO_2 can be calculated using the Frank correlation, Eq. (4), which is valid for temperatures within the range of 298 – 3120 K [9].

$$k_{UO_2}(T)=\frac{100}{7.5408+17.692\times10^{-3}T+3.6142\times(10^{-3}T)^2}+\frac{6400}{(10^{-3}T)^{5/2}}\exp\left(\frac{-16.35}{10^{-3}T}\right) \quad (4)$$

The thermal conductivity for Thoria fuel is calculated with the following equation [10]:

$$k_{ThO_2}=\frac{1}{0.0327+(1.603\cdot10^{-4}T)} \quad (5)$$

3.1 Neutronic Considerations

There are neutronic challenges that must be discussed when an alternative fuel is under consideration for application in a NPP. Due to Thorium being a fertile material, a fissile material must be added into the core so that a sufficient quantity of neutrons can be supplied to start the nuclear chain reaction [11]. For this paper, the heat-transfer analysis only pertains to one fuel channel containing UO_2, ThO_2, or MOX fuel bundles. In addition, this study does not propose that the core entirely or partially consists of any of the above mentioned fuels. The use of any of these fuels or a combination of them requires further research and reactor physics analysis.

4. Heat-Transfer Analysis

One-dimensional, steady-state heat transfer calculations where conducted to calculate the fuel centerline temperature of one fuel element of the proposed 64-element fuel bundle. The Heat Transfer Coefficients (HTCs) along the heated length of the fuel channel were then calculated using the Mokry et al. correlation, as seen in Eq. (6). The values of HTC were then used to determine the temperature profile for the outer sheath of the fuel elements. It was assumed that there was no gap between the sheathing and the fuel pellets; therefore, the inner-sheath temperature is equal to the outer surface temperature of the fuel pellet. The profile of the fuel centerline temperature was then calculated. Throughout all calculations, the heated length of an SCWR was assumed to be 5.772-m, the mass flow rate of the coolant was 4.4 kg/s, and the thermal power per channel was 8.5 MW. It was also assumed that all the fuel elements have the same heat flux.

The Mokry et al. correlation was used to calculate the HTC values because of results from a recent study performed at the University of Ottawa by Zahlan et al. [12]. The study has proven that the Mokry et al. correlation results in the best agreement with experimental data within the supercritical region. The Mokry et al. correlation is based on average, wall- and bulk-fluid properties and is represented by the equation below;

$$Nu_b = 0.0061\,Re_b^{0.904}\,\overline{Pr}_b^{0.684}\left(\frac{\rho_w}{\rho_b}\right)^{0.564} \qquad (6)$$

4.1 Axial Heat Flux Profiles

There are many possible Axial Heat Flux Profiles within a reactor core. In an ideal situation, the heat flux within a reactor core would be distributed evenly in a symmetrical cosine shape; however in reality due to normal operating conditions, including control rod and adjuster rod manipulations, fuel burn-up, and on-line refueling, this is not the case. More specifically, the heat flux in each of the individual fuel channels is different and varies from one to another. In general, these AHFPs can be enveloped with uniform and non-uniform cosine-type AHFPs. Further, the latter category includes: 1) cosine, 2) upstream-skewed cosine, and 3) downstream-skewed cosine. The uniform AHFP is representative of an ideal axial heat-flux distribution within the core (heat flux maintained at average value). The cosine AHFP is representative of a normal axial heat-flux distribution within the core. The upstream- and downstream-skewed cosine AHFPs are representative of biased refueling schemes. The power ratios used to determine the non-uniform cosine AHFPs are shown in Figure 5.

Steady-state conditions were assumed throughout the calculations presented in this paper. The mass-flow rate was assumed to be constant at 4.4 kg/s per channel with an average channel power of 8.5 MW_{th}.

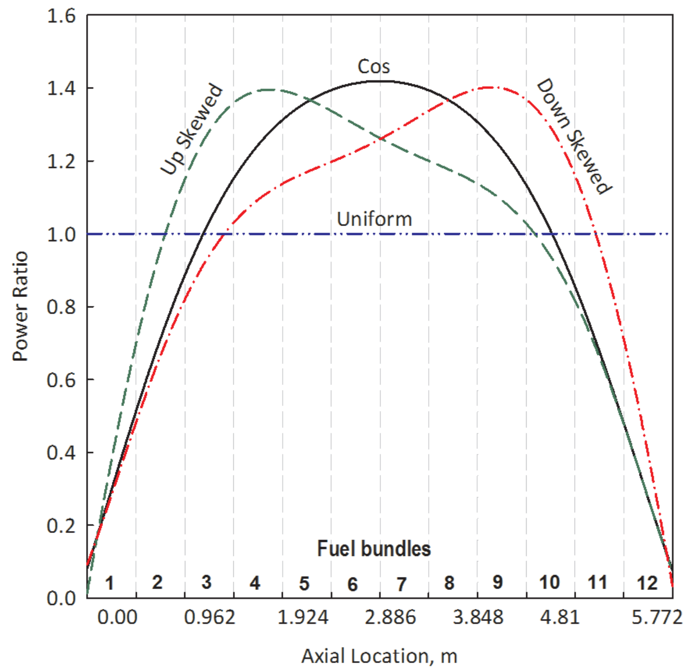


Figure 5: Power Ratios Used to Determine AHFPs at Average Channel Power of 8.5 MW_{th} (Based on [13]).

4.2 Bulk-Fluid Temperature Profile

As mentioned, the bulk-fluid temperatures were found using the heat balance method. The relation is listed below, where *h* is the enthalpy along the heated length of the channel, *x*;

$$h_{\square} = \square\cdot\square \quad \square\cdot\Delta\square \quad + h_{\square-1}$$

(7)

4.3 Outer-Sheath Temperature Profile

The outer-sheath temperature profile is calculated using Eq. (8), while the HTC values are calculated via Eq. (6).

$$\square\square,\square h = \square\square\square\square + \square\square$$

(8)

4.4 Inner-Sheath Temperature Profile

Once the outer-sheath temperatures are calculated, the inner-sheath temperatures can be obtained using Eq. (9) [14]. The design accepted limit for the sheath temperature is 850°C.

□ □,□h,□= □□h,□□□□□□□2·□·□□h,□+□□,□h

(9)

Inconel-600 has been selected as the material of choice for the sheath, and its thermal conductivity, *k*, is calculated using Eq. (10) [15], where *T* is the temperature in Kelvin.

$$k=8.116+0.0176\cdot T \quad (10)$$

4.5 Fuel Centerline Temperature Profile

The fuel centerline temperature was determined with Eq. (11), where the thermal conductivity *k* is of the fuel and determined with Eq. (1), (4) or (5).

□□□= □□,□h+□□□□·(□□+□□+□)24□□□□□

(11)

The fuel centerline temperatures were calculated iteratively using 50 increments along the radius of the fuel pellet, while the thermal conductivity was assumed to change as a function of temperature. The industry accepted limit for the fuel centerline temperature is 1850°C.

5. Results

As expected, the thermal conductivities of each of the nuclear fuels analysed affected the fuel centreline temperature profiles along the heated length of a fuel channel. The MOX fuel resulted in higher fuel centreline temperatures compared to the UO_2 and ThO_2 fuels in all four AHFPs. Whereas the ThO_2 fuel, which has the highest thermal conductivity of all three fuel materials, resulted in a lower fuel centreline temperature profile. The following four figures show the MOX fuel for a 64-element fuel bundle within the uniform, cosine, upstream-skewed, and downstream-skewed AHFPs.

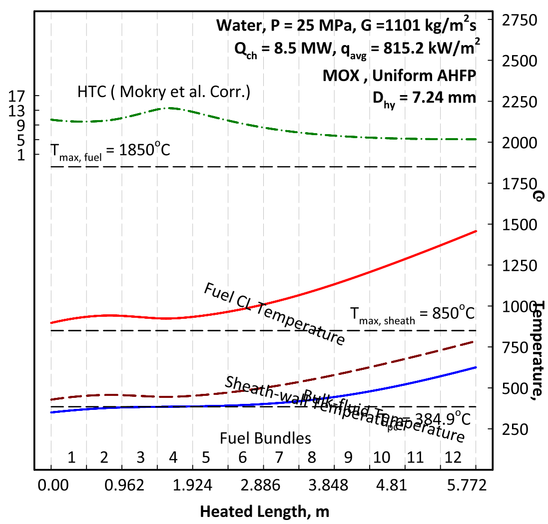


Figure 6: Uniform AHFP: 64-el. with MOX Fuel.

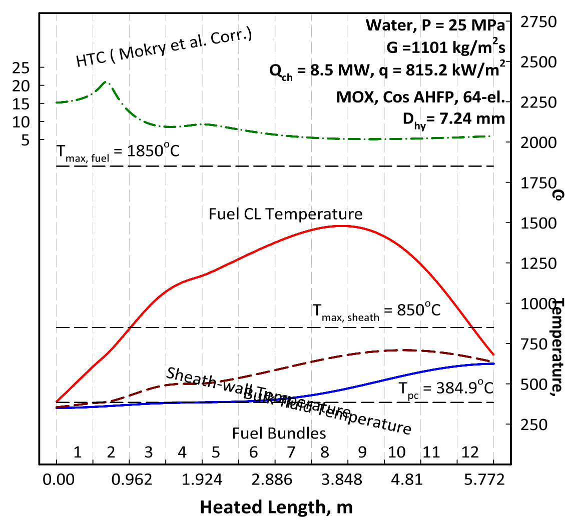


Figure 7: Cosine AHFP: 64-el. with MOX Fuel.

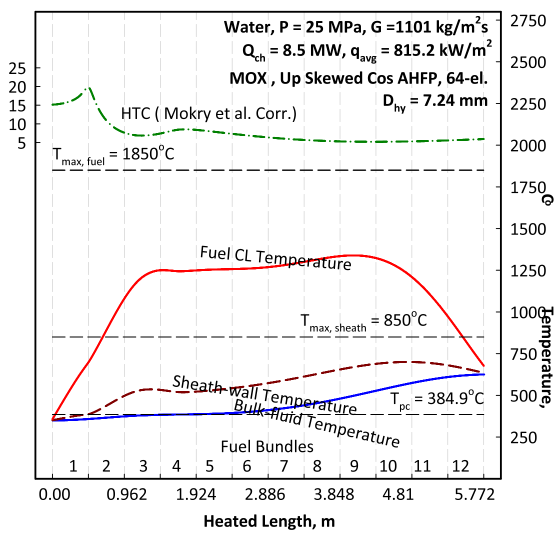


Figure 8: Upstream-Skewed AHFP: 64-el. with MOX Fuel.

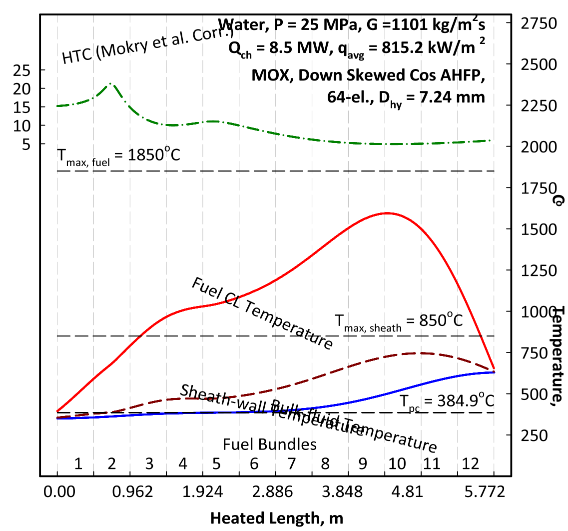
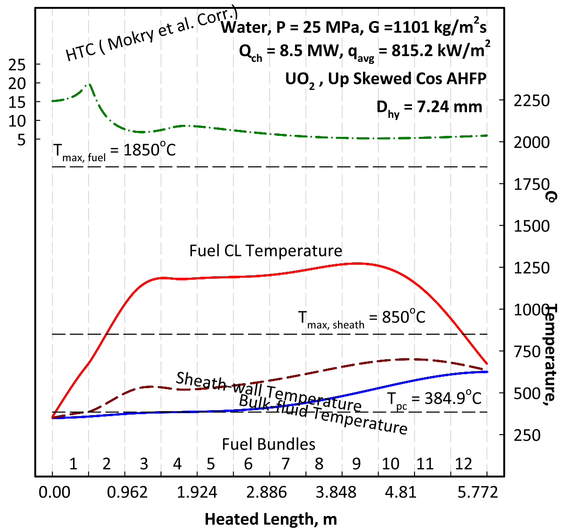


Figure 9: Downstream-Skewed AHFP: 64-el. with MOX Fuel.

All three fuels tested, resulted in similar trends within each AHFP analysed. When comparing Figures 6 – 9, the downstream-skewed AHFP resulted in the highest temperature values along the heated length of the fuel channel, while the upstream-skewed AHFP resulted in the lowest maximum temperature values. In order to take a conservative approach, it is then wise to utilize the downstream-skewed AHFP to make concluding remarks. The downstream-skewed AHFP for the UO_2 and ThO_2 fuel are also shown below (see Figures 11 and 13). To compare maximum and minim calculated temperatures, the upstream-skewed AHFPs are also shown (see Figures 10 and 12).



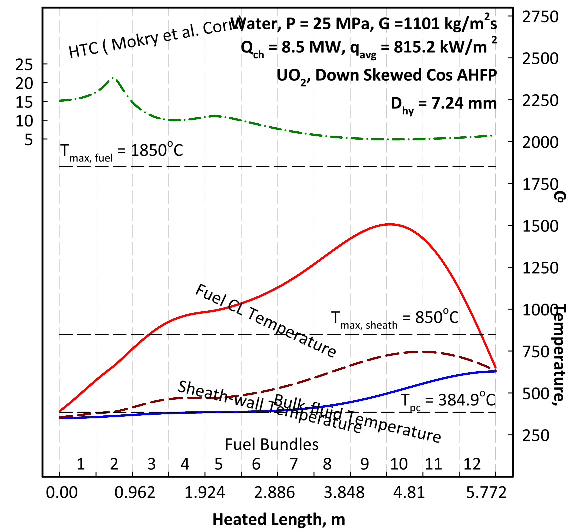
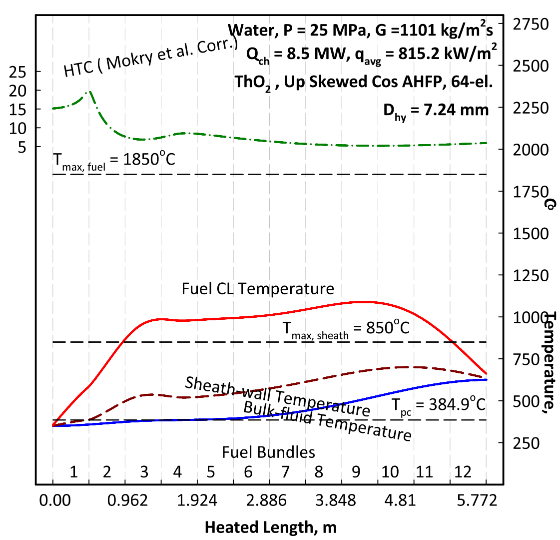


Figure 10: Upstream-Skewed AHFP: 64-el. with UO_2 Fuel.

Figure 11: Downstream-Skewed AHFP: 64-el. with UO_2 Fuel.



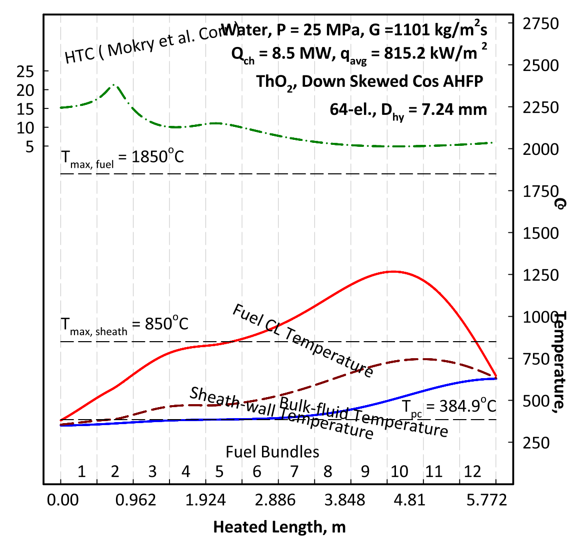


Figure 12: Upstream-Skewed AHFP: 64-el. with ThO_2 Fuel.

Figure 13: Downstream-Skewed AHFP: 64-el. with ThO_2 Fuel.

It is also important to note, that within each of the tested AHFPs, no matter the fuel tested, neither the fuel centreline industry accepted limit of 1850°C nor the design accepted limit of 850°C for the sheath were exceeded. Although the industry accepted limit may differ for the use of MOX or ThO_2 fuel as it is based on the UO_2 fuel, the change in value is expected to be minimal. The fuel centreline melting temperature limit is determined according to the fuel materials melting temperature, and according to the data in Table 3, the MOX fuel possesses the lowest melting temperature of 2750°C. With a reduced fuel melting temperature of 100°C compared to UO_2 fuel, it can be expected that the fuel centreline temperature limit may be decreased by 100°C, i.e., 1750°C. Although the limit is decreased, the MOX fuel still remains below an expected limit.

6. Conclusion

Low thermal conductivity fuels were tested within a 64-element fuel bundle designed for application in an SCWR. Four AHFPs were investigated; uniform, cosine, upstream-skewed, and downstream-skewed. MOX, UO_2, and ThO_2 fuels showed similar trends within the AHFPs. The MOX fuel, possessing the lowest thermal conductivity of the three fuels tested, resulted in the highest fuel centreline temperature values along the heated length of the fuel channel, whereas the ThO_2 fuel, possessing the highest thermal conductivity of the three fuels, resulted in the lowest fuel centreline temperature values. The UO_2 fuel, possessing a thermal conductivity between those of the MOX and ThO_2 fuels, thus resulted in fuel centreline temperatures between the MOX and ThO_2 fuels along the heated length of the fuel channel.

The most extreme condition analysed within this paper, the downstream-skewed AHFP, is used primarily to make a conservative conclusion. The MOX, UO_2 and ThO_2 fuel reached a maximum fuel centreline temperature of about 1600°C, 1500°C and 1270°C, respectively, within the downstream-skewed AHFP. The increased safety margin of 330°C achieved with the use of ThO_2 fuel over the use of the MOX fuel illustrates the effect of material thermal conductivity, and shows that the ThO_2 fuel is a better option compared to the MOX fuel, with respect to the thermal analysis presented within this paper. As well, in all cases analysed, the design temperature limit for the sheath material of 850°C was not exceeded. Although all three fuels analysed remained below the industry accepted fuel centreline temperature of 1850°C and are thus acceptable for use in SCWRs, the ThO_2 nuclear fuel is a better option due to the added safety margin. However fuels that possess higher thermal conductivities, such as Uranium mononitride (UN) or Uranium carbides (UC, UC_2) should be tested for further comparison.

7. Acknowledgements

Financial supports from the NSERC Discovery Grant and NSERC/NRCan/AECL Generation IV Energy Technologies Program are gratefully acknowledged.

8. Nomenclature

A area, m^2
c_p specific heat at constant pressure, J/kg K
$\bar{c}_p$ averaged specific heat within the range of $(T_w - T_b)$; $\left(\frac{H_w - H_b}{T_w - T_b}\right)$, J/kg K
D diameter, m
□□□□ heat generation, W/m^3
G mass flux, kg/m^2s;
H, h enthalpy, J/kg

k	thermal conductivity, W/m·K
L	length, m
⍰	mass-flow rate, kg/s
P	pressure, MPa
p	perimeter, m
Q	thermal power, W
q	heat flux, W/m^2
r	radius, m
T, t	temperature, ºC
x	axial coordinate, m

Greek Letters

π	pi
μ	dynamic viscosity, Pa·s
ρ	density, kg/m^3

Non-dimensional Numbers

Nu	Nusselt number; $\left(\frac{HTC\,D}{k}\right)$
Pr	Prandtl number; $\left(\frac{\mu\, c_p}{k}\right)$
$\overline{\mathbf{Pr}}$	averaged Prandtl number within the range of $(t_w - t_b)$; $\left(\frac{\mu\ \bar{c}_p}{k}\right)$
Re	Reynolds number; $\left(\frac{G\,D}{\mu}\right)$

Subscripts or Superscripts

avg	average
b	bulk-fluid
CL	fuel centerline
ch	channel
cr	critical
hy	hydraulic-equivalent
i	inner
j	radial coordinate, m
max	maximum
o	outer
pc	pseudocritical
sh	sheath
th	thermal
w	wall

x axial location along heated length

Abbreviations and Acronyms

AHFP	Axial Heat Flux Profile
CANDU	CANada Deuterium Uranium (reactor)
GFR	Gas-cooled Fast Reactor
GIF	Generation-IV International Forum
HTC	Heat Transfer Coefficient
ID	Inner Diameter
LFR	Lead-cooled Fast Reactor
MOX	Mixed Oxide (nuclear fuel)
MSR	Molten Salt-cooled Reactor
NIST	National Institute of Standards and Technology
NPP	Nuclear Power Plant
OD	Outer Diamter
PCh	Pressure-Challed
PT	Pressure Tube (reactor)
PV	Pressure Vessel (reactor)
PWR	Pressurrized Water Reactor
REFPROP	REFerence Fluid PROPerties
SCW	SuperCritical Water
SCWR	SuperCritical Water-cooled Reactor
SFR	Sodium-cooled Fast Reactor
TD	Theoretical Density
ThO_2	Thorium Dioxide (nuclear fuel)
UC	Uranium Carbide (nuclear fuel)
UC_2	Uranium Dicarbide (nuclear fuel)
UN	Uranium Mononitride (nuclear fuel)
UO_2	Uranium Dioxide (nuclear fuel)
VHTR	Very High-Temperature gas-cooled Reactor
VVER-SCP	Water Water Power Reactor SuperCritical Pressure (in Russian abbreviations)

9. References

[1] I.L. Pioro and R.B. Duffey (2007). “Heat Transfer and Hydraulic Resistance at Supercritical Pressures in Power Engineering Applications”, ASME Press, New York, NY, USA, 334 pages.

[2] I.L Pioro, H.F. Khartabil and R.B. Duffey, “Heat Transfer to Supercritical Fluids Flowing in Channels – Empirical Correlations (Survey)”, Nuclear Engineering and Design, Vol. 230, No. 1–3, 2004. pp.

[3] I.L Pioro and R.B. Duffey, "Literature Survey of the Heat Transfer and Hydraulic Resistance of Water, Carbon Dioxide, Helium and Other Fluids at Supercritical and Near-Critical Pressures", Report AECL-12137/FFC-FCT-409, CRL AECL, 2003 April, ISSN 0067-0367, 182 pages.

[4] National Institute of Standards and Technology, 2007. NIST Reference Fluid Thermodynamic and Transport Properties-REFPROP. NIST Standard Reference Database 23, Ver. 8.0. Boulder, CO, U.S.: Department of Commerce.

[5] Mokry, S., Gospodinov, Ye., Pioro, I. and Kirillov, P., 2009a. "Supercritical Water Heat-Transfer Correlation for Vertical Bare Tubes", Proceedings of the 17th International Conference on Nuclear Engineering (ICONE-17), Brussels, Belgium, July 12-16, Paper#76010, 8 pages.

[6] Boczar et al. "Thorium fuel-cycle studies for CANDU reactors", Proceedings of three IAEA meetings held in Vienna in 1997, 1998, and 1999, Vienna, Austria, IAEA-TECDOC-1319, pp. 25-41.

[7] Kirillov, P.L., Terent'eva, M.I. and Deniskina, N.B., 2007. Thermophysical Properties of Nuclear Engineering Materials (In Russian, 2nd edition revised and augmented, IzdAT Publ. House, Moscow, Russia, pp. 194.

[8] Bowman, A. L., Arnold, G. P., Witteman, W. G., and Wallace, T. C., (1966). "Thermal Expansion of Uranium Dicarbide and Uranium Sesquicarbide". *Journal of Nuclear Materials, Vol. 19*, 111-112.

[9] Carbajo, J. J., Yoder, G. L., Popov, S. G., and Ivanov, V. K., (2001). "A Review of the Thermophysical Properties of MOX and UO_2 Fuels". *Journal of Nuclear Materials, Vol. 299*, 181-198.

[10] Jain, D., Pillai, C., Rao, B. K., & Sahoo, K., 2006. "Thermal diffusivity and thermal conductivity of thoria-lanthana solid solutions up to 10 mol.%LaO(1.5)". *Journal of Nuclear Materials, Vol. 353*, pp. 35-41.

[11] Cochran, R. G., and Tsoulfanidis, N., (1999). "The Nuclear Fuel Cycle: Analysis and Management". Second ed. American Nuclear Society, Illinois, USA.

[12] Zahlan, H., Groeneveld, D.C., Tavoularis, S., "Look-up table for trans-critical heat transfer", the 2nd Canada-China Joint Workshop on Supercritical Water-Cooled Reactors (CCSC-2010), Toronto, Canada, April 25-28, 2010, 18 pages

[13] Leung, L., 2008. "Effect of CANDU bundle-geometry variation on dryout power". Proceedings of the 16th International Conference on Nuclear Engineering (ICONE-16), Orlando, Florida, USA, Paper #48827, pp. 8.

[14] Cengel, Y.A., 2007. "Heat and Mass Transfer: A Practical Approach", 3rd Ed., McGraw-Hill Companies, Inc., New York, NY, USA, pp. 901.

[15] IAEA, 2006, Thermophysical properties database of materials for light water reactors and heavy water reactors. Vienna, Austria.