On the Potential Use of High Thermal-Conductivity Fuels in SuperCritical Water-cooled Reactors (SCWRs)

W. Peiman, I. Pioro, K. Gabriel[1], K., and M. Hosseiny
University of Ontario Institute of Technology
Faculty of Energy Systems and Nuclear Science
[1]Faculty of Engineering and Applied Science
2000 Simcoe Street North, Oshawa, Ontario, L1H 7K4, Canada
wargha.peiman@uoit.ca, igor.pioro@uoit.ca
kamiel.gabriel@uoit.ca, mohammad.hosseiny@uoit.ca

Abstract

Chosen as one of six Generation IV nuclear-reactor concepts, SuperCritical Water-cooled Reactors (SCWRs) will have thermal efficiencies within the range of 45-50% owing to high reactor-outlet temperatures. A generic SCWR operates at a pressure of 25 MPa with inlet- and outlet-coolant temperatures of 350°C and 625°C. On the other hand, the high operating temperatures of SCWRs leads to high fuel centerline temperatures. Previous studies have shown that the fuel centerline temperature could exceed the industry limit of 1850°C when UO_2 or other low thermal-conductivity fuels such as ThO_2 or MOX are used in certain conditions. Therefore, there is a need for alternative fuels for future use in SCWRs.
The objective of this paper is to evaluate several high thermal-conductivity fuels, namely, uranium carbide, uranium nitride, and uranium dioxide composed of graphite fibers. Consequently, the fuel centerline and sheath temperature profiles of these fuels have been calculated and compared against the industry accepted limits of 1850°C and 850°C for the fuel and the sheath, respectively. Further, other factors such as thermal conductivity value and trend, melting point, uranium atom density, high temperature stability, chemical compatibility, thermal-shock resistance, steady-state and irradiation-induced creep, and volumetric swelling were considered to determine the most suitable fuel option for future use in SCWRs. The result showed that uranium carbide can be considered as a promising fuel option for SCWRs.

1. Introduction

Currently, the Generation IV International Forum (GIF) has focused on the development of six nuclear reactor concepts, which pave the road to clean and sustainable energy production. These six nuclear reactor concepts are Gas-cooled Fast Reactor (GFR), Sodium-cooled Fast Reactor (SFR), Lead-cooled Fast Reactor (LFR), Very-High-Temperature Reactor (VHTR), SuperCritical-Water-cooled Reactor (SCWR), and Molten Salt Reactor (MSR) [1]. One common feature of these reactors is that they operate at higher temperatures between 510°C and 1000°C, compared with those of the current water-cooled reactors (e.g., less than 330°C) [1]. The high operating temperatures not only increase the thermal efficiency of the Generation IV Nuclear Power Plants (NPPs), but also it allows for the co-generation of hydrogen.

Chosen as one of six Generation-IV nuclear-reactor concepts, SCWRs will have high thermal efficiencies within the range of 48 - 52% owing to high reactor-outlet temperatures. A generic SCWR operates at a pressure of 25 MPa with inlet- and outlet-coolant temperatures of 350°C and 625°C [2].

The high outlet temperature and pressure make it possible to use supercritical "steam" turbines, which lead to high thermal efficiencies at coal-fired power plants. Additionally, there is a possibility for co-generation of hydrogen using high-temperature heat from an SCWR during off-peak hours. For instance, hydrogen production using copper-chlorine cycle requires steam at temperatures as high as 530°C [3, 4], which can be supplied from an SCW nuclear power plant through an intermediate heat exchanger(s).

High operating temperatures of SCWRs leads to high fuel centerline temperatures. Previous studies [5-7] have shown that the fuel centerline temperatures could exceed the industry limit of 1850°C when UO_2 is used at SCWR conditions. Therefore, there is a need for alternative fuels for future use in SCWRs. The objective of this paper is to investigate a possibility of using high thermal-conductivity fuels such as Uranium Carbide (UC), Uranium Nitride (UN), and uranium dioxide composed of graphite fibers (UO_2-C) in SCWRs. The fuel centerline temperature has been calculated for a pressure channel reactor in which the core is composed of distributed fuel channels. Each fuel channel consists of a pressure tube as the pressure boundary, fuel bundles, and other components depending on the fuel-channel design. In the present paper, we have tended to use a conservative analysis approach, which is based on the fuel channels with the maximum thermal power, i.e., +15% above the average channel power, instead of using an average thermal power per channel. Additionally, other factors such as volumetric swelling, chemical stability, thermal conductivity, and melting point of the fuel have been considered in order to determine the best fuel options for SCWRs.

1.1 Parameters of a Generic PT SCWR

In terms of the pressure boundary, SCWRs are classified into two categories: 1) Pressure Tube (PT) or Pressure Channel (PCh) SCWRs and 2) Pressure Vessel (PV) SCWRs. The core of a generic 1200-MW_{el} PT SCWR with single-reheat cycle consists of 300 fuel channels, which are located inside a cylindrical vessel, which is called the Calandria vessel. As shown in Fig. 1, there are 220 SuperCritical Water (SCW) fuel channels with an average thermal power of 8.5 MW, and 80 Steam Re-Heat (SRH) fuel channels with an average thermal power of 5.5 MW. The inlet and outlet temperatures of the coolant (e.g. supercritical water) in SCW channels are 350°C and 625°C at a pressure of approximately 25 MPa. The inlet temperature of the coolant (e.g. superheated steam) in the SRH fuel channels is 400°C, and reaches an outlet temperature of 625°C at an operating pressure of 4.7 MPa. There are 12 fuel bundles located in each fuel channel. In this paper, the latest fuel bundle design, which has been developed by AECL, has been chosen for the purpose of calculating the fuel centreline and sheath temperatures. This fuel bundle, which is called Variant-20 [10], has 42 fuel elements with an outer diameter of 11.5 mm and a central element, which contains burnable neutron-absorbing material and has an outer diameter of 20 mm. Table 1 lists the operating parameters of a generic PT SCWR [8].

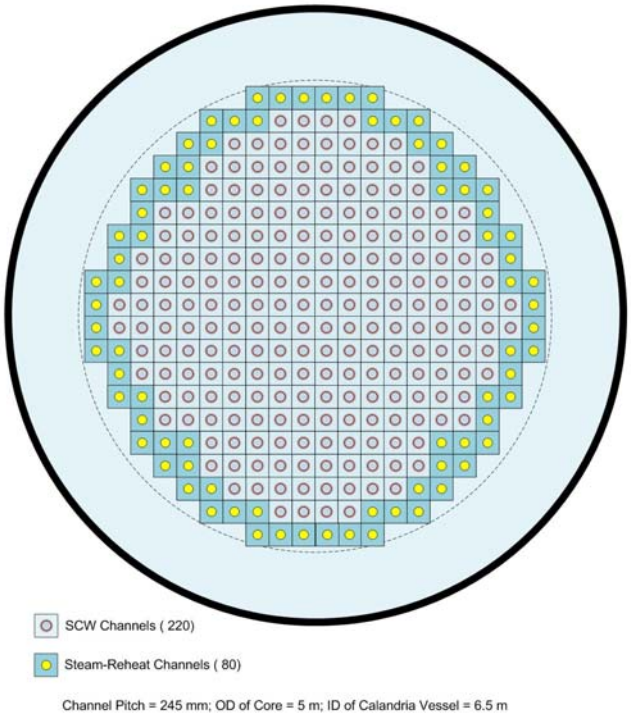


Figure 1: Core Configuration of a Generic 1200-MW_{el} PT SCWR.

In a single-reheat cycle, the supercritical "steam", coming out of SCW fuel channels, expands through a high-pressure turbine. Then, the steam is sent back to the SRH channels, where the temperature of the steam raised to superheated conditions. Next, the superheated steam expands through the intermediate-pressure turbine. Finally, the steam is transferred to the low-pressure turbines, where the steam is exhausted to the condenser [9].

Table 1: Operating Parameters of Generic PT SCWR [8].

Parameters	**Unit**	**Generic PT SCWR**	
Electric Power	MW	1220	
Thermal Power	MW	2540	
Thermal Efficiency	%	52%	
Coolant	-	H_2O	
Moderator	-	D2O	
Pressure of SCW at Inlet/Outlet	MPa	25.8	25
Pressure of SHS at Inlet/Outlet	MPa	6.1	5.7
T_{in} / T_{out} Coolant (supercritical water)	°C	350	625
T_{in} / T_{out} Coolant (superheated steam)	°C	400	625
Mass Flow Rate per SCW/SRH Channel	kg/s	4.4	9.8
Thermal Power per SCW/SRH Channel	MW	8.5	5.5
# of SCW/SRH Channels	-	220	80
Heat Flux in SCW/SRH Channel	kW/m^2	970	628
Fuel Bundle	-	Variant-20[10]	

1.2 Fuel Channel Description and Parameters

The Atomic Energy of Canada Limited (AECL) has developed several fuel channel designs for a proposed SCWR. These fuel channel designs can be classified into two categories: 1) direct-flow and 2) re-entrant channel concepts. From each category, one fuel channel design is presented in this paper.

1.2.1 Re-Entrant Fuel Channels

The presented re-entrant fuel channel design consists of a fuel bundle string, a flow tube, and a Pressure Tube (PT). The outer surface of PT is in contact with the moderator. There is a gap between the flow tube and PT. The coolant flows along the gap between the flow tube and PT. When reaching the end of the fuel channel, the coolant flows inside the flow tube where the fuel bundle is placed. The flow of the coolant through the flow tube removes the heat from the fuel bundle, which in turn increases the temperature of the coolant to a desired outlet temperature. Figure 2 shows a 3-D view of the fuel channel and the flow directions of the coolant along the fuel channel.

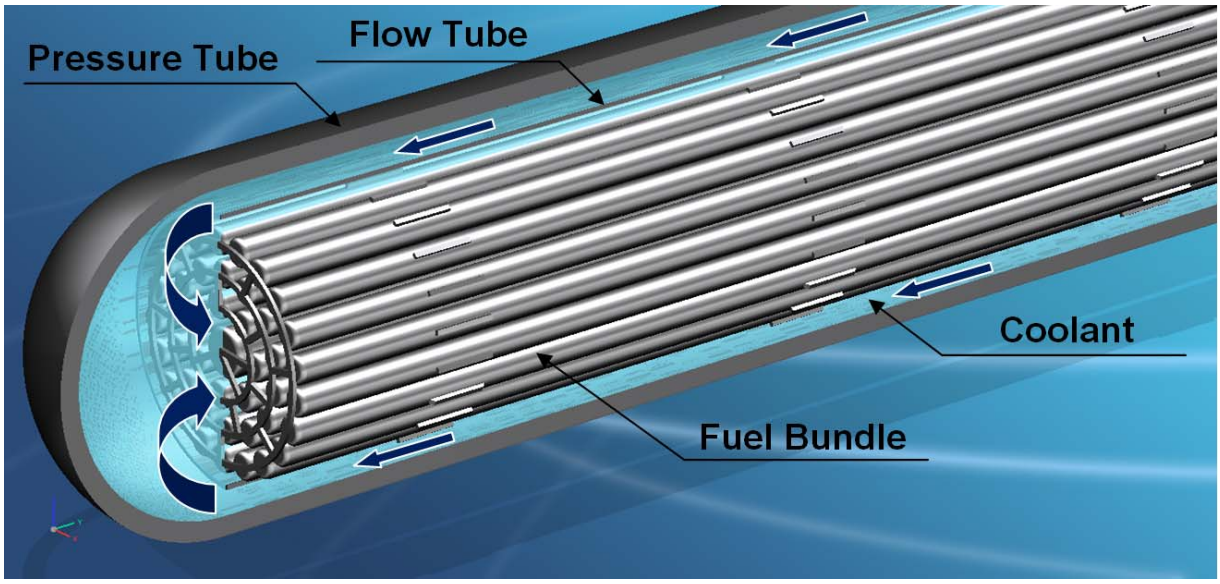


Figure 2: 3-D View of Re-Entrant Fuel Channel (based on [11]).

1.2.2 High Efficiency Fuel Channel

The High Efficiency Channel (HEC) design is a direct-flow fuel channel concept, which consists of 12 fuel bundles, a perforated liner tube, a ceramic insulator, and a pressure tube. In order to minimize the heat losses from the coolant to the moderator a porous ceramic insulator, which is made of Yittria Stabilized Zirconia (YSZ), is placed between the "hot" coolant and "cold" PT. In addition to minimizing the heat losses from the coolant, the ceramic insulator reduces the operating temperature of PT. This allows for the use of currently available materials such as Zr-2.5% Nb, which have low absorption cross sections for thermal neutrons [11]. The liner is a perforated tube made of stainless steel. The ultimate purpose of the liner is to protect the ceramic insulator from being damaged during operation and refuelling due to stresses introduced by the fuel bundles and from erosion by the coolant flow. Figure 3 shows a 3-D view of HEC.

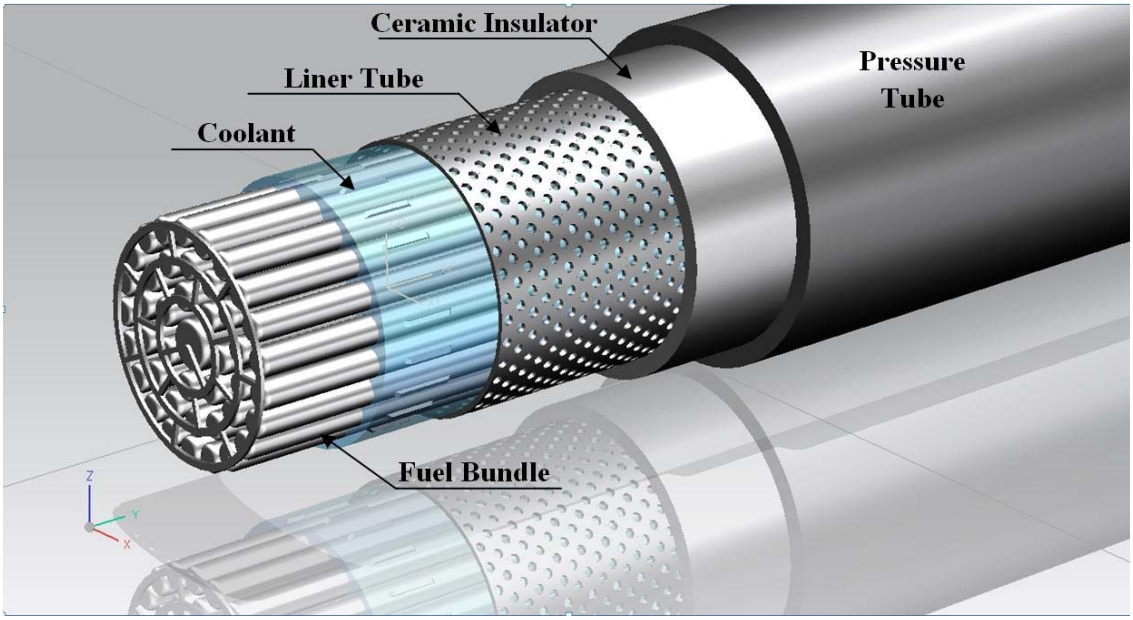


Figure 3: 3-D View of High Efficiency Channel (based on [11]).

2. Alternative Fuels

A potential fuel must have a high melting point, high thermal conductivity, good irradiation, and mechanical stability [11] due to high operating temperatures of SCWRs. These requirements eliminate various nuclear fuels categorized under metallic fuels because of their low melting point, high irradiation induced creep, and high irradiation swelling [11]. On the other hand, ceramic fuels have promising properties, which make these fuels suitable candidates for SCWR and other high-temperature applications.

In terms of thermophysical properties of a fuel, melting point and thermal conductivity are the most important properties. The thermal conductivity of the fuel governs the rate of heat transfer removal from the fuel under specific conditions (e.g., mass flow rate, heat flux, and fuel bundle geometry). UO_2 has been used as the fuel of choice in PWRs and BWRs. As shown in Fig. 4, the thermal conductivity of UO_2 is between 2 and 3 W/m K within the operating temperature range of SCWRs. On the other hand, fuels such as UN, UC, UC_2, and UO_2-C have significantly higher thermal conductivities compared to that of UO_2. Thus, under the same conditions, the fuel centerline temperature of these fuels should be lower than the UO_2 fuel centerline temperature. As a result, this paper only focuses on several high thermal conductivity fuels, which are UC, UN, and UO_2-C. Table 2 provides basic properties of several selected fuels at 0.1 MPa and 298K [13-19]. The following sections provide a brief review of these fuels. In addition, a comprehensive comparison of these fuels, specifically UC and UN, has been provided in Section 3.

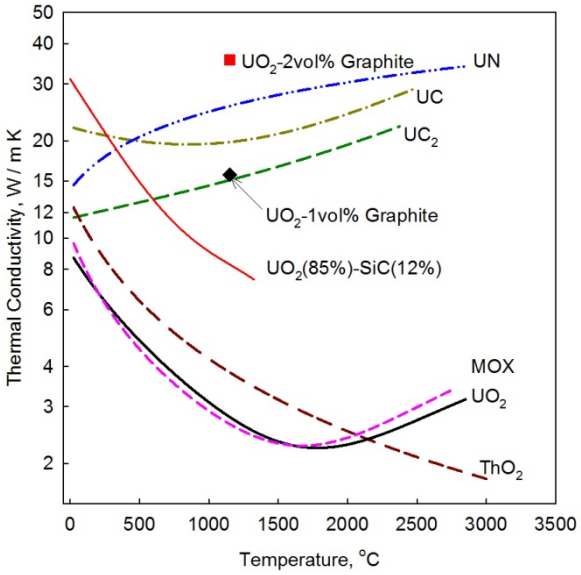


Figure 4: Thermal Conductivities of UO_2, UN, UC, and UC_2, UO_2 Plus Graphite-Fiber Fuels as a Function of Temperature [13-19].

Table 2: Properties of Uranium Dioxide, Uranium Mononitride, Uranium Monocarbide, and Uranium Dicarbide at 0.1 MPa and 298K [13-18; 20-21].

Property	**UO_2**	**UC**	**UC_2**	**UN**
Molecular Mass, amu	270.3	250.04	262.05	252.03
Theoretical density, kg/m^3	10960	13630[15]	11680[15]	14300
Melting Point, °C	2847±30	2507[16] 2532[17]	2375[16] 2562[17]	2850±30[1]
Heat Capacity J/kgK	235	203[18]	233[13]	190
Heat of Vaporization, kJ/kg	1530	2120	1975±20[3]	1144[2] 3325[3]
Thermal Conductivity, W/mK	8.68	21.24	11.57	14.58
Linear Expansion Coefficient, 1/K	9.75 10^{-6}	10.1 10^{-6}	(18.1 10^{-6})[4]	7.52 10^{-6}
Electric Resistivity, Ω·m	7.32 10^{-8}	72.7 10^{-8}	(1.2 10^{-6})[15]	1.46 10^{-6}
Crystal Structure	FCC	FCC	BCT, *t* <1820°C FCC, *t* >1820°C	FCC

[1] at nitrogen pressure ≥ 0.25 MPa
[2] $UN(s)=U(l)+0.5N_2(g)$, [20]
[3] $UN(s)=U(g)+0.5N_2(g)$, [20]
[4] At 1000°C, [21]

2.1 Uranium Carbide

Thermophysical properties of UC, specifically its melting point and thermal conductivity, have made this fuel a suitable candidate for use in various high-temperature applications including SCWRs. For instance, UC has been proposed as the fuel for a SCWR concept in Russia [2]. In terms of thermophysical properties, UC has a high melting point approximately 2507°C and a high thermal conductivity, above 19 W/m K at all temperatures up to the melting point. UC has a density of 13630 kg/ m^3, which is lower than that of UN but higher than those of UO_2 and UC_2. Moreover, UC has a higher fissile density compared to UO_2 but lower than that of UN.

2.2 Uranium Nitride

Uranium mononitride or uranium nitride can be produced by the carbothermic reduction of uranium dioxide plus carbon. This process produces UN with densities in the range of 65% to 90% of TD [22]. Uranium mononitride has a high melting point, high thermal conductivity, and high radiation stability. These properties enhance the safety of operation and allow the fuel to achieve high burn-ups. Additionally, UN has the highest fissile density, which is approximately 1.4 times that of UO_2 and greater than those of other examined fuels. In contrast, one disadvantage of the UN fuel is that under some conditions it decomposes to liquid uranium and gaseous nitrogen [13, 14], increasing the possibility of releasing gaseous fission products due to the formation of cracks in the fuel.

2.3 Uranium Dioxide Composed of Graphite Fibers

The desire for the advancement of the nuclear technology has led to the development of high-thermal conductivity fuels such as UC and UN. Similarly, high thermal conductivities limit the release of gaseous fission products [23]. There is a possibility to increase the thermal conductivity of a low-thermal conductivity fuel such as UO_2 by adding long, thin fibers of a high thermal-conductivity material to the fuel. A high thermal-conductivity material must have a low thermal-neutron absorption cross-section, assuming that the fuel will be used in a thermal-spectrum nuclear reactor [23]. Additionally, it must have a high melting point and be chemically compatible with the fuel, the sheath, and the coolant. Silicon and carbon are good candidates because they have low neutron absorption cross sections, high thermal conductivities, and high melting points. In this paper, the fuel centerline temperature of the UO_2-C fuel has been calculated at SCWR conditions.

3. Calculation of Fuel Centerline Temperature

In order to calculate the fuel centerline temperature, steady-state one-dimensional heat-transfer analysis has been conducted. The MATLAB and NIST REFPROP software have been used for programming and retrieving thermodynamic properties of a light-water coolant, respectively. First, the heated length of the fuel channel is divided into small segments of one-millimeter lengths. Second, temperature of the coolant is calculated for each node. Third, inner and outer surface temperatures of the sheath are calculated. Finally, fuel centerline temperature of the fuel is calculated. It should be mentioned that it has been assumed that there is no gap between the inner surface of the sheath and the outer surface of the fuel pellets. The following equations in sequence were used to determine the coolant, sheath [24], and fuel centerline temperature profiles.

$$H_{i+1} = H_i + \frac{p \cdot q_x}{\dot{m}} \cdot \Delta x \quad (1)$$

Equation (1) was used to calculate the enthalpy profile of the coolant. Then, NIST REPFROP was used to determine the corresponding temperature profile of the coolant based on calculated enthalpies. In other words, for each point along the heated length of the fuel channel enthalpy was calculated. Next, the calculated enthalpy and pressure of the coolant were entered into the NIST REFPROP software as two independent variables to calculate the corresponding temperature of the coolant.

$$T_{sheath} = \frac{q}{h} + T_{coolant} \tag{2}$$

Since the temperature profile of the coolant was calculated based on Eq. (1), the outer surface temperature of the sheath was calculated using Eq. (2). The latter equation requires the calculation of the Heat Transfer Coefficient (HTC) between the sheath-wall and the coolant. HTC was calculated using the Mokry et al. correlation shown as Eq. (3). Then, the inner sheath temperature was calculated based on conduction through the sheath using Eq. (4).

The experimental data, on which the Mokry et al. correlation was developed, were obtained within conditions similar to those of proposed SCWR concepts. The experimental dataset was obtained for supercritical water flowing upward in a 4-m-long vertical bare tube. The data was collected at a pressure of approximately 24 MPa for several combinations of wall and bulk fluid temperatures which were below, at, or above the pseudocritical temperature. The mass flux ranged from 200-1500 kg/m^{2}s; coolant inlet temperature varied from 320 to 350°C, for heat flux up to 1250 kW/m^2 [24].

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

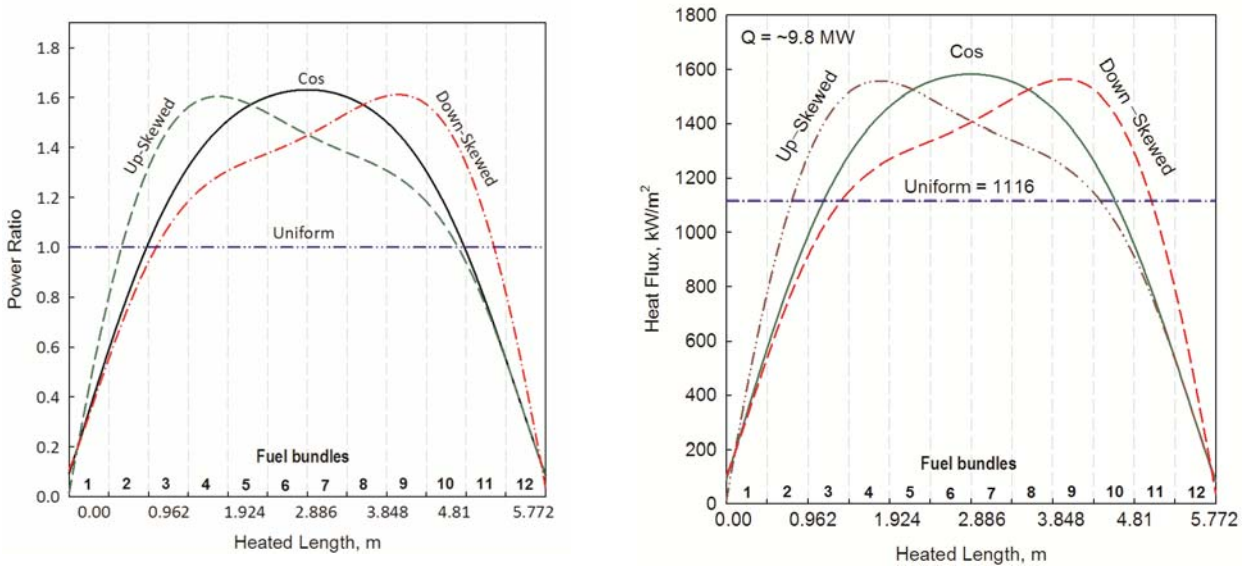
$$T_{sheath,i} = T_{sheath,o} + Q \cdot \frac{ln(r_o/r_i)}{2\pi Lk} \tag{4}$$

The outer surface temperatures of the fuel pellets are equal to the inner surface temperatures of the sheath due to the assumption that there is no gap between the fuel pellets and the sheath. Equation (5) was used in order to calculate the fuel centerline temperature profile. It should be mentioned that the radius of the fuel was divided into 20 segments in order to increase the accuracy of the calculations.

$$T_{r,i+1} = \frac{Q_{gen}(r_i^2 - r_{i+1}^2)}{4 \cdot k_{avg}} + T_{r,i} \tag{5}$$

In Equation (1), q is the heat flux value, which varies along the axial direction of the fuel channel. In this paper, several Axial Heat Flux Profiles (AHFPs) have been used to calculate the fuel centerline temperature at the channels with the maximum thermal power. The maximum channel power was assumed to be 15% (10% variation in thermal power and 5% uncertainty) above the average thermal power. These AHFPs, which include cosine, upstream-skewed cosine, downstream-skewed cosine, and uniform, listed in Reference [24]. Figure 5 shows the power ratios based on which AHFPs have been determined. The power ratio has been defined as the ratio of the local heat flux to the average heat flux. A generic 43-element bundle (e.g. Variant-20) was used in order to determine the average heat flux. Our next intent is to develop a diffusion code coupled with a lattice code in order to determine the thermal power and flux in each fuel channel of the core. This physics calculation will

allow us to determine the radial power profiles as well as required enrichment for each fuel. However, in the present paper it has been assumed that the maximum thermal power per channel is 15% above the average thermal power per channel of 8.5 MW. Consequently, heat flux profiles have been calculated based on a maximum thermal power per channel of 9.8 MW_{th}.



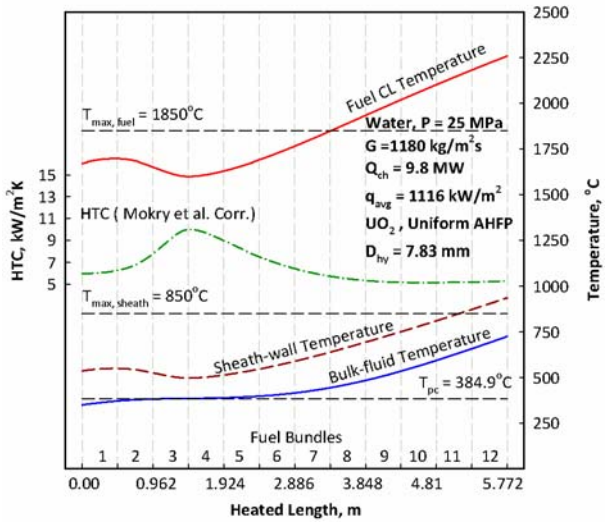
(a) **(b)**

Figure 5: (a) Power Ratios [24], and (b) Heat Fluxes for Uniform, Cosine, Upstream-Skewed, and Downstream-Skewed Profiles for SCW Fuel Channels with Maximum Thermal Power.

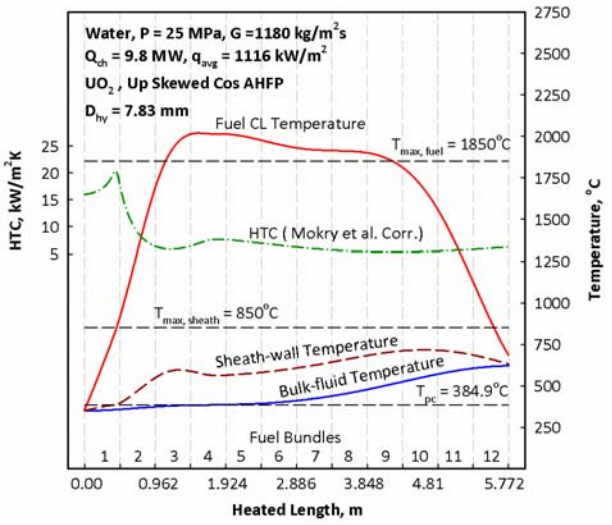
4. Results and Discussion

The fuel centerline temperature was calculated at SCW channel (~ 9.8 MW_{th}) conditions. A steady-state one-dimensional heat transfer analysis was conducted with fuel channel specifications as follows: a mass flow rate of 4.4 kg/s, a constant pressure of 25 MPa, a coolant inlet temperature of 350°C, a thermal power per channel of 9.8 MW_{th}. The heat flux profiles were calculated based on a 43-element fuel bundle known as the Variant-20 fuel bundle. Each of the 42 fuel elements of the Variant-20 fuel bundle has an outer diameter of 11.5 mm while the thickness of the sheath has been determined to be 0.47 mm. Further, it was assumed that there is no gap between the fuel and the cladding. Therefore, the outer diameter of the fuel pellets was 10.56 mm. Inconel-600 was chosen as the material of the sheath. Moreover, several nuclear fuels were examined for the purpose of this study.

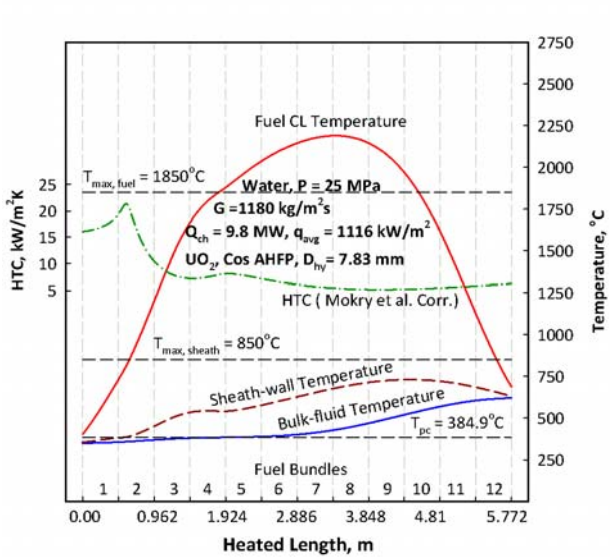
The examined fuels were UN, UC, and UO_2 composed of 1 vol% graphite-fiber with 95% TD. For each fuel, the fuel centerline temperature was analyzed for uniform, cosine, upstream-skewed cosine and downstream-skewed cosine AHFPs calculated based on the maximum thermal power per channel of 9.8 MW_{th}. Additionally, the fuel centerline temperature was calculated for a UO_2 fuel with 95% TD in order to provide a basis for comparison. Figures 6 through 8 show the coolant, sheath, and fuel centerline temperature profiles along the heated length of the fuel channel for UO_2, UC, UN, and UO_2-C fuels, respectively. Since the maximum fuel centerline temperatures were reached when the downstream-skewed cosine AHFP was applied only the results corresponding to the downstream-skewed cosine AHFP have been presented for UN and UO_2-C.



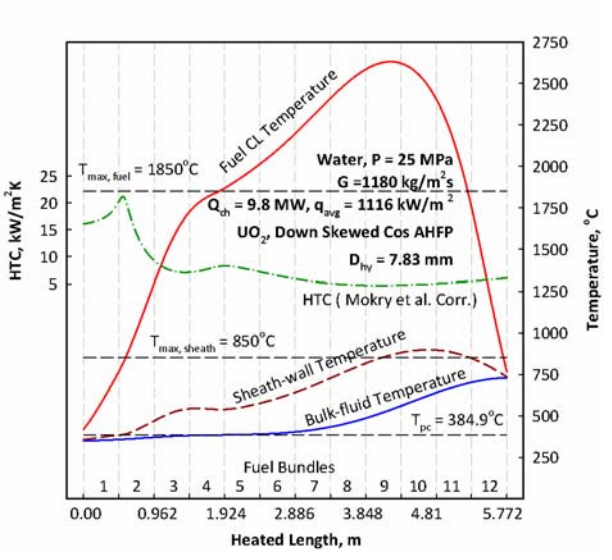
a) **Uniform AHFP.**



b) **Upstream-Skewed Cosine AHFP.**

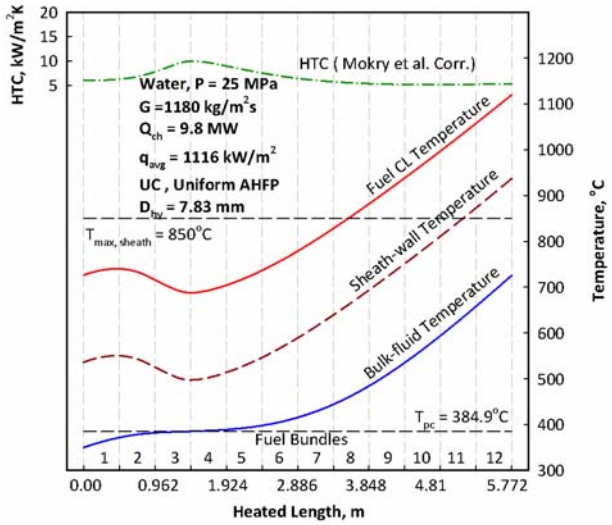


c) **Cosine AHFP.**

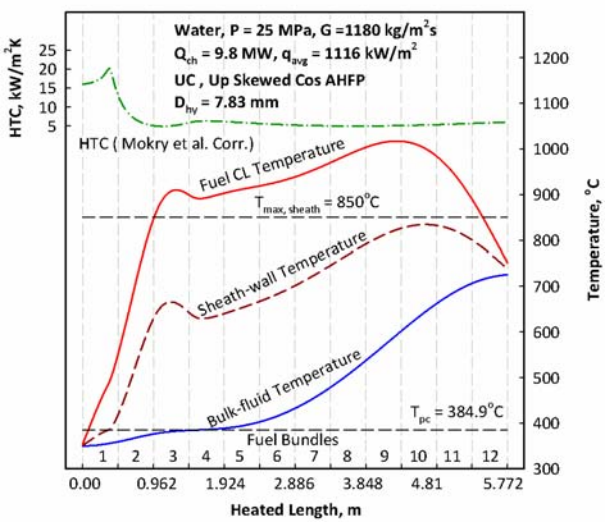


d) **Downstream-Skewed Cosine AHFP.**

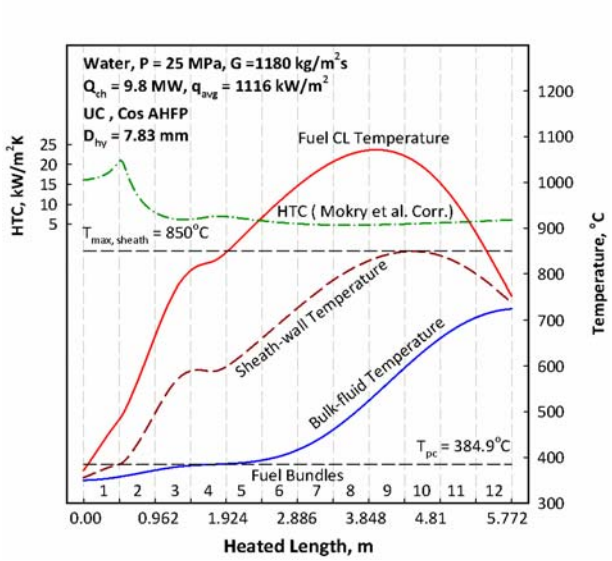
Figure 6: Temperature and HTC profiles for UO_2 Fuel at Maximum Channel Power.



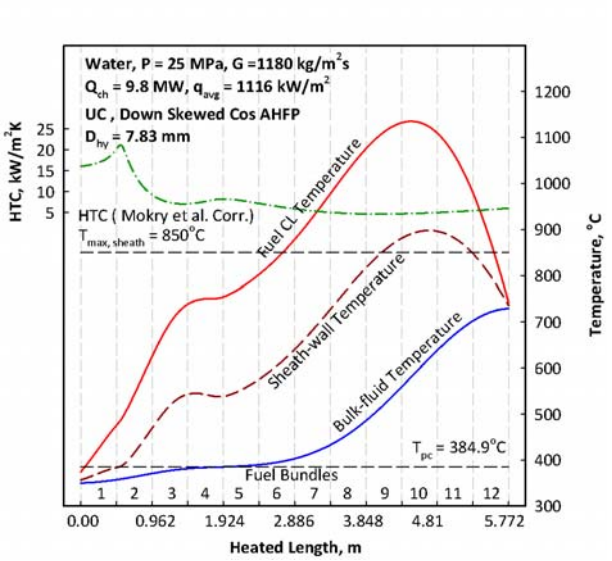
a) **Uniform AHFP.**



b) **Upstream-Skewed Cosine AHFP.**

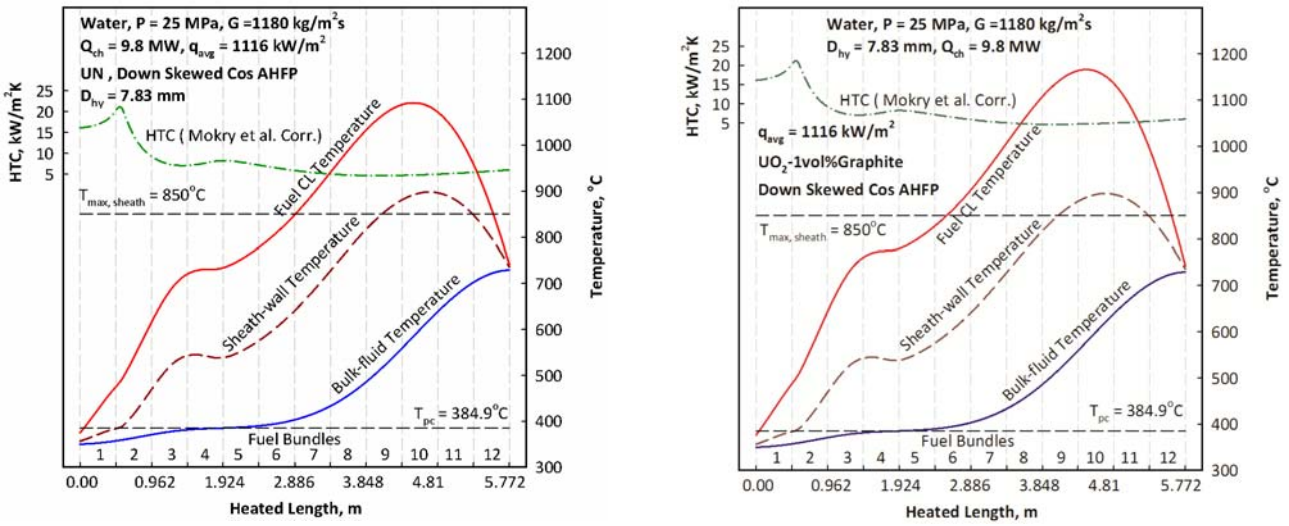


c) **Cosine AHFP.**



d) **Downstream-Skewed Cosine AHFP.**

Figure 7: Temperature and HTC profiles for UC Fuel at Maximum Channel Power.



a) Downstream-Skewed Cosine AHFP. **b) Downstream-Skewed Cosine AHFP.**

Figure 8: Temperature and HTC profiles for a) UN, and b) UO_2 Fuel Composed of 1 vol % Graphite Fibers at Maximum Channel Power.

As already indicated, the fuel centerline temperature and the sheath temperature must comply with the temperature limits of 1850°C [25] and 850°C (design limit) [11], respectively. The result of this analysis showed that in fuel channels with maximum thermal power (e.g., 15% above the average power per channel), the fuel centerline temperature exceeds the industry limit when UO_2 fuel is utilized in Variant-20 Fuel bundles. The maximum UO_2 fuel centerline temperature reached approximately 2630°C while the established fuel centerline temperature limit is 1850°C for the UO_2 fuel. In contrast, the maximum centerline temperatures of the UC, UN, and UO_2-C fuels were approximately 1135°C, 1092°C, and 1166°C, respectively. Thus, the fuel centerline temperatures of these fuels are well below their melting points and the established limit such that an adequate safety margin is left. However, it should be noted that the fuel centerline temperature limit does depend on the fuel and may vary for fuels other than UO_2.

In regards to the fuel centreline temperature limit, there are several factors that may affect a fuel centerline temperature limit for a fuel. These factors include melting point, high-temperature stability, and phase change of the fuel. The melting point of a fuel plays an important role in determining a fuel centerline temperature for that fuel. For instance, the accepted fuel centerline temperature limit (e.g. 1850°C) of the UO_2 fuel is approximately 1000°C below its melting point. Similarly, the corresponding limit for the UC fuel would be 1500°C since the melting point of UC is approximately 2500°C. In regards to UN, this fuel decomposes to uranium and gaseous nitrogen at temperatures above 1600°C. Therefore, the fuel centerline temperature limit for UN should be lower than that of UO_2 under normal operating conditions. As a result, a fuel centerline temperature limit of 1500°C has been recommended for UN [12].

In addition to fuel centerline temperature, the sheath temperature must be kept below a certain limit established, based on the material of the sheath. When the sheath temperature was analyzed, it was determined that the sheath temperature exceeded the design limit of 850°C under some conditions (e.g.,

at downstream skewed cosine AHFP). Therefore, it is necessary to implement some changes to the design of the fuel bundle or the fuel channel specifications (e.g., mass flow rate of the coolant).

There are other factors in addition to the fuel centerline temperature and sheath temperature that should be considered in the process of evaluating various fuels. The most important issues associated with nuclear fuels for SCWRs and other high-temperature applications include melting point, evaporation, high temperature chemical stability, release of fission products, radiation-induced swelling, thermal shock resistance, density, high temperature creep, and mass of fissile elements [17]. All examined fuels (e.g., UC, UN, and UO_2-C) have relatively high melting points and high thermal conductivities, which lead to lower fuel centerline temperatures than those of low thermal-conductivity fuels (e.g., UO_2, MOX, or ThO_2) for a given thermal power and fuel bundle geometry. Thus, other factors should be considered in order to determine the best fuel options.

The atomic density of uranium is another important factor. Both UN and UC have high uranium atom density, approximately 1.40 and 1.34 times that of UO_2. Additionally, UC has a higher fission-to-capture ratio of 54.3%, which is higher than 43.7% for UN. This indicates that a higher neutron economy is achieved when the UC fuel is used. On the other hand, the fission-to-capture ratio for UO_2 is higher than that of UC, but UO_2 has a smaller uranium atom density compared to those of UN and UC while UN has the highest value. A high uranium atom density indicates a smaller core size which in turn reduces the costs. Thus, both UN and UC result in smaller core sizes, which in turn reduce the capital cost of the plant. Moreover, the UC fuel enhances the neutron economy due to its higher fission-to-capture ratio.

In terms of irradiation-induced creep, both UN and UC have significantly lower irradiation –induced creep rates compared to UO_2 [26]. Several authors [26-29] have studied the steady-state creep strength and irradiation-induced creep of the UN and UC fuels, and provided several correlations, which have been used in order to determine the creep rates of fully dense UN and UC for a stress of 25 MPa. The results, shown in Fig. 9, demonstrate that when the UC and UN fuels are compared, the irradiation-induced creep rate of UC was lower than of UN at 1500°C. In other words, UC has a better creep strength and resistance to deformation than UN. With the UC fuel, it is recommended to use hyperstoichiometric UC [28] because it has a lower steady-state creep rate compare to hypostoichiometric UC. Additionally, hyperstoichiometric UC has a higher mechanical strength than hypostoichiometric UC due to higher values of long-range stress [28], which result in higher proportional limit values. For instance, the proportional limit of hyperstoichiometric UC is 32 times that of the hypostoichiometric UC [28]. As a result, hyperstoichiometric UC has better mechanical behaviour at high temperatures than hypostoichiometric UC and UN.

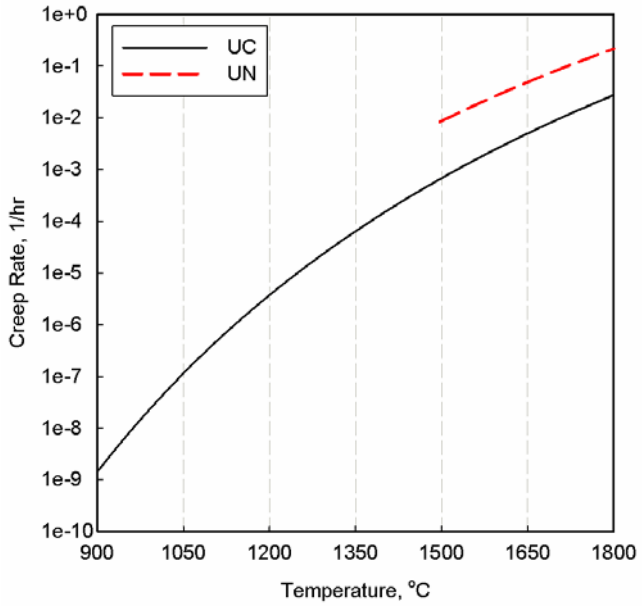
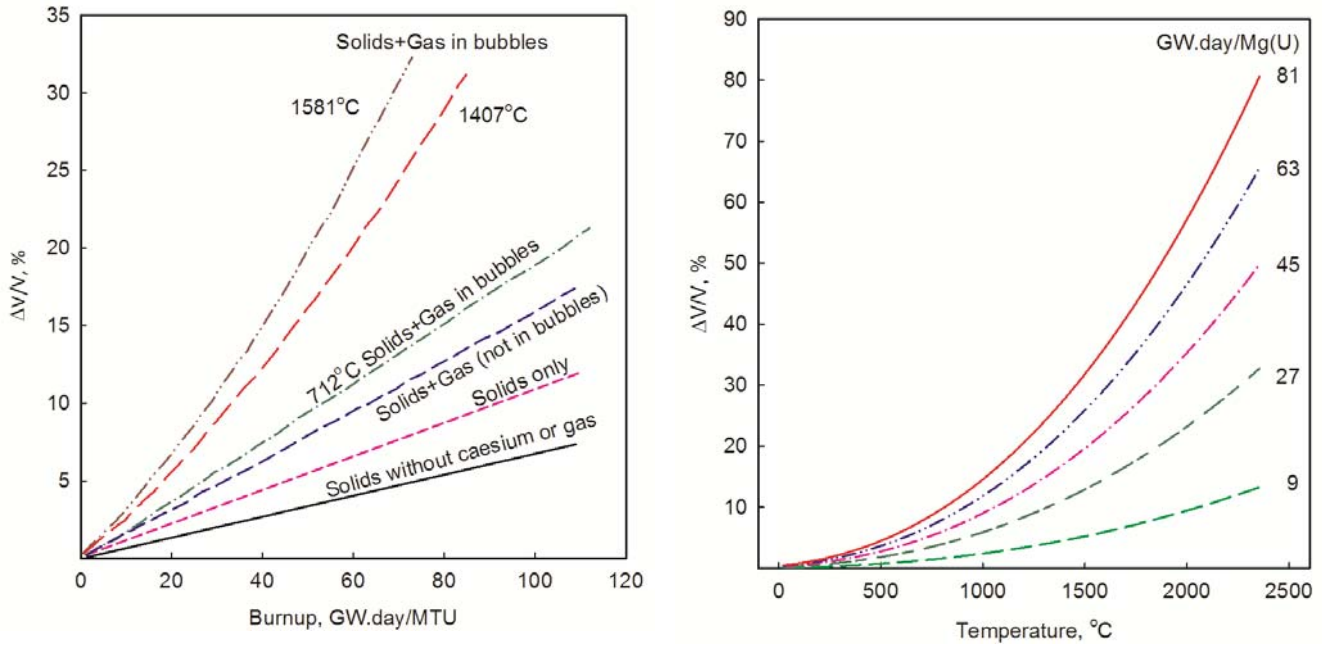


Figure 9: Creep Rates of UC and UN at 25 MPa stress and 95%TD.

In addition to creep resistance, hardness is another mechanical property, which is an indication of the resistance of a material to deformation. According to [26], the hardness values, in kg/mm^2, are 100, 120, and 50 for $UC_{1.05}$, $UC_{0.98}$, and UN respectively. The result of their investigation shows that UC has a higher hardness compared to UN; therefore, UC has a higher resistance against deformation which in turn increases the mechanical integrity of the fuel under operating conditions of SCWRs and other high-temperature nuclear applications.

The formation of fission products results in exertion of stresses on the sheath due to the swelling of the fuel if the fission products are retained in the fuel. A comparison between the volumetric swelling of the UN and UC fuels shows that the percent volumetric swelling of UN is higher than that of UC. As shown in Fig. 10, the percent volumetric swelling of UN is approximately 17% and that of UC 12%, approximately at 1400°C and a burn-up of 40 GW.day/Mg(U) [15, 30]. There is a possibility to reduce the swelling of both fuels by increasing the porosity of the fuel [15]; however, the fission gas release is higher for fuels with high porosity compared with dense fuels, which have less porosity. Therefore, either a high volumetric swelling or a high fission gas release may result the exertion of high stresses on the sheath. Nevertheless, UC has a lower percent volumetric swelling compared to UN. The lower volumetric swelling of UC safeguards the mechanical integrity of both the fuel and the sheath and minimizes the possibility of any sheath failure.



a) **b)**

Figure 10: Percent Volumetric Swelling as a Function of Burn-up and Temperature for a) UC and b) UN [15, 30].

The thermal shock resistance of a nuclear fuel is an indication of the degree to which the fuel withstands sudden changes in temperature. A low thermal shock resistance may result in the formation of cracks in the fuel which in turn reduces the mechanical integrity and the thermal conductivity of the fuel and increases the fission product release rate. As indicated by Eq. (6) [31], the thermal shock resistance of a fuel depends on its thermal conductivity, compressive strength, Poisson's ratio, coefficient of thermal expansion, and Young' modulus of elasticity. The thermal shock resistances of UC, UN, and UO_2 have been calculated based on Eq. (6) for a temperature range between 800°C and 1800°C. All required properties were calculated for 95% TD fuels except the linear thermal expansion coefficient, which was based on 100% TD fuels. The result shows that the thermal shock resistances of both UN and UC are 5 to 15 times higher than those of UO_2 within the examined temperature range. The low thermal shock resistance of UO_2 is mostly due to its low thermal conductivity, which makes this fuel vulnerable to sudden changes in temperature at high operating temperatures. Thus, UN and UC have significantly higher thermal shock resistances compared with UO_2.

$$R' = \frac{k \cdot \sigma (1-\nu)}{\alpha \cdot E} \qquad (6)$$

The chemical compatibility of a nuclear fuel with coolant, which is an essential factor that affects the integrity of the fuel, can be studied in terms of the oxidization behaviour of the fuel when exposed to the coolant. For instance, the UO_2 fuel is stable in water and has a high resistance to oxidation in light-water and heavy-water at the LWR and HWR conditions (e.g., up to 320°C). However, UO_2 oxidizes at temperatures above 320°C if it comes in direct contact with air or water in case of the sheath breach [12]. Similarly, UC has a poor resistance to oxidization when it comes in contact with water even at temperatures as low as 55°C [12]. Likewise, UN oxidizes in water at temperatures above 100°C due to the deformation of the protective layer, which is formed on the surface of UN. Additionally, the

oxidation resistance of UN is highly dependent on deviation from stoichiometry [12]. In other words, the presence of free uranium or U_2N_3 significantly increases the oxidation rate. Therefore, the detrimental effects of exposure of these fuels to water require further consideration.

In terms of high temperature stability, two issues are associated with UN. First, hyperstoichiometric UN co-exists with uranium sequinitride (U_2N_3) in the temperature range of approximately 1075°C and 1375°C for hyperstoichiometric UN with N/U atomic ratios approximately between 1.2 and 1.5 [32]. Uranium sequinitride decomposes to UN and nitrogen at temperatures approximately above 1375°C. The release of nitrogen gas results in severe cracking of the fuel. This problem can be solved by using hypostoichiometric UN. However, hypostoichiometric UN has a higher fission gas release than hyperstoichiometric UN [32]. Moreover, if UN is chosen as a nuclear fuel, hypostoichiometric UN with adequate porosity should be utilized in order to minimize the negative impacts of the decomposition of U_2N_3 and accommodate for the fission products.

Another issue related to the UN fuel is that UN decomposes to uranium and nitrogen gas, which leads to cracking of the fuel due to the release of nitrogen. The results of several studies have indicated that within a temperature range of 1130°C and 1800°C the incongruent vaporization of hypostoichiometric UN leads to the release of nitrogen and the formation of free uranium [33, 35]. Additionally, the experimental results of most authors indicate that UN dissociates at temperatures higher than 1600°C, which is in agreement with other values published in the literature. Therefore, the release of nitrogen gas and formation of cracks in the fuel should be studied thoroughly if UN is chosen as the fuel of choice for SCWRs, but it should be mentioned that this effect might not be significant since the maximum fuel centerline temperature for the UN fuel is below 1100°C under normal operating conditions of SCWRs.

Regardless of the issues associated with the stability of UN at high temperatures, if UN is used as a fuel, it must be enriched in ^{15}N in order to avoid the formation of ^{14}C. This additional process increases the fuel cost [35].

5. Conclusions

In order to investigate the possibility of the use of high thermal-conductivity fuels, the fuel centerline temperature for UC, UN, and UO_2-1 vol % graphite-fiber was calculated. Additionally, the fuel centerline temperature was calculated for the UO_2 fuel to provide a reference for comparison. The fuel centerline temperature reached a maximum of 2630°C when the UO_2 fuel was analyzed at the operating conditions of SCW fuel channels with the maximum thermal power per channel of approximately 9.8 MW. On the other hand, the fuel centerline temperatures of UC, UN, and UO_2-C fuels were well below their melting points and the established limit of 1850°C for the fuel centerline temperature. Thus, the result of fuel centerline calculation supports the potential use of high thermal-conductivity fuels such as UC, UN, and UO_2-C in SCWRs.

One of the most important thermodynamic properties of a fuel is its thermal conductivity. There is a possibility to increase the thermal conductivity of a low thermal-conductivity fuel such as UO_2 by adding long, thin fibers of graphite into the fuel (e.g., UO_2-1 vol % graphite-fiber). However, further research and development is required to determine the effects of irradiation on these new fuels and their thermodynamic and mechanical properties. On the other hand, unlike UO_2, the thermal conductivities of UC and UN increase at high temperatures (e.g., temperatures above 1000°C). The thermal conductivity trend of these fuels increases the margin between the operating temperature of the

fuel and its established temperature limit, and enhances the safety of operation. Therefore, high thermal-conductivity fuels such as UC and UN are more suitable for future use in SCWRs.

One concern with the UC fuel is its chemical compatibility with water, which remains ambiguous due to the discrepancy between the two available sources (e.g.12, 14), and requires further investigation and research. On the other hand, the two main concerns associated with the UN fuel are its dissociation at temperatures over 1600°C and the oxidation reaction between UN and water. Furthermore, the UN fuel must be enriched in ^{15}N in order to avoid the formation of ^{14}C. Therefore, these issues must be contemplated and reflected on the final decision.

When mechanical properties are examined, the UC fuel has a higher hardness, lower volumetric swelling, lower thermal expansion, and acceptable thermal shock resistance. Additionally, UC is more stable at high temperatures than UN. Moreover, UC has a higher fission-to-capture ratio than of UN while its uranium atom density is comparable with that of UN and higher than that of UO_2. Consequently, according to the available literature and the judgment of the authors, the UC fuel demonstrates desirable mechanical, thermodynamic, and neutronic properties, which make the UC fuel a promising candidate for the future use in SCWRs. However, the oxidation reaction between UC and water might be an issue; therefore, a study concerning the chemical compatibility of UC with water at high temperatures should be performed for adopting UC.

6. Acknowledgements

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

7. Nomenclature

A	cross-sectional area, m^2
A_{fl}	flow area, m^2
B	burn-up, MW.day/Mg(U)
c_p	specific heat at constant pressure, J/kg K
$\bar{c}_p$	average specific heat, $(\frac{H_w - H_b}{T_w - T_b})$, J/kg K
D	diameter, m
D_{hy}	hydraulic diameter, m
E	Young's modulus, MPa
G	Gibb's free energy, J/kg
G	mass flux, (m/A_{fl}), kg/m^2 s
H	enthalpy, J/kg
h	heat transfer coefficient, W/m^2 K
HD	hardness, MPa
k	thermal conductivity, W/m K
L	length, m
m	mass flow rate, kg/s
P	pressure, Pa
P	percent porosity
p	heated perimeter, m
Q	heat transfer rate, W

q	heat flux, W/m^2
Q_{gen}	volumetric heat generation, W/m^3
R	gas constant, cal/ K mol
R'	thermal shock resistance, W/K
T	temperature, K
t	temperature, °C

Greek symbols

α	coefficient of thermal expansion, 1/K
α	thermal diffusivity, ($k/\rho\ C_p$), m^2/s
$\acute{\varepsilon}$	creep rate, 1/hr
μ	dynamic viscosity, kg/m s
ν	kinematic viscosity, m^2/s
ν	Poisson's ratio
ρ	density, kg/m^3
σ	stress, Pa
σ_F	modulus of rupture, MPa

Non-dimensional numbers

$\mathbf{Nu}_D$	Nusselt number, $\mathbf{Nu}_D = h \cdot D/k$
$\mathbf{Pr}$	Prandtl Number, $\mathbf{Pr} = \mu \cdot c_p/k$
$\overline{\mathbf{Pr}}$	average Prandtl Number, $\overline{\mathbf{Pr}} = \mu \cdot \bar{c}_p/k$
$\mathbf{Re}_D$	Reynolds number, $\mathbf{Re_D} = \boldsymbol{G} \cdot \frac{\boldsymbol{D_{hy}}}{\boldsymbol{\mu}}$

Subscripts

atm	atmospheric
b	properties calculated at bulk fluid temperature
cond	conduction
conv	convection
i	inner
m	melting
o	outer
pc	pseudocritical point
vol	volume
w	properties calculated at wall temperature

Abbreviations

AECL	Atomic Energy of Canada Limited
AGR	Advanced Gas-cooled Reactor
AHFP	Axial Heat Flux Profile
BCT	Body-Centered Tetragonal
BWR	Boiling Water Reactor
CANDU	CANada Deuterium Uranium
FCC	Face-Centered Cubic
GFR	Gas-cooled Fast Reactor
GIF	Generation IV International Forum

HTC	Heat Transfer Coefficient
HTR	High Temperature Reactor
IAEA	International Atomic Energy Agency
LFR	Lead-cooled Fast Reactor
LOCA	Loss Of Coolant Accident
LWR	Light Water Reactor
MSR	Molten Salt Reactor
NIST	National Institute of Standards and Technology (USA)
NPP	Nuclear Power Plant
PCh	Pressure Channel
PT	Pressure Tube
PV	Pressure Vessel
RBMK	Russian Acronym for Reactor of High-Power Channel-type
SCW	SuperCritical Water
SCWR	SuperCritical Water-cooled Reactor
SFR	Sodium-cooled Fast Reactor
SHR	Steam Re-Heat
TD	Theoretical Density
UC	Uranium Carbide
UC_2	Uranium Dicarbide
UN	Uranium Nitride
UO_2	Uranium Dioxide
UO_2-C	Uranium Dioxide composed of graphite fibers
VHTR	Very-High-Temperature Reactor

8. References

[1] U.S. DOE Nuclear Energy Research Advisory Committee, "A Technology Roadmap for Generation IV Nuclear Energy Systems", Retrieved 2010 July 12, from the Generation IV International Forum: http://www.ne.doe.gov/genIV/documents/gen_iv_roadmap.pdf.

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

[3] G., Naterer, S., Suppiah, M., Lewis, "Recent Canadian Advances in Nuclear-Based Hydrogen Production and the Thermochemical Cu-Cl Cycle", International Journal of Hydrogen Energy (IJHE), Vol. 34,2009, pp. 2901-2917.

[4] G.F. Naterer, S. Suppiah, L. Stolberg et al., "Canada's Program on Nuclear Hydrogen Production and the Thermochemical Cu-Cl Cycle", International Journal of Hydrogen Energy (IJHE), Vol. 35, 2010, pp. 10905-10926.

[5] L. Grande, S. Mikhael, B. Villamere, A. Rodriguez-Prado, L. Allison, W. Peiman and I. Pioro, "Thermal Aspects of Using Uranium Nitride in Supercritical Water-Cooled Nuclear Reactors", Proceedings of the 18th International Conference On Nuclear Engineering (ICONE-18), Xi'an, China, 2010 May 17-21.

[6] I. Pioro, S. Mokry, W. Peiman, L. Grande and Eu. Saltanov, "Supercritical Water-Cooled Nuclear Reactors: NPP Layouts and Thermal Design Options of Pressure Channels", Proceedings of the 17th Pacific Basin Nuclear Conference (PBNC-2010), Cancun, Mexico, 2010 October 24-30.

[7] B. Villamere, L. Allison, L. Grande, S. Mikhael, A. Rodriguez-Prado and I. Pioro, "Thermal aspects for uranium carbide and uranium dicarbide fuels in supercritical water-cooled nuclear

reactors", Proceedings of the 17th International Conference On Nuclear Engineering, Brussels, Belgium, 2009 July 12-16.

[8] M. Naidin, I. Pioro, R. Duffey, U. Zirn and S. Mokry, "SCW NPPs: Layouts and Thermodynamic Cycles Options", Proceeding of International Conference "Nuclear Energy for New Europe", Bled, Slovenia, 2009 Sep. 14-17.

[9] I. Pioro, S. Mokry, W. Peiman, L. Grande and Eu. Saltanov, "Supercritical Water-Cooled Nuclear Reactors: NPP Layouts and Thermal Design Options of Pressure Channels", Proceedings of the 17th Pacific Basin Nuclear Conference (PBNC-2010), Cancun, Mexico, 2010 October 24-30.

[10] L. K. H. Leung, "Effect of CANDNU Bundle-Geometry Variation on Dryout Power", Proceedings of the 16th International Conference On Nuclear Engineering, Orlando, USA, 2009 May 11-15.

[25] F. Reisch, "High Pressure Boiling Water Reactor", Journal Nuclear Technology, Vol. 172, 2009, pp. 101-107.

[11] C. K. Chow and H.F. Khartabil, " Conceptual Fuel Channel Designs for CANDU-SCWR", Journal of Nuclear Engineering and Technology, Vol. 40, 2008, pp. 1−8.

[12] B. M. Ma, "Nuclear Reactor Materials and Applications", Van Nostrand Reinhold Company Inc., New York, 1983.

[13] IAEA, "Thermophysical Properties of Materials for Nuclear Engineering: A Tutorial and Collection of Data", Vienna, Austria, 2008.

[14] P.L. Kirillov, M.I. Terent'eva and N.B. Deniskina, "Thermophysical Properties of Nuclear Engineering Materials", third ed. revised and augmented, IzdAT Publ. House, Moscow, Russia, 2007.

[15] B. R. Frost, "The Carbides of Uranium", Journal of Nuclear Materials, Vol. 10, 1963, pp. 265-300.

[16] D. Cox and A. Cronenberg, "A Theoretical Prediction of the Thermal Conductivity of Uranium Carbide Vapor", Journal of Nuclear Materials, Vol. 67, 1977, pp. 326-331.

[17] L. B. Lundberg and R. R. Hobbins, "Nuclear Fuels for Very High Temperature Applications", Intersociety Energy Conversion Engineering Conference, San Diego, USA, 1992.

[18] J. M. Leitnaker and T. G. Godfrey, "Thermodynamic Properties of Uranium Carbide", Journal of Nuclear Materials, Vol. 21, 1967, pp. 175-189.

[19] J. A. Khan, T. W. Knight, S. B. Pakala, W. Jiang and R. Fang, "Enhanced Thermal Conductivity for LWR Fuel", Journal of Nuclear Technology, Vol. 169, 2010, pp. 61-72.

[20] K. A. Gingerich, "Vaporization of Uranium. Mononitride and Heat of Sublimation of Uranium", Journal of Chemical Physics, Vol. 51, Iss. 10. 1969, pp. 4433-4439.

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

[22] R. D. Shoup and W. R. Grace, "Process Variables in the Preparation of UN Microspheres", Journal of American Ceramic Society, Vol. 60, Iss. 7-8, 1977, pp. 332-335.

[23] D. F. Hollenbach and L. J. Ott, "Improving the Thermal Conductivity of UO2 Fuel with the Addition of Graphite Fibers", Transactions of the American Nuclear Society and Embedded Topical Meeting "Nuclear Fuels and Structural Materials for the Next Generation Nuclear Reactors", San Diego, California, USA, 2010 June 13-17.

[24] S. Mokry, Ye. Gospodinov, I. Pioro and P.L. Kirillov, "Supercritical Water Heat-Transfer Correlation for Vertical Bare Tubes", Proceedings of the 17th International Conference On Nuclear Engineering, Brussels, Belgium, 2009 July 12-16.

[25] F. Reisch, "High Pressure Boiling Water Reactor", Journal Nuclear Technology, Vol. 172, 2009, pp. 101-107.

[26] J. L. Routbort and R. N. Singh, "Carbide and Nitride Nuclear Fuels", Journal Nuclear Materials, Vol. 58, 1975, pp. 78-114.

[27] D. E. Stellrecht, M. S. Farkas and D. P. Moak, "Compressive Creep of Uranium Carbide", Journal of American Ceramic Society, Vol. 51, Iss. 8, 1968, pp. 455-458.

[28] J. L. Routbort, "High-Temperature Deformation of Polycrystalline Uranium Carbide", Journal of Nuclear Materials, Vol. 44, 1972, pp. 24-30.

[29] S. Hayes, J. Thomas and K. Peddicord, "Material Property Correlations for Uranium Mononitride-II Mechanical Properties", Journal of Nuclear Materials, Vol. 171, 1990, pp. 271-288.

[30] S. B. Ross, M. S. El-Genk and R. B. Matthews, "Uranium nitride fuel swelling correlation", Journal Nuclear Materials", Vol. 170, 1990, pp. 169-177.

[31] M. Kutz, "Mechanical Engineers' Handbook, Materials and Mechanical Design", John Wiley & Sons, 2005.

[32] R. B. Matthews, K. M. Chidester, C. W. Hoth, R. E. Mason and R. L. Petty, "Fabrication and Testing of Uranium Nitride Fuel for Space Power Reactors", Journal of Nuclear Materials, Vol. 151, 1988, pp. 334-344.

[33] S. A. Balankin, L. P. Loshmanov, D. M. Skorov and V. S. Skolov, "Thermodynamic Stability of Uranium Nitride", Journal of Atomic Energy, Vol. 44, Iss. 4, 1978, pp. 327-329.

[34] K. A. Gingerich, "Vaporization of Uranium. Mononitride and Heat of Sublimation of Uranium", Journal of Chemical Physics, Vol. 51, Iss. 10, 1969, Pp. 4433-4439.

[35] S. M Oggianu, H. C. No and M. Kazimi, "Analysis of Burnup and Economic Potential of Alternative Fuel Materials in Thermal Reactors", Journal of Nuclear Technology, Vol. 143, 2003, pp. 256-269.