Thermal Aspects of Uranium Nitride, Mixed Oxide and Thoria Fuels as Applied to SuperCritical Water-Cooled Nuclear Reactors

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Abstract

Generation IV International Forum (GIF) design options consist of 6 reactor concepts. One concept is a SuperCritical Water-cooled nuclear Reactor (SCWR). The coolant in this option is light water heated and pressurized to supercritical pressures and temperatures, i.e., 25 MPa and 350 — 625°C, respectively. SuperCritical Water (SCW) Nuclear Power Plants (NPPs) are beneficial, because they will have increased thermal efficiencies by 10 — 15% compared to that of existing subcritical-water-cooled NPPs. Additionally, SCW NPPs will utilize a simplified steam circuit as they can operate with a direct cycle, eliminating the need for steam generators, steam dryers, etc. Furthermore, SCW is a single-phase fluid, which has no dryout phenomena.

The objective of this paper is to demonstrate the feasibility of alternative nuclear-fuel options such as uranium nitride (UN), Mixed OXide (MOX) and Thoria (ThO_2) in application to SCWRs. The latter two fuels are currently considered as alternatives to uranium dioxide (UO_2) accounting for fast depleting uranium resources. Moreover, the UN fuel may be a suitable fuel choice to UO_2, MOX and ThO_2 due to its higher thermal conductivity, which will have significantly lower fuel centerline temperature.

A generic pressure-tube-type SCWR fuel channel is analyzed with a 43-element Inconel-600 bundle filled with either UN, MOX or ThO_2 fuel. A uniform Axial Heat Flux Profile (AHFP) is applied.

The design constraints are: 1) fuel centerline temperature must not exceed the industry accepted limit of 1850°C and 2) sheath temperature must not exceed the design limit of 850°C. The bulk-fluid, sheath and fuel centreline temperatures and Heat Transfer Coefficient (HTC) profiles for each nuclear fuel with a uniform AHFP were calculated along the heated length of a fuel channel.

1. Introduction

SuperCritical Water-cooled nuclear Reactor (SCWR) concepts were initially developed 50 years ago. Some fossil electrical-generating plants have used supercritical "steam" as the working fluid to drive turbines. However, use of reactor coolant at supercritical temperatures and pressures has not been demonstrated, and no SCWR prototype has been constructed yet. Canada and Russia are interested in developing Pressure-Tube (PT) or Pressure-Channel (PCh) SCWR concepts. Table 1 provides a comparison of PT-SCWR concepts.

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

Parameters	Unit	SCW CANDU®[1]	KP-SKD
Reference	–	Khartabil et al. (2005) [1]	Gabaraev et al. (2005) [2]
Country	–	Canada	Russia
Organization	–	AECL	RDIPE
Reactor spectrum	–	Thermal	Thermal
Power thermal	MW_{th}	2540	1960
electrical	MW_{el}	1220	850
linear max/ave	kW/m	–	69/34.5
Thermal efficiency	%	48	42
Pressure	MPa	25	25
T_{in} **coolant**	ºC	350	270
T_{out} **coolant**	ºC	625	545
Flow rate	kg/s	1300	922
Core height	m		5
diameter	m	~7	6.45
Fuel	–	UO_2/ThO_2	UO_2
Enrichment	% wt.	4	6
Cladding material	–	Ni alloy	SS
# of fuel bundles	–	300	653
# of fuel rods in bundle	–	43	18
D_{rod}/δ_w	mm/mm	11.5 and 13.5*	10/1
T_{max} **cladding**	ºC	<850	700
Moderator	–	D_2O	D_2O

* For 43-element bundle.

Currently, SCWRs are one of six Generation IV International Forum (GIF) design options. The renewed interest is created due to an increased thermal efficiency and reduced capital costs. Thermal efficiency of a SCWR is an improvement by 10 — 15% compared to that of existing subcritical-water-cooled Nuclear Power Plants (NPPs). Use of water at supercritical temperatures and pressures enables application of a direct-cycle steam circuit that decreases capital costs. This steam cycle does not need steam generators, steam dryers and other related equipment.

Supercritical water used as a coolant enhances safety by eliminating the potential for dryout conditions. Dryout occurs in two-phase flow when vapour blankets a heated surface. Supercritical water remains in a single phase allowing for appropriate heat transfer at all temperatures. SCWRs are an appealing reactor concept offering increased thermal efficiency and safety at a reduced capital cost.

This paper provides a preliminary investigation of alternative fuel options suited for SCWR use. This study models a single fuel channel from a generic PT-type SCWR with an existing 43-element fuel-bundle design. Uranium nitride (UN), Mixed OXide (MOX) and Thoria

[1] CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited (AECL).

(ThO_2) nuclear fuels are analyzed with a uniform Axial Heat Flux Profile (AHFP). Bulk fluid, outer sheath and fuel centerline temperatures and Heat Transfer Coefficient (HTC) profiles are calculated for every millimetre along the heated length of the fuel channel. The design constraints are: 1) fuel centerline temperature must not exceed the industry accepted limit of 1850°C and 2) sheath temperature must not exceed the design limit of 850°C.

2. Fuel options

Alternative fuels to uranium dioxide (UO_2) are being investigated because of: 1) previous studies indicated that the fuel centreline temperature may exceed the industry accepted limit [3] (for details, see Figure 1); and 2) decrease dependency on uranium reserves.

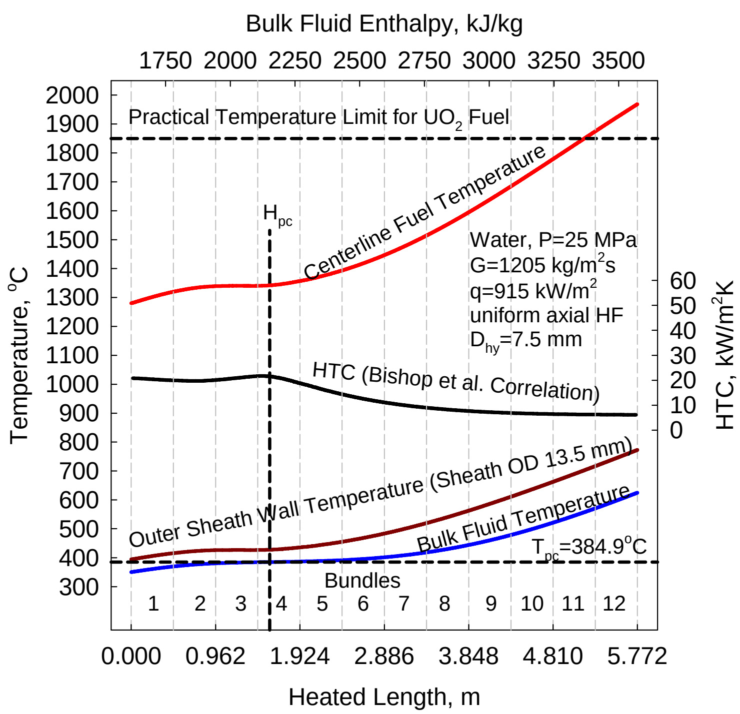

Figure 1. Temperature and HTC profiles along heated length of fuel channel calculated according to Bishop et al. correlation (fuel centerline temperature based on average thermal conductivity of UO_2) [3].

Since UO_2 might not be an acceptable option, other fuel choices should be investigated. Features of alternative nuclear fuels include improved thermophysical properties (refer to Table (2) [4, 5]) and resource availability. Uranium nitride is a uranium-based fuel being considered here due to its high thermal conductivity. However, a disadvantage of UN as a nuclear fuel is that it will react with nickel − a constituent of the sheath material [6]. To prevent this reaction the UN fuel may be stabilized with an addition of hafnium nitride (HfN) or thorium nitride (ThN).

Mixes oxide fuel is selected for its sustainability, since it is composed of irradiated fuel. There is an abundance of used fuel that may be reprocessed and reused, reducing the need to dispose of nuclear-fuel wastes. Some challenges with use of MOX include a shorter neutron life, lower delayed neutron fraction and irradiated-fuel temperature is higher than that for UO_2 [7].

Thoria has the highest melting point of the selected alternative fuels. This feature increases safety by reducing the likelihood of fission product release due to fuel melting. Thoria has fuelled both test and power reactors [8]. Although Thoria is fertile, it has the ability to be fertilized by fast, epithermal or thermal neutrons [9]. Used ThO_2 produces higher gamma-radiation fields than UO_2 [10]. However, irradiated ThO_2 is more stable and oxidation resistant than irradiated UO_2 [9].

Table 2. Major thermophysical properties of selected ceramic nuclear fuels at 0.1 MPa and 25ºC [4, 5 (only ThO_2)].

Property	Unit	Fuel			
		UO_2	MOX*	ThO_2	UN
Molar mass	kg/kmol	270.3	271.2	264	252
Theoretical density	kg/m^3	10,960	11,074	10,000	14,300
Melting temperature	ºC	2850	2750	3227	2850
Boiling temperature	ºC	3542	3538	> 4227	-
Heat of fusion	kJ/kg	259	285.3	-	-
Specific heat	kJ/kg·K	0.235	0.240	0.235	0.190
Thermal conductivity	W/m·K	8.68	7.82**	9.7	13.0
Coefficient of linear expansion	1/K	$9.75 \cdot 10^{-6}$	-	$8.9 \cdot 10^{-6}$	$7.52 \cdot 10^{-6}$

* MOX – Mixed Oxides $(U_{0.8}Pu_{0.2})O_2$, where 0.8 and 0.2 are the molar parts of UO_2 and PuO_2.
** at 95% density.

3. Fuel-bundle options

A fuel bundle chosen is based on the existing 43-element design [11]. The central element has an Outer Diameter (OD) of 20 mm and is assumed unheated. The remaining 42 elements have an OD of 11.5 mm. The hydraulic-equivalent diameter of the bundle is 7.83 mm. A fuel-bundle string consists of 12 bundles with a heated length of 5.772 m.

The sheath material chosen is Inconel-600, which has high mechanical strength and resistance to corrosion [12]. A sheath thermal conductivity as a function of temperature can be calculated according to Equation (1) [13]:

$$k = 14.2214 + 0.01625\,T, \tag{1}$$

where T is in Kelvin.

4. SuperCritical-water properties

A PseudoCritical (PC) point occurs above the critical pressure and at a temperature corresponding to the maximum value of specific heat at this particular pressure [14]. Within this region all thermophysical properties undergo dramatic changes (for details, see Figure 2). In general, thermal conductivity and density of water decrease along the heated length of the fuel channel. It should be noted that thermal conductivity has a small peak at the PC point. Dynamic viscosity has the minimum value right after the PC point.

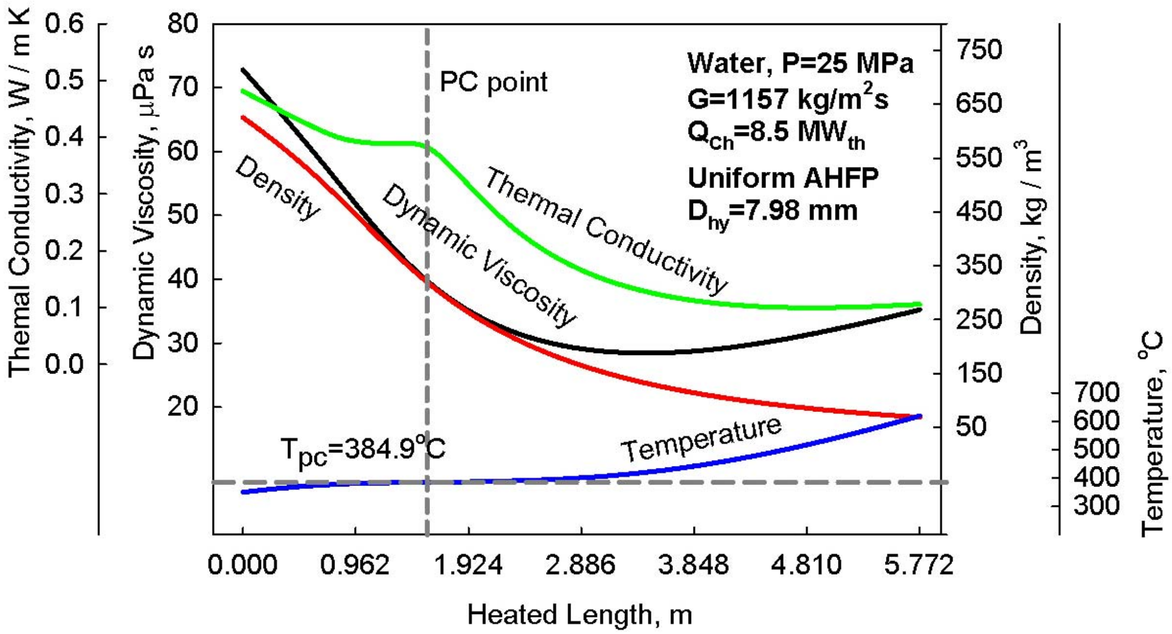

Figure 2. Profiles of thermal conductivity, density, dynamic viscosity and bulk-fluid temperature along heated length of fuel channel.

5. Method

A heat-transfer analysis was completed using MATrix LABoratory (MATLAB) programming and National Institute of Standards and Technology (NIST) REFPROP software [15]. Calculations were performed for each millimetre of the heated-channel length. Iterations were used in calculations of outer-sheath temperature and fuel thermal conductivities. The iterations halt when the assumed and calculated values differ one from another not more than by 0.5 K for sheath temperature and 0.05 W/m K for fuel thermal conductivity.

This thermalhydraulic modelling was conducted with the following assumptions: heat flux in the radial direction was uniform, fuel thermal conductivity varies only with temperature, a contact resistance between a fuel pellet and sheath is negligible due to a perfect contact, and a coolant pressure is a constant value of 25 MPa (a pressure drop is insignificant due to relatively low mass flux and coolant dynamic viscosity).

5.1 Bulk-fluid temperature

Our computation is initiated by determination of the bulk-fluid temperature (T_b), which is based on the heat-balance method. The bulk-fluid enthalpy along the channel was calculated based on a uniform AHFP of 967 kW/m^2 (corresponding to a channel power of 8.5 MW$_{th}$), an inlet temperature of 350 °C and pressure of 25 MPa (see Equation (2)).

$$h_x = \frac{\dot{Q}_{loc_{mm}}}{\dot{m}} + h_{x-1} \tag{2}$$

5.2 Outer-sheath temperature

The outer-sheath temperature ($T_{o,sh}$) was found through iterations based on Equation (3). The bulk-fluid temperature for each millimetre increment along the heated length was calculated through NIST based on the known pressure and calculated bulk-fluid enthalpy. Heat Transfer Coefficient (HTC) was calculated according to the Bishop et al. correlation [16] (see Equation (4)). The Bishop et al. correlation applicable within a pressure range from 22.8 to 27.6 MPa, bulk-fluid temperature range of 282 − 527°C, and heat flux range of 0.31 − 3.46 MW/m^2. All these conditions correspond to those of a generic SCWR.

$$T_{o,sh} = \frac{\dot{q}}{HTC} + T_b \tag{3}$$

$$\mathbf{Nu}_x = 0.0069\mathbf{Re}_x^{0.9}\overline{\mathbf{Pr}}_x^{0.66}\left(\frac{\rho_w}{\rho_b}\right)_x^{0.43}\left(1 + 2.4\frac{D}{x}\right) \tag{4}$$

The Bishop et al. correlation is considered as a preliminary and conservative approach in relation to bundle HTC calculations. This statement is based on the following: 1) Currently, there is no one correlation applicable for HTC calculations in power-reactor bundles and 2) The Bishop et al. correlation was obtained in bare vertical tubes, and it is known that HTC in fuel-bundle string will be enhanced by various appendages (endplates, bearing pads, spacers, etc.). The Bishop et al. correlation was altered by removing the last term in application for bundles (see Equation (5)). This term accounts for the entrance effect.

$$\mathbf{Nu}_x = 0.0069\mathbf{Re}_x^{0.9}\overline{\mathbf{Pr}}_x^{0.66}\left(\frac{\rho_{o,sh}}{\rho_b}\right)_x^{0.43} \tag{5}$$

The outer-sheath temperatures were compared against the design limit of 850°C.

5.3 Inner-sheath temperature

Inner-sheath temperatures ($T_{i,sh}$) were calculated based on the heat conduction through the cylindrical sheath wall [17] as shown in Equation (6), because the fuel sheath is a thin-walled tube.

$$\dot{Q}_{sh,x} = 2\pi k_{sh}\frac{T_{i,sh} - T_{o,sh}}{\ln\left(r_{o,sh}/r_{i,sh}\right)} \tag{6}$$

5.4 Fuel-centreline temperature

The second design constraint is for the fuel centreline temperature, which must be less than 1850°C. The fuel centreline temperature was found by iterations through Equation 7 [17]. Due to a significant temperature gradient through a fuel pellet, the radius is sectioned into 5 increments in which thermal conductivity was assumed to be a constant value to improve accuracy of the calculations.

$$T_{n-1} = \frac{\dot{e}_{gen,mm}\left[{r_{i,sh,n}}^2 - {r_{i,sh,n-1}}^2\right]}{4\,k_{fuel}} + T_n \tag{7}$$

6. Results

The output of the analysis is shown in Figures 3 — 5. All of the studied fuels satisfied the sheath-temperature design limit of 850°C. Consistently, the maximum sheath and fuel centerline temperatures occur at the end of the channel. Both UN and ThO_2 fuel centerline temperatures remain below the industry accepted limit of 1850°C. However, MOX fuel centerline temperature exceeds the industry accepted limit at the channel exit.

Therefore, the best nuclear fuel from the temperature design limits point of view is UN, because it has the minimum fuel centerline temperature.

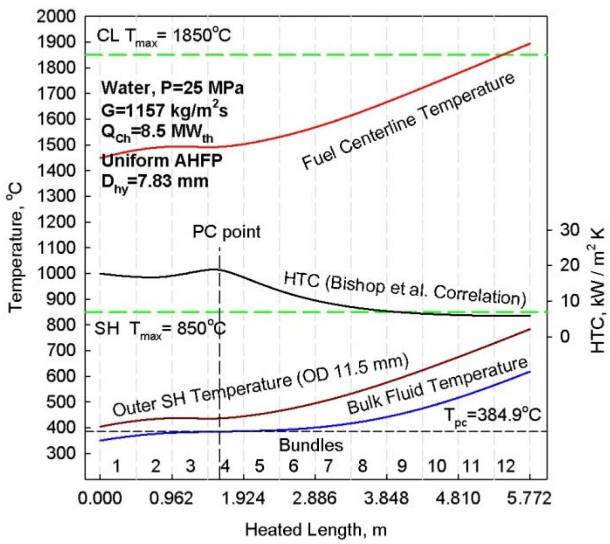

Figure 3. Temperature and HTC Profiles along Heated Length with Uniform AHFP and MOX fuel.

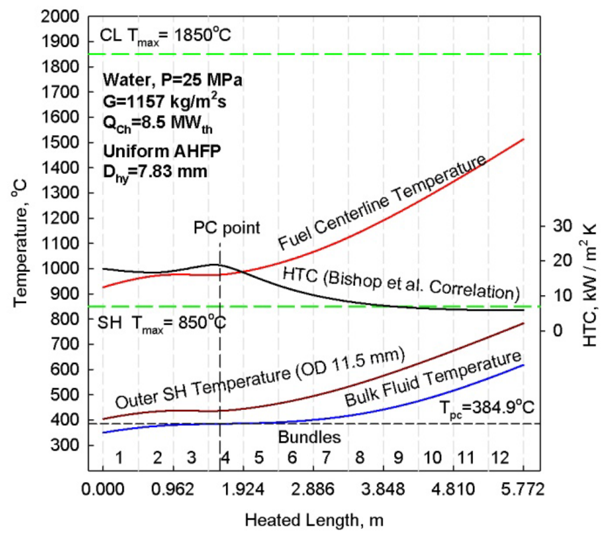

Figure 4. Temperature and HTC Profiles along Heated Length with Uniform AHFP and Thoria fuel.

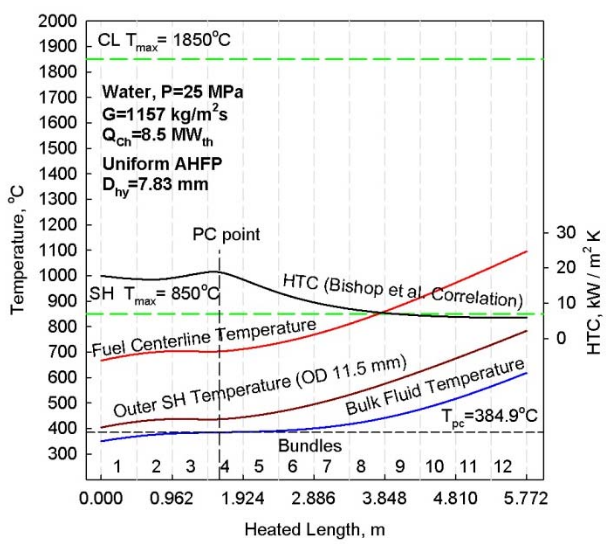

Figure 5. Temperature and HTC Profiles along Heated Length with Uniform AHFP and UN fuel.

7. Conclusions

1. All analyzed nuclear fuels: MOX, ThO_2 and UN, have sheath temperatures below the design limit of 850°C.
2. MOX fuel centreline temperature surpassed the industry accepted limit of 1850°C.
3. Thoria and UN are suitable for use in SCWRs as the fuel centreline temperatures remain below the design criterion.
4. UN is the optimal fuel choice for SCWRs, because of its very low fuel centreline temperatures.

8. Acknowledgement

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

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