THERMAL ASPECTS OF ALTERNATIVE FUELS FOR USE IN SUPERCRITICAL WATER-COOLED NUCLEAR REACTORS

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ABSTRACT – SuperCritical Water-cooled nuclear Reactors (SCWRs) are a Generation IV concept currently being developed. This reactor type uses light water coolant above its critical point. This paper presents a thermal-hydraulic analysis on a single channel within a Pressure Tube (PT) – type SCWR with a single reheat cycle. Uniform and non-uniform Axial Heat Flux Profiles (AHFPs) applied with a variety of alternative fuels (mixed oxide, thorium dioxide, uranium dicarbide, uranium nitride and uranium carbide). The results depict bulk-fluid, outer sheath and fuel centreline temperature profiles along with the Heat Transfer Coefficient (HTC) profile against sheath and fuel centreline temperature limits.

1. Introduction

This paper provides a potential configuration of a SuperCritical Water-cooled nuclear Reactor (SCWR). This Generation IV reactor is in its conceptual design phase. It is currently undergoing research and development activities; a prototype has yet to be built. The benefit of theoretical plant design is the ability to interchange and conduct analysis of various equipment combinations. This paper discusses the selection of nuclear fuels, sheath materials, sheath geometries, and steam cycles suitable for SCWRs.

There are two reactor core arrangements for SCWRs: 1) use of reactor vessel (similar to pressurized water reactors and boiling water reactors) and 2) use of multiple pressurized fuel channels (analogous to pressurized heavy water reactors). Canada and China are developing the second core option. This paper investigates the properties of a single channel of a multi-channelled reactor core.

A variety of nuclear fuels were analysed under SCWR normal operating conditions. A fuel was deemed suitable it the fuel centreline temperature remained below the industry accepted limit of 1850°C. Uranium dioxide (UO_2) was the primary choice due to its extensive and historical use as a nuclear fuel. However, previous studies by Pioro et al. [1] have concluded that fuel centreline temperature can potentially exceed the industry acceptable limit at the channel outlet. Non-traditional fuels such as: mixed oxide (MOX), Thoria (ThO_2), uranium dicarbide (UC_2), uranium nitride (UN) and uranium carbide (UC) are considered in this study.

Both uniform and non-uniform Axial Heat Flux Profiles (AHFPs) were applied. Non-uniform AHFPs such as: upstream cosine, cosine and downstream cosine were analyzed since they demonstrate channelized fuelling activities. The sheath materials selected were zirconium alloy, Inconel 600, Inconel 718 and stainless steel 304. A sheath material is declared acceptable when the outer sheath temperature is less than the design limit of 850°C. The bundle geometries were referenced from Leung et al. [2]. The fuel channel is approximately 6 metres long and contains 12 bundles.

A unique attribute of the SCWR is the use of SuperCritical Water (SCW) as the reactor coolant. Supercritical water is light water that is above its critical point. Some existing fossil fired plants utilize SCW as the working fluid for the turbine. Operational data and experience exists for conventional uses of SCW. The analysis and behaviour of SCW as reactor coolant for Nuclear Power Plants (NPPs) is a dynamic area of research. The development of SCW heat transfer correlations for flow over bundles is actively studied. Supercritical water HTC calculation is based on flow in bare vertical tubes. This provides a conservative estimate due the neglect of tuburlization of bundle appendages.

The potential thermodynamic cycle options for direct cycles SCWR are with no-reheat and single-reheat. Direct cycles are permitted due the increased coolant parameters (elevated temperature and pressure). Supercritical water does not require use of steam generators and steam dryers, etc. The no-reheat cycle SCW exits the channel and then enters the turbine. The single-reheat cycle is achieved by use of Steam-ReHeat (SRH) channels or by use of a Moisture-Separator-Reheater (MSR). The modelled SCWR channel utilizes a single-reheat cycle with a MSR. [3]

This study provides a thermalhydraulic analysis of a single PT-type fuel channel cooled with light water with an inlet temperature of 350°C, an outlet temperature of 625°C, a constant pressure of 25 MPa, mass flow rate of 4.4 kg/s and a channel power output of 8.5 MW_{th}. The bulk-fluid, outer sheath and fuel centreline temperature profiles along with the HTC profile against sheath and fuel centreline temperature limits are plotted for various fuels and AHFPs.

2. Nuclear Fuels

The preferential nuclear fuel is UO_2 because it has a wealth of operational data and defined thermal physical properties. Previously, UO_2 was analyzed by Pioro et al. [1] at SCWR conditions (see Figure 1). Near the channel exit, the fuel centreline temperature may exceed the industry accepted limit. These results promoted the investigation of alternative fuels.

The most influential thermophysical parameter is thermal conductivity. The thermal conductivity trends of the fuels analyzed are depicted in Figure 2. Thermal conductivities that increase with temperature increase are inherently safe as they are able to dissipate the heat faster to decrease fuel centreline temperature. However, the fuels with these desired trends, UC, UN and UC_2 (for details, see Figure 2), require extensive testing since they are not widely used. More information is required regarding fuel fabrication, thermophysical changes from neutron bombardment and properties of the irradiated fuel. There is abundant operational experience for the remaining nuclear fuels (ThO_2, MOX and UO_2) from use in test and power reactors [7 and 8].

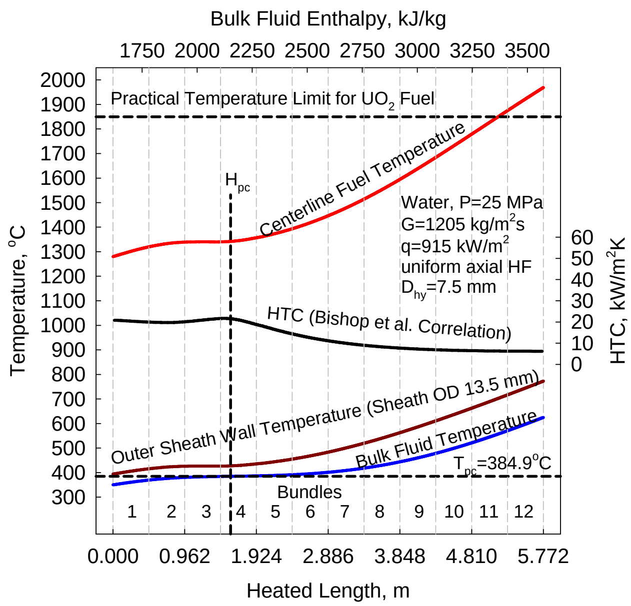


Figure 1 Temperature and HTC profiles along the heated length of a fuel channel calculated according to Bishop et al. correlation [1].

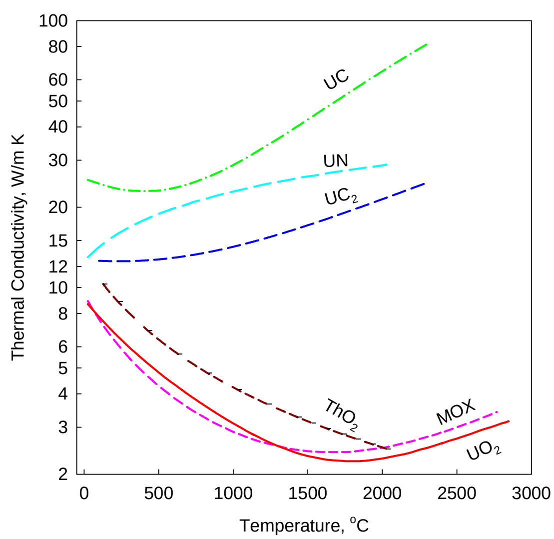


Figure 2 Thermal conductivities of selected nuclear fuels [4] ([5] – UC_2 and [6] – ThO_2).

The thermal conductivity of MOX is only slightly improved from that of UO_2 above 1500°C, however, MOX reduces nuclear waste disposal since it is formed from reprocessing of irradiated fuel. The disadvantages to MOX include: a shorter neutron life, lower delayed neutron fraction and higher irradiated fuel temperature compared to that of UO_2. [9]

Thoria is beneficial because there is an increase in thermal conductivity at all measured temperatures compared to UO_2 and it is a non-uranium based fuel. Thoria decreases dependency on uranium reserves and may operate on a closed fuel cycle with a high conversional ratio. The downside to thoria fuel is that a seed fuel is required to initiate the fertilization process and upon irradiation there are more gamma rays emitted then used UO_2. [10]

Uranium nitride is considered because of its high actinide density (increases probability of fission with fast neutrons) along with desired increased thermal conductivity and rising trend. The drawback of UN fuel use is that the decomposition products are reactive with nickel (a constituent Inconel sheath material) and requires a hafnium nitride (HfN) or thorium nitride (ThN) additive to become inert. [11]

The analyzed carbide based fuels have large thermal conductivities. Uranium carbide is superior over all studied fuel options since its thermal conduction is the highest. With both UC and UC_2 fuels there are factors beyond the scope of this paper in regards to porosity, density and manufacturing process, which must be accounted for when determining the feasibility of these substances as nuclear fuels for SCWRs. It has been identified for UC that for temperatures above 700°C and burn-ups greater than 5×10^{20} fissions/cm^3 swelling occurs from gas bubble formation. [12]

3. Sheath Materials

A sheath material is adequate for SCWR use if the temperature remains below the design limit defined at 850°C [13]. The primary choice for sheath material is zirconium alloy due to its high mechanical strength and excellent neutron transparency. Conversely, when zirconium alloy reaches 500°C, the corrosion rate increases significantly [14]. For this reason, zirconium alloy is unacceptable as a sheath material since the coolant temperature range for the channel inlet and outlet are 350°C and 625°C, respectively.

The alternative sheath material options are: Inconel-600, Inconel-718 as and stainless steel 304. The Inconels are non-magnetic nickel-based high-temperature alloys with high mechanical strength, hot and cold workability, and good corrosion resistance [15]. At high temperatures above 750°C, Inconel-718 experiences considerable decreases in its yield stress and tensile strength [16]. Stainless steel 304 was chosen for its corrosion resistance. Unfortunately, the structural strength was deemed too low and the wall thickness would require substantial increase. This would not be feasible as the thermal efficiency and neutron economy would be greatly decreased. Based on the review, Inconel-600 is the choice sheath material.

4. Sheath Geometries

Initially, four existing bundle geometries were chosen (one 37-element design and three 43-element designs) as described in Figure 3. The newest bundle design with the largest diameter centre rod described in Leung et al. was used for this analysis. The most recent bundle design was preferred due to the innovation of a large central pin filled with a neutron poison to increase safety in the event of a Loss Of Coolant Accident (LOCA). [2]

The central element has an Outer Diameter (OD) of 20 mm and is assumed to be unheated. The remaining 42 elements have an OD of 11.5 mm (for details refer to Figure 3). 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. [2]

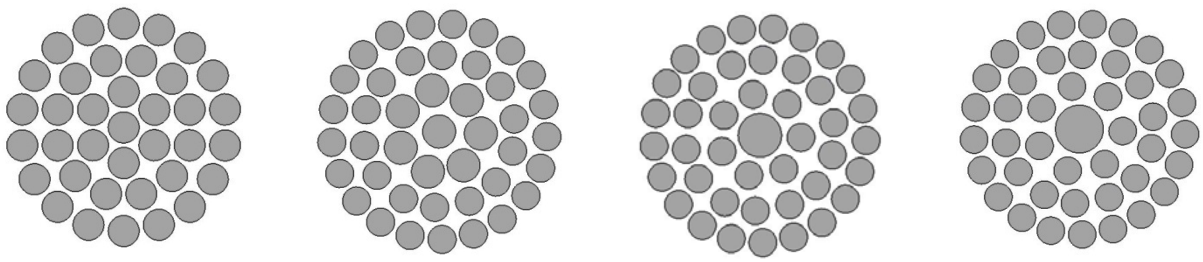


Figure 3 Comparison of fuel bundle geometries: (a) 37 element, (b) 43 element with centre &inner ring OD 13.5mm, (c) 43 element with centre OD 18mm, and (d) 43 element with centre OD 20mm [2]. (Courtesy of S. Mikhael)

5. Steam Cycles

Possible steam cycles are discussed to provide completeness of an SCWR plant layout. The reactor coolant is light SCW and is able to operate with a direct steam cycle to maximize thermal efficiency. As a result, indirect and dual cycles are not considered. The direct steam cycles has the option of steam no-reheat or single-reheat as the SCW cascades through the turbine series. It is assumed that a regenerative cycle is utilized regardless of the reheat arrangement. A regenerative cycle implies the feedwater temperature is increased by the use of steam extracted from various turbines exhausts. Furthermore, regeneration improves cycle efficiently and feedwater quality by removal of entrapped air and non-condensable gases. [3]

The flow path for a non-reheat direct steam cycle is once the SCW exits the fuel channel it enters and remains strictly within the turbine set (high pressure turbine, intermediate turbine and low pressure turbines). The disadvantages of the no-reheat selection is decreased thermal efficiency (by approximately 1 — 2%) and high moisture content in the low pressure turbine exhaust. Conversely, the no-reheat cycle reduces complexity of the reactor core design without the need for specialized reheat channels. [3]

The single-reheat steam scheme is permitted with the use of SRH channels or by a MSR. For the SRH regime, after the high pressure turbine steam re-enters the reactor within channels devoted to reheating steam. Alternatively, steam reheat can be accomplished outside the reactor with a MSR as shown in Figure 4. The MSR is located in between the intermediate and low pressure turbines and is heated via intermediate turbine exhaust. The double-reheat cycle offers the highest thermal efficiency but the design and capital costs increase substantially. The benefits to single reheat steam cycles are an increased thermal efficiency and reduced cost since SCW single-reheat via MSR are currently in use with some fossil plants. As aforementioned, reheat channels would require further research and development. In this analysis, a direct single-reheat steam cycle with MSR is applied. [3]

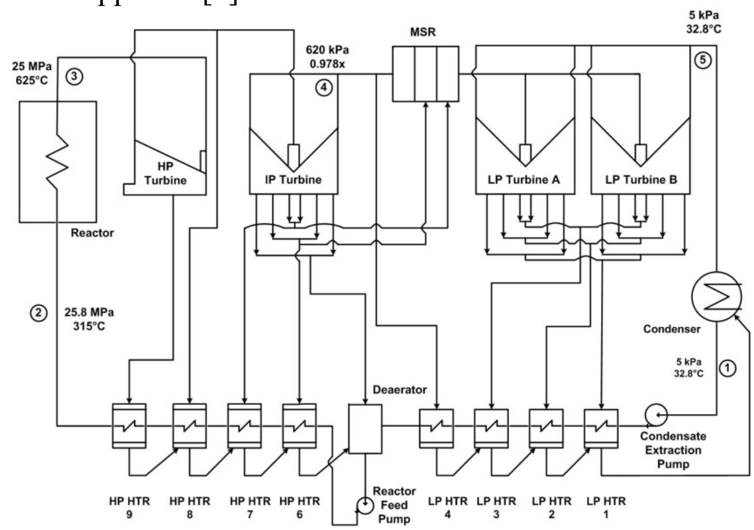


Figure 4 SCWR Single-Reheat Cycle with MSR [17]. (Courtesy of M. Naidin)

6. Properties of Supercritical Water

The pseudocritical point which is characterized with T_{pc}, is a point at a pressure above the critical pressure and at a temperature corresponding to the maximum value of specific heat for this particular pressure [17]. For water at 25 MPa the pseudocritical temperature is 384.9°C. There are significant changes in all thermophysical properties within the pseudocritical region (+/-25°C) (see Figure 5).

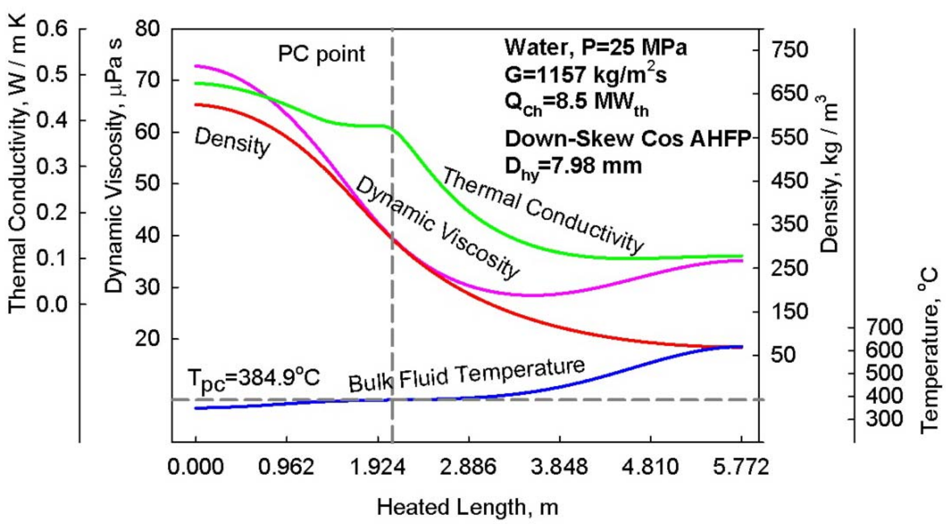


Figure 5 Profiles of thermal conductivity, density, dynamic viscosity and bulk-fluid temperature along heated length of fuel channel for downstreamed-skewed AHFP [19].

7. Calculations

The presented heat-transfer analysis proceeds as follows: 1) Calculation of the heat-transfer rate per each millimeter based on the selected AHFP, 2) Calculation of the bulk-fluid-temperature profile based on the heat balance method, 3) Calculation of the HTC profile with use of a modifed Bishop et al. correlation [20], 4) Calculation of the inner-sheath-temperature profile, and 5) Calculation of the fuel-centreline-temperature profile. A perfect contact was assumed between the inner-sheath surface and the outer surface of a fuel pellet.

The abovementioned calculations for HTC and conduction through sheath and fuel pellet required iterations. The criterion used in the temperature and thermal conductivity iterations was ±0.5 K and ±0.05 W/m K respectively. The MATrix LABoratory (MATLAB) software was used to develop the computational code.

This thermophysical investigation was performed with the following assumptions: heat flux in the radial direction was uniform, the bundle-string length to be equal to the heated channel length (end-plates and end-caps of each bundle were not taken into consideration), the fuel thermal conductivity varies only with temperature, the contact resistance between a fuel pellet and sheath is negligible due to a perfect contact, and the coolant pressure is constant along the channel.
The generic PT-type SCWR parameters are: the channel power is 8.5 MW_{th}, the inlet temperature − 350°C, the outlet temperature − 625°C the pressure − 25 MPa and the coolant mass-flow rate − 4.4 kg/s.

7.1 Axial heat flux profiles

Both uniform and non-uniform AHFPs are analyzed (see Figure 6). The uniform AHFP was 967kW/m^2 ($\dot{q}_{ave}$). The non-uniform AHFPs were fitted with 6th-degree polynomials. The cosine and upstream-cosine AHFPs were taken from Leung et al. [2]. The downstream-skewed cosine AHFP was produced from a mirror image of the upstream-skewed cosine.

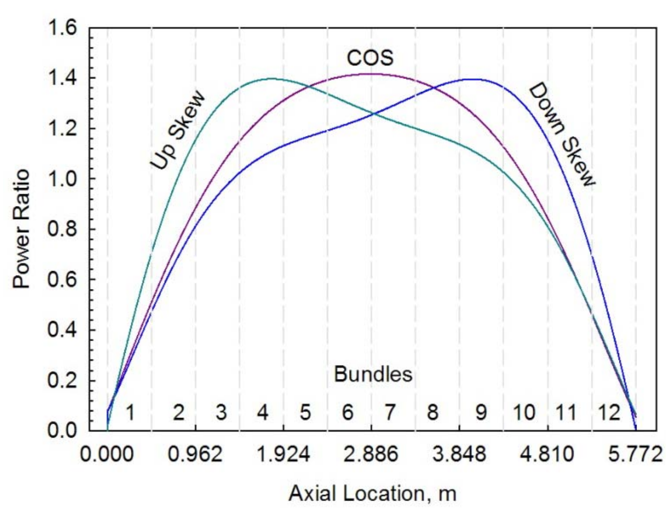


Figure 6 Non-uniform AHFPs [2].

7.2 Bulk fluid temperature

The initial step in this heat-transfer analysis is to determine the bulk-fluid temperature profile along the heated length, which was obtained by the heat-balance method (see Equation (1)). The inlet bulk-fluid enthalpy was obtained based on an inlet temperature of 350°C and constant pressure of 25 MPa.

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

Based on h_x and constant pressure of 25 MPa the bulk-fluid temperature profile was determined by National Institute of Standards and Technology REference Fluid thermodynamic and transport Properties (NIST REFPROP) [19].

7.3 Outer-sheath temperature

Outer-sheath temperature was obtained based on the corresponding bulk-fluid temperature (T_b), i.e., in the same axial position, and HTC. The HTC was calculated according to the Bishop et al. correlation (Equation (2)). The Bishop et al. correlation is suitable for a pressure range from 22.8 to 27.6 MPa, bulk-fluid temperature is between 282 and 527°C, and heat flux is between 0.31and 3.46 MW/m² which corresponds to the generic SCWR operating conditions. [20]

$$\mathbf{Nu}_x = \mathbf{0.0069Re}_x^{0.9}\overline{\mathbf{Pr}_x}^{0.66}\left(\frac{\rho_w}{\rho_b}\right)_x^{0.43}\left(1 + 2.4\frac{D}{x}\right) \tag{2}$$

Use of Bishop et al. correlation is a conservative approach because the correlation was obtained in bare tubes but the HTC due to bundles will be enhanced with flow turbulziation from various appendages (endplates, bearing pads, spacers, etc.). Also, the original Bishop et al. correlation was modified to better suit bundle-flow conditions. Actually, the last term in Equation (2) which is responsible for the inlet effect in bare tubes, was eliminated (see Equation (3)).

$$\mathbf{Nu}_x = \mathbf{0.0069Re}_x^{0.9}\overline{\mathbf{Pr}_x}^{0.66}\left(\frac{\rho_{o,sh}}{\rho_b}\right)_x^{0.43} \tag{3}$$

7.4 Inner-sheath temperature

To determine the inner-sheath temperatures the outer-sheath temperatures must be known. The outer-sheath temperature is calculated through Equation (4).

$$\boldsymbol{T_{o,sh} = \frac{\dot{q}}{HTC} + T_b} \tag{4}$$

The outer-sheath temperature was evaluated against the design limit of 850°C. Inner-sheath temperatures were then calculated based on the heat conduction through the cylindrical sheath wall [21] as shown in Equation (5).

$$\dot{Q}_{sh,x} = 2\pi k_{sh} \frac{T_{i,sh}-T_{o,sh}}{\ln\left(r_{o,sh}/r_{i,sh}\right)} \tag{5}$$

7.5 Fuel centreline temperature

The fuel centreline temperature was found through Equation (6) [21].

$$T_{n-1} = \frac{\dot{e}_{gen,mm}\left[{r_{i,sh,n}}^2 - {r_{i,sh,n-1}}^2\right]}{4\,k_{fuel}} + T_n \tag{6}$$

A significant temperature gradient through the fuel pellet was observed. To improve accuracy of the calculations the pellet radius was sectioned into 5 increments. Within each increment the thermal conductivity was assumed to be a constant value. The fuel centreline temperatures were compared to the industry accepted limit of 1850°C.

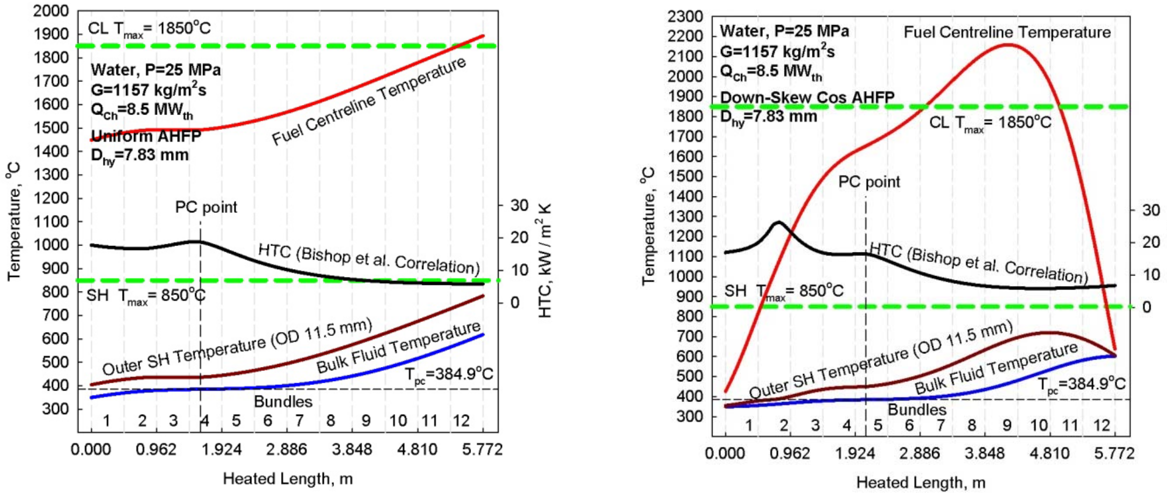
8. Results

Each nuclear fuel was analyzed within a 43 element bundle with a centre rod OD of 20mm composed of Inconel 600 against each of the four AHFPs. The presented results were limited to two cases per fuel type depicted in Figure 7 — 11. The lowest and highest fuel centreline temperature profiles for each fuel are show in the proceeding figures. In all investigated cases the sheath temperature remained below the design limit of 850°C.

The highest fuel centreline temperatures were calculated using MOX fuel with down-stream skewed cosine AHFP, as shown in Figure 7(b). This combination of MOX fuel and AHFP exceeds the fuel centreline industry accepted of 1850°C. Mixed oxide fuel may still be used as SCWR fuel, however, the fuel bundle design requires modification. Fuel centreline temperatures may be decreased by using smaller diameter pellets, hollowing the pellets or decreasing channel output.

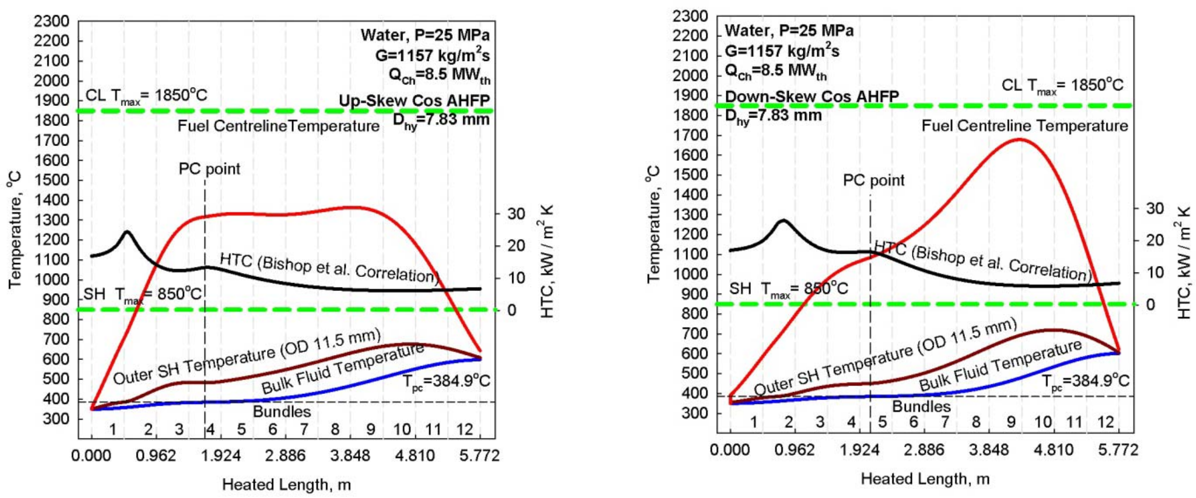
The lowest fuel centreline temperatures occur with UC fuel with upstream-skewed cosine AHFP (see Figure 11(a)). Within this scenario the UC fuel centreline temperatures does not even exceed the sheath temperature constraint. Although, UC largely increases the margin of safety by having a minimum fuel centreline temperature there is limited data on operational performance, fabrication methods and irradiation behaviour.

The remaining fuels (ThO_2, UN and UC_2) are shown in Figure 8 — 10 are deemed acceptable to be applied for SCWR use by this thermal-hydraulic analysis as both sheath and fuel centreline temperature constraints are respected. In terms of fuel supply, MOX is the most sustainable option since it is formed from irradiated UO_2 and Thoria is unique because it is a non-uranium based fuel.



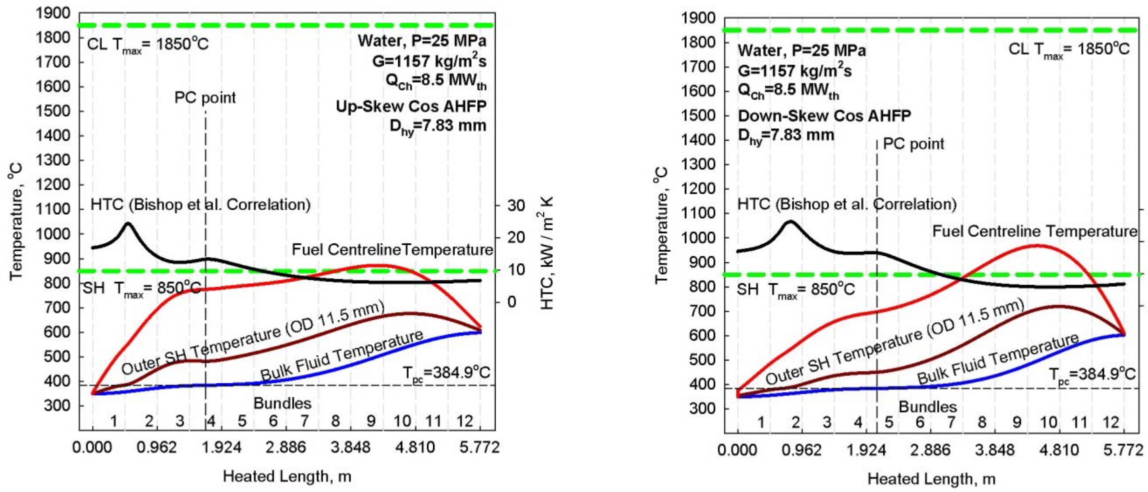
(a) **(b)**

Figure 7 Temperature and HTC profiles for MOX fuel along heated length with: (a) uniform AHFP and (b) downstream-skewed AHFP. [22]



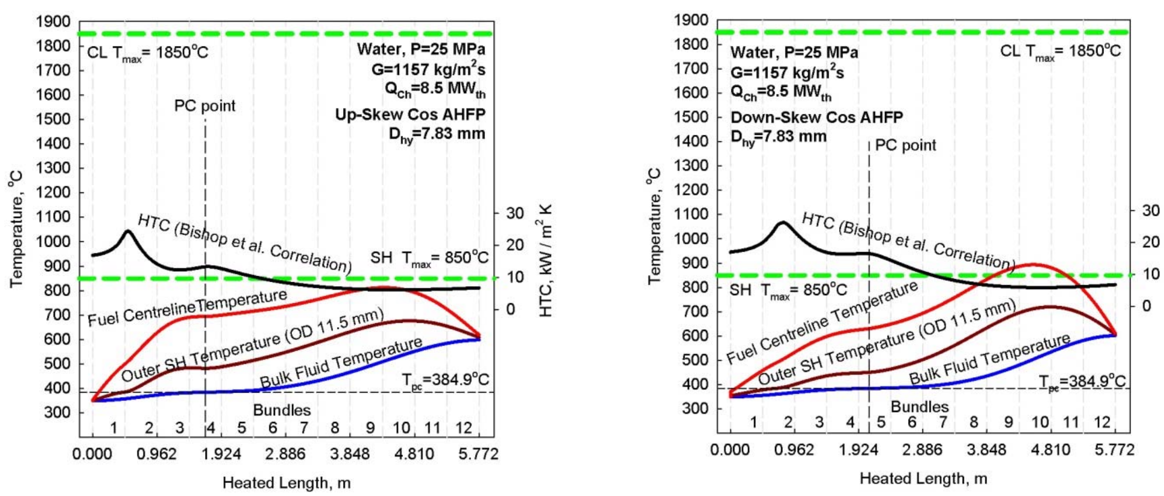
(a) **(b)**

Figure 8 Temperature and HTC profiles for ThO_2 fuel along heated length with: (a) upstream-skewed AHFP and (b) downstream-skewed AHFP. [23]



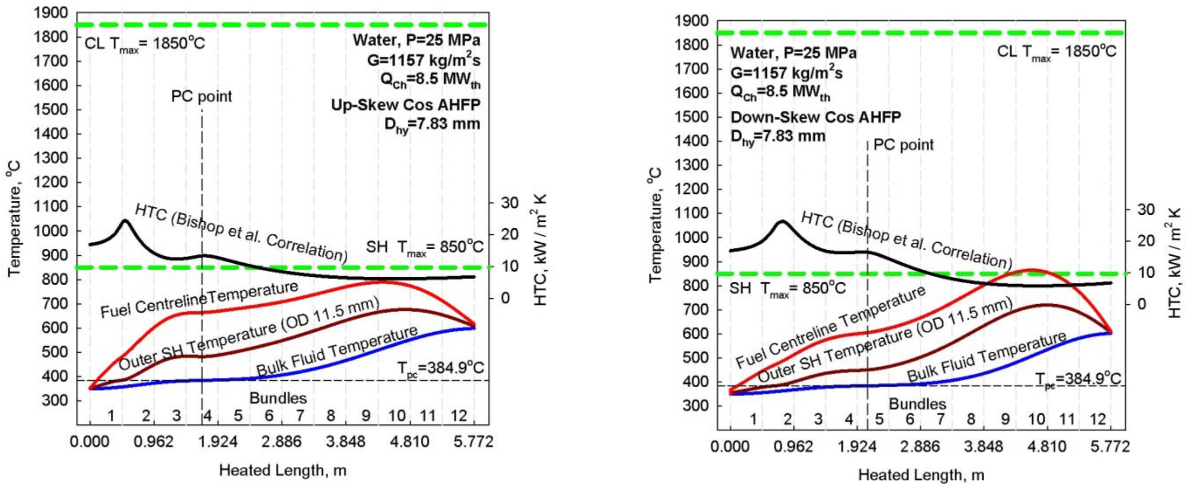
(a) (b)

Figure 9 Temperature and HTC profiles for UC_2 fuel along heated length with: (a) upstream-skewed AHFP and (b) downstream-skewed AHFP. [24]



(a) (b)

Figure 10 Temperature and HTC profiles for UN fuel along heated length with: (a) upstream-skewed AHFP [25] and (b) downstream-skewed AHFP.



(a) (b)

Figure 11 Temperature and HTC profiles for UC fuel along heated length with: (a) upstream-skewed AHFP and (b) downstream-skewed AHFP. [24]

9. Conclusions

For all investigated cases the sheath design temperature limit of 850°C was not exceeded. The fuel centreline temperature industry limit was surpassed by only MOX fuel. For MOX to be suitable for SCWR use fuel bundle design changes are needed or channel power needs to be reduced. Thoria, uranium dicarbide, uranium nitride and uranium carbide are feasible SCWR nuclear fuels as their fuel centreline temperatures remain below the industry accepted limit of 1850°C. Thoria offers an additional advantage because it decreases dependency on uranium reserves.

10. Acknowledgements

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Nomenclatures

$\bar{c}_p$:	average specific heat, J/kg K $\left(\frac{h_{o,sh}-h_b}{T_{o,sh}-T_b}\right)$
D:	diameter, m
$\dot{e}_{gen}$:	volumetric heat flux, W/m^3
G:	mass flux, kg/m^{2}s, $\left(\frac{\dot{m}}{A_{flow}}\right)$
h:	enthalpy, J/kg
HTC:	Heat Transfer Coefficient, W/m^{2}K
k:	thermal conductivity, W/m K
$\dot{m}$:	mass-flow rate, kg/s
Nu	: Nusselt number, $\left(\frac{HTC\, D_{hy}}{k}\right)$

P : pressure, Pa or porosity (volume fractions)

$\overline{\mathbf{Pr}}$: average Prandtl number, $\left(\frac{\overline{c_p}\,\mu}{k}\right)$

$\dot{Q}$: heat transfer rate, W

$\dot{q}$: heat flux, W/m^2, $\left(\frac{\dot{Q}}{A_h}\right)$

r: radius, m

Re Reynolds number, $\left(\frac{G D_{hy}}{\mu}\right)$

T: temperature, ℃

Subscripts

b: bulk

Ch: channel

fuel: fuel

h: heated

hy: hydraulic equivalent

i: inner

mm: per millimetre increment

n: radial location within fuel pellet

o: outer

pc: pseudocritical point

sh: sheath

w: wall

x: axial location along heated length

Acronyms

AHFP: Axial Heat Flux Profile

CL: CenterLine (related to fuel pellet)

HTC: Heat Transfer Coefficient

OD: Outer Diameter

PC: PseudoCritical (point)

SH: sheath (fuel)