# THERMAL ASPECTS OF MIXED OXIDE FUEL IN APPLICATION TO SUPERCRITICAL WATER-COOLED NUCLEAR REACTORS

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## Abstract

SuperCritical Water-cooled nuclear Reactors (SCWRs) are a renewed technology being developed as one of the Generation IV reactor concepts. This reactor type uses a light water coolant at temperatures and pressures above its critical point. These elevated operating conditions will improve Nuclear Power Plant (NPP) thermal efficiencies by 10 – 15% compared to those of current NPPs. Also, SCWRs will have the ability to utilize a direct cycle, thus decreasing NPP capital and operational costs.

The SCWR core has 2 configurations: 1) Pressure Vessel (PV) -type enclosing a fuel assembly and 2) Pressure Tube (PT) -type consisting of individual pressurized channels containing fuel bundles. Canada and Russia are developing PT-type SCWRs. In particular, the Canadian SCWR reactor has an output of 1200 $MW_{el}$ and will operate at a pressure of 25 MPa with inlet and outlet fuel-channel temperatures of 350 and 625°C, respectively.

These extreme operating conditions require alternative fuels and materials to be investigated. Current CANadian Deuterium Uranium (CANDU) nuclear reactor fuel-channel design is based on the use of uranium dioxide ($UO_2$) fuel; zirconium alloy sheath (clad) bundle, pressure and calandria tubes. Alternative fuels should be considered to supplement depleting world uranium reserves.

This paper studies general thermal aspects of using Mixed OXide (MOX) fuel in an Inconel-600 sheath in a generic PT-type SCWR. The bulk fluid, sheath and fuel centerline temperatures along with the Heat Transfer Coefficient (HTC) profiles were calculated at uniform and non-uniform Axial Heat Flux Profiles (AHFPs).

## 1. Introduction

Dating back to the 1950s and 1960s, SuperCritical Water (SCW) was proposed to be used as a coolant in nuclear reactors. The United States and Russia led this research. However, this interesting and promising development was abandoned at the end of the 1960s – early 1970s. After a 30-year break, the idea of developing nuclear reactors cooled with SCW became attractive once again. At the moment, there are various Generation IV SuperCritical Water-cooled nuclear Reactor (SCWR) concepts under development worldwide. This interest is regained due to improved economic features such as increased thermal efficiency and simplification of reactors systems. SCW Nuclear Power Plants (NPPs) will operate at higher parameters than conventional NPPs.

SCWRs will operate at supercritical pressures and temperatures (i.e., pressures of about 25 MPa and outlet-channel temperatures up to 625°C).

Mixed OXide (MOX) was selected as a candidate fuel, because uranium resources are becoming depleted. However, MOX has some disadvantages, in particularly, shorter neutron life, lower delayed neutron fraction and irradiated-fuel temperatures higher than that of $UO_2$ [2]. At lower temperatures MOX fuel has a lower thermal conductivity than $UO_2$, but at higher temperatures this trend is reversed(see Figure 1).

A MOX-fuelled reactor is proliferation compliant since it has the ability to dispose of plutonium produced from weapons programs. Additionally, MOX fuel enables recycling of plutonium from Light Water Reactor (LWR) fuel. This reprocessing reduces the stockpiling of plutonium in waste facilities.

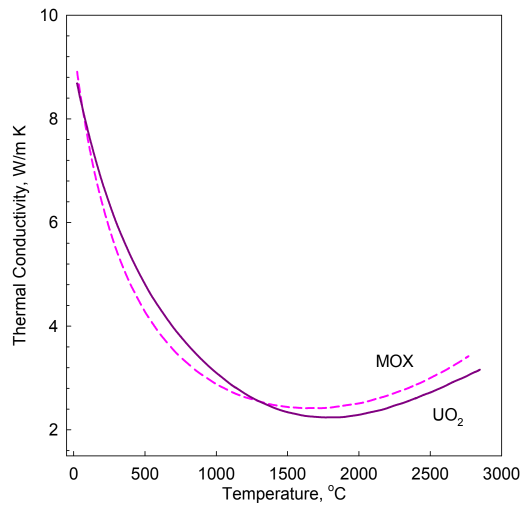


**Figure 1. Comparison of thermal conductivities of $UO_2$ and MOX fuels [1].**

The Pressure Tube (PT)-type core design supports MOX fuel usages. Studies by Boczar et al. [3] consider the use of MOX fuel provided by the recycling of LWR fuel in a CANadian Deuterium Uranium (CANDU) nuclear reactor. The benefits include: fabrication and irradiation tests conducted at Chalk River Laboratories (CRL), the high neutron economy does not require manipulation of the used LWR fissile content and the ability to accommodate a full core of only MOX fuel [3]. This study demonstrates the practical use of irradiated LWR fuel as a MOX supply for PT-type SCWRs.

A SCWR has not yet been constructed; it is currently in its early design phase. Several research activities have been conducted to determine suitable configurations of fuel and material for SCW conditions. This paper describes the thermal aspects of MOX fuel within an Inconel-600 bundle for a generic PT-type SCWR while enforcing the following temperature constraints: 1) the industry accepted limit of 1850°C for fuel centerline temperature and 2) the design limit of 850°C for the sheath temperature.

## 2. Mixed oxide fuel

MOX nomenclature is utilized to describe a heterogeneous fuel consisting of uranium-plutonium oxides. The standard MOX stoichiometric composition is by molar fractions ratio of 0.8 uranium dioxide ($UO_2$) compared to 0.2 plutonium dioxide ($PuO_2$). This composition is described in the form of $(U_{0.8}\,Pu_{0.2})O_2$ [1].

As early as the 1950s, MOX-fuel fabrication activities have been conducted in Belgium and in USA [4]. A decade later, France, Germany, Japan, Russia and UK became interested; India also supported research into various MOX developments. The initial testing of MOX was in the 1960s [5]. In the 1980s, MOX became used commercially.

Currently, MOX is being used extensively in Europe and is intended to be used in Japan. In Belgium, Switzerland, Germany and France, 40 reactors are licensed to use MOX fuel. Over 30 other countries are in the process of becoming licensed to operate with MOX fuel. Today, France intends to have all of its 900-$MW_{el}$ series reactors operating with at least one third of MOX fuel. Japan has prospects to use MOX in one third of its reactors in the near future and is going to start up a 1383-$MW_{el}$ reactor at the Ohma plant with loading of MOX by late 2014 [4].

### 2.1 Thermophysical properties

Thermophysical properties of MOX are similar to $UO_2$, see Table 1. The only advantage is that at higher temperatures the thermal conductivity of MOX is slightly higher than that of $UO_2$ (see Figure 1).

**Table 1. Major thermophysical properties of selected ceramic nuclear fuels at 0.1 MPa and 25ºC [1].**

| Property | Unit | Fuel | |
|---|---|---|---|
| | | $UO_2$ | MOX |
| **Molar mass** | kg/kmol | 270.3 | 271.2 |
| **Theoretical density** | kg/m$^3$ | 10,960 | 11,074 |
| **Melting temperature** | ºC | 2850 | 2750 |
| **Boiling temperature** | ºC | 3542 | 3538 |
| **Heat of fusion** | kJ/kg | 259 | 285.3 |
| **Specific heat** | kJ/kg·K | 0.235 | 0.240 |
| **Thermal conductivity** | W/m·K | 8.68 | 7.82* |
| **Coefficient of linear expansion** | 1/K | $9.75\cdot10^{-6}$ | - |

* at 95% density.

Various thermophysical properties of MOX are listed by Kirillov et al. [1]. Some of these values are shown in Figure 2. Thermal conductivity reaches its minimum values at 1500 and 2000°C until a plateau is reached above ~2000°C. After this point, the thermal conductivity increases to about 4 W/m K. The integral thermal conductivity can be used to describe gas release from the fuel [6]. This parameter increases as temperature rises. At lower thermal conductivities, the integral thermal conductivity values are higher due to increased gas production.

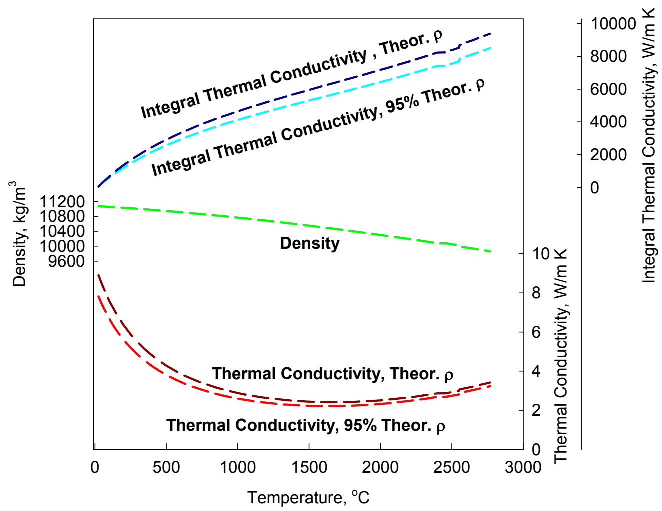


**Figure 2. Thermophysical properties of MOX fuel of stoichiometric composition $(U_{0.8}\,Pu_{0.2})O_2$ in solid state [1].**

## 2.2 Inventory

There are four plants producing commercial quantities of MOX fuel. The MOX fuel fabrication capacities are listed in Table 2. Two of them are located in France, one in Belgium, and one was commissioned in the UK in 2001 [5].

**Table 2. World MOX fuel fabrication capacities (tonnes per year) for LWR [5].**

| Country | Year | | |
|---|---|---|---|
| | **2006** | **2008** | **2012** |
| **France** | 145 | 195 | 195 |
| **Japan** | 0 | 0 | 130 |
| **UK** | 40 | 40 | 40 + |
| **Total for LWR** | 185 | 235 | 445 |

Presently, the output from MOX reprocessing plants is greater than the amount of plutonium required. This creates a reserve of plutonium. This inventory is expected to exceed 250 tonnes until MOX usage increases [5].

## 3. Fuel bundle design

In the previous study conducted by Pioro et al. [7], $UO_2$ fuel in a zirconium sheath was investigated at similar thermalhydraulics conditions and compared to the same temperature constraints. At some conditions, $UO_2$ fuel centerline temperature surpassed the industry accepted limit of 1850°C. Zirconium alloy is commonly used as a sheath material in existing NPPs. However, at SCW conditions the sheath temperature will be significantly above current values [8]. Here it should be noted that zirconium-based alloys corrosion rates are much higher at temperatures above 500°C. Therefore, this requires other sheath materials to be investigated [9].

### 3.1 Sheath Design

The current analysis is based on the existing fuel-bundle geometry of 43 elements (fuel rods) [10]. The center element is assumed to be filled with neutron poison and is considered as a non-heating element. Therefore, only 42 heated elements were considered in these thermal calculations. The appropriate sheath thickness was determined based on a particular collapse pressure. Inconel-600 was selected, because nickel-based alloys are one of the materials being considered for SCWRs [8]. Nickel-based alloys have high mechanical strength, high temperature resistance and low corrosion rates at high temperatures [11].

### 3.2 Considered sheath materials

Inconel-600, Inconel-718 and stainless steel SS-304 were considered as SCWR sheath materials. Inconel-600 and Inconel-718 are nonmagnetic nickel-based high-temperature alloys with high mechanical strength, hot and cold workability, and high resistance to corrosion [11]. At temperatures above 750°C Inconel-718 exhibits a significant decrease in its yield stress and tensile strength [12]. SS-304 has excellent corrosion resistance. However, at 850°C its structural strength decreases significantly, requiring an increase in the wall thickness [13]. Having a thick sheath would increase static loading, decrease thermal efficiency and decrease neutron economy. Therefore, Inconel-600 was chosen as the sheath material in the current thermal analysis.

## 4. Calculations

The sequence of calculations are: 1) Determination of the bulk-fluid temperature, 2) Calculation of the Heat Transfer Coefficient (HTC), 4) Calculation of and outer-sheath and inner-sheath temperatures and 5) Calculation of fuel centerline temperature.

An iterative process is used to solve for unknown variables in dynamic environments based on an initial estimation. The initial estimate for the solution is stated and then refined through iteration until the "stopping" criteria is reached [14]. The temperatures of the coolant at the sheath wall, the thermal conductivities of the sheath and fuel centerline temperatures are calculated using iterative methods. Their values are dependent on temperature changes through the media. The stipulated "stopping" criteria for the iterations performed for the determination of temperature and thermal conductivity was a difference of 0.5 K and 0.5 W/m K, respectively.

In this thermal analysis, operational parameters were taken from a generic PT-type SCWR. The power output per channel is 8.5 $MW_{th}$ with a constant coolant mass-flow rate of 4.4 kg/s. The fuel string consists of 12 bundles with a total heated channel length of 5.772 m. The calculations consider the fuel-rod length to be equal to the heated channel length; the end-plates and the end-caps of the bundle are not taken into consideration. The pressure along the channel was assumed to be constant 25 MPa. Uniform axial and radial heat fluxes were applied in one-dimensional conduction in the radial direction assuming steady state conditions. Gap/contact thermal resistances (between the fuel and sheath) are neglected due to a perfect contact.

## 4.1 Axial Heat Flux Profiles

Both uniform and non-uniform AHFPs were analyzed (see Figure 3). The uniform AHFP was 967 $kW/m^2$.

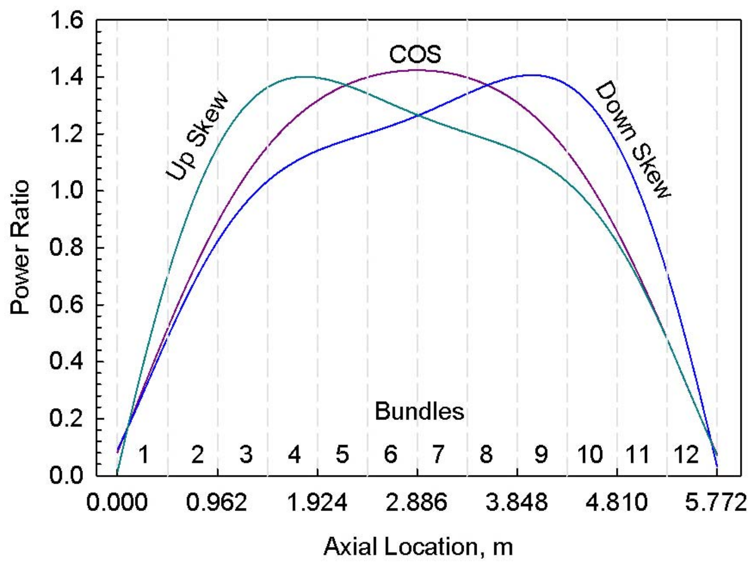


**Figure 3. Non-uniform AHFPs [10].**

## 4.2 Bulk-Fluid Temperature

The first step performed in the heat-transfer analysis is the determination of bulk-fluid temperature using the heat-balance method (Equation (1)).

$$h_x = \frac{\dot{Q}_{loc_{mm}}}{\dot{m}} + h_{x-1} \quad (1)$$

## 4.3 Heat transfer coefficient and outer-sheath temperature

Finding the outer-sheath temperature requires iterations for HTC (Nusselt number). The Bishop et al. correlation [15] for heat transfer in bare vertical tubes with SCW is suitable for the following conditions: pressure is between 22.8 and 27.6 MPa, bulk-fluid temperature is between 282 and 527°C, and heat flux is between 0.31 and 3.46 MW/m². These conditions suit those of the generic PT-type SCWR. Additionally, the last term in the Bishop et al. correlation (1 + 2.4 *D*/*x*), which represents the entrance effect, is not used in the calculations (see Equation (2)). The use of the Bishop et al. correlation is a conservative approach, since this correlation was obtained in bare tubes, and the HTC in fuel bundles will be enhanced with flow turbulization from various appendages (endplates, bearing pads, spacers, etc).

$$\mathbf{Nu}_x = 0.0069\mathbf{Re}_x^{0.9}\overline{\mathbf{Pr}_x}^{0.66}\left(\frac{\rho_{o,sh}}{\rho_b}\right)_x^{0.43} \tag{2}$$

The average Prandtl number ($\overline{\mathbf{Pr}}$) is used, because of significant variations in the bulk-fluid temperature in a cross section at high heat flux. This $\overline{\mathbf{Pr}}$ uses the average specific-heat capacity ($\overline{c_{\mathrm{p}}}$). These average values are used in HTC calculations instead of the regular Prandtl number and regular specific-heat capacity (see Figures 4 and 5).

As it was mentioned previously, the calculation of HTC through the Bishop correlation does require iterations, because the outer-sheath temperature $\left(T_{\mathrm{o,sh}}\right)$ and the bulk-fluid density at the outer-sheath temperature $\left(\rho_{\mathrm{o,sh}}\right)$ are initially unknown. The starting point for the iterations was assumed to be $T_{\mathrm{o,sh}} = T_{\mathrm{b}} + 25$ K. Equation (3) is used to find the outer-sheath temperature.

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

The $T_{\mathrm{o,sh}}$ profile along the heated length is calculated based on the iterations mentioned above, and the maximum value for $T_{\mathrm{o,sh}}$ is compared to the design limit of 850°C.

## 4.4 Inner-sheath temperature

The inner-sheath temperature is calculated by iterations using Equations (6) and (7). Equation (6) describes heat conduction through a thinned-walled cylinder [14]. Equation (7) is the correlation used for the thermal conductivity of Inconel-600 [16].

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

$$k = 14.2214 + 0.01625\,T \tag{7}$$

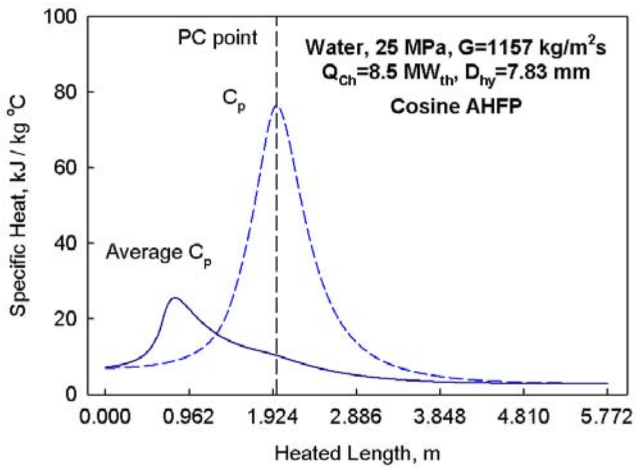


**Figure 4. Bulk-fluid specific-heat capacity and average specific-heat capacity along heated length with cosine AHFP.**

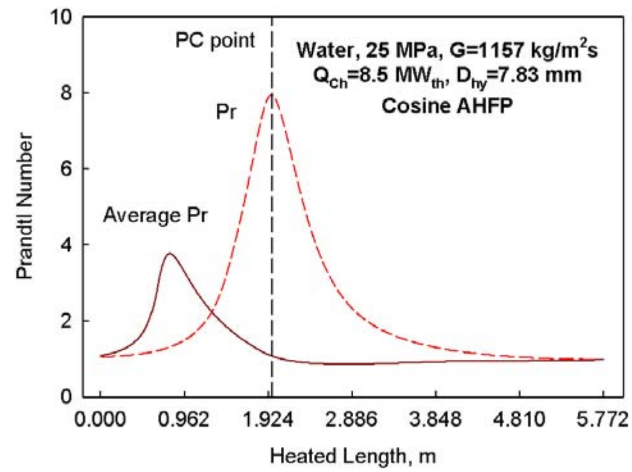


**Figure 5. Bulk-fluid Prandtl number and average Prandtl number along heated length with cosine AHFP.**

### 4.5 Fuel centreline temperature

The fuel centerline temperature must not exceed industry accepted limit of 1850°C. Equation (8) is used to find the fuel centerline temperature [14]. Equation (9) is the thermal conductivity correlation of MOX fuel based on tabulated data [1]. Iterations are used to solve Equation (8) and (9) concurrently.

$$T_{fc} = \frac{\dot{e}_{gen,mm} {r_{i,sh}}^2}{4\, k_{fuel}} + T_{i,sh} \tag{8}$$

$$k = 8.9111 - 0.01393T + 1.1451 \times 10^{-5}T^2 - 4.2535 \times 10^{-9}T^3 + 6.0729 \times 10^{-13}T^4 \tag{9}$$

## 5. Results

The results are presented in Figures 6 − 9. In all the investigated cases, the sheath temperature for the calculated conditions does not exceed the sheath maximum temperature of 850°C. However, the fuel centerline temperature does surpass the industry accepted limit of 1850°C in all considered cases.

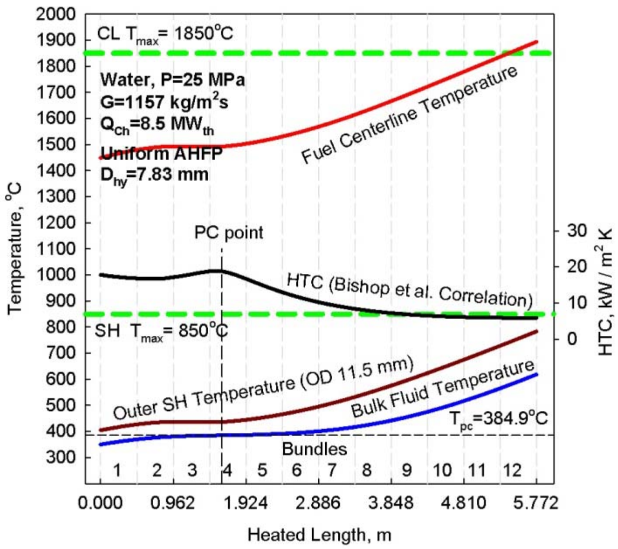


**Figure 6. Temperature and HTC profiles along heated length with uniform AHFP.**

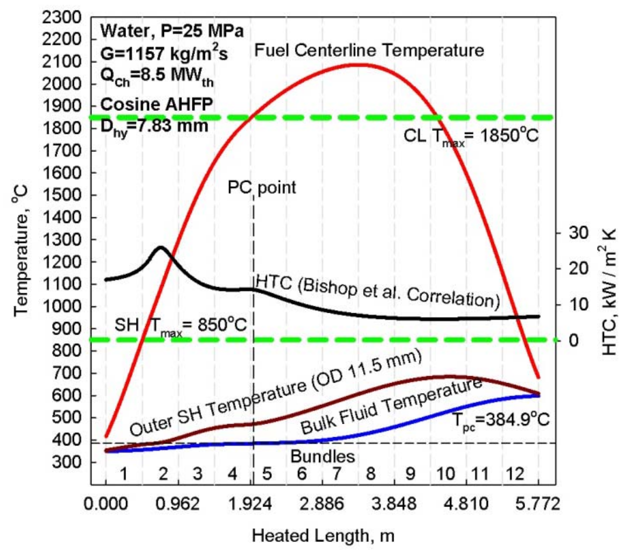


**Figure 7. Temperature and HTC profiles along heated length with cosine AHFP.**

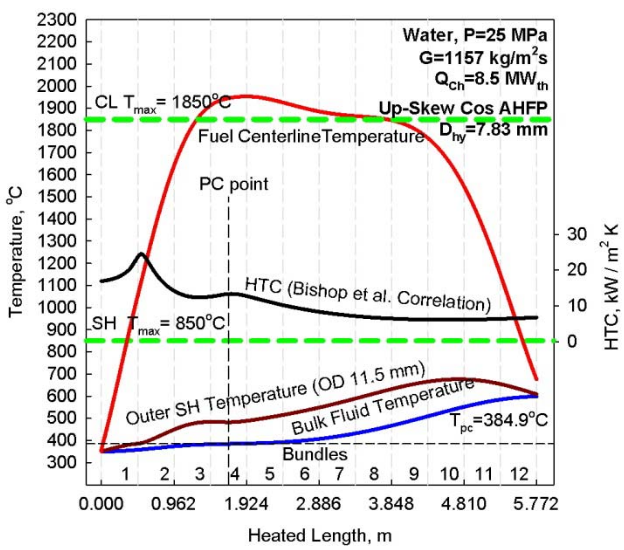


**Figure 8. Temperature and HTC profiles along heated length with upstream-skewed cosine AHFP.**

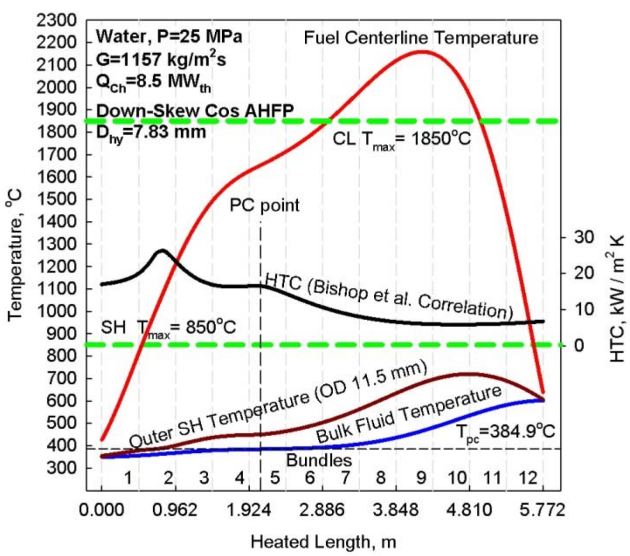


**Figure 9. Temperature and HTC profiles along heated length with downstream-skewed cosine AHFP.**

## 6. Conclusions

Mixed oxide fuel might be used at SCWR conditions with caution, because the fuel centerline temperature exceeds the industry accepted limit of 1850°C at some conditions. The fuel centerline temperature may be decreased by increase flow turbulization with bundle appendages thus improving the heat transfer. A HTC for bundles with SCW has yet to be developed. However, there are known solutions in terms of decreasing the fuel centerline temperature:

- Hollow fuel pellet.
- Smaller diameter fuel rods, but with the addition of more fuel rods per bundle.
- In the worst case, the power per channel could be decreased.

## 7. Acknowledgement

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