Sensitivity Analysis of Fuel Centerline Temperatures in SuperCritical Water-cooled Reactors (SCWRs)

By

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ABSTRACT

SuperCritical Water-cooled Reactors (SCWRs) are one of the six nuclear-reactor concepts currently being developed under the Generation-IV International Forum (GIF). A main advantage of SCW Nuclear Power Plants (NPPs) is that they offer higher thermal efficiencies compared to those of current conventional NPPs. Unlike today’s conventional NPPs, which have thermal efficiencies between 30 – 35%, SCW NPPs will have thermal efficiencies within a range of 45 – 50%, owing to high operating temperatures and pressures (i.e., coolant temperatures as high as 625°C at 25 MPa pressure).

The use of current fuel bundles with UO$_2$ fuel at the high operating parameters of SCWRs may cause high fuel centerline temperatures, which could lead to fuel failure and fission gas release. Studies have shown that when the Variant-20 (43-element) fuel bundle was examined at SCW conditions, the fuel centerline temperature industry limit of 1850°C for UO$_2$ and the sheath temperature design limit of 850°C might be exceeded. Therefore, new fuel-bundle designs, which comply with the design requirements, are required for future use in SCWRs.

The main objective of this study to conduct a sensitivity analysis in order to identify the main factors that leads to fuel centerline temperature reduction. Therefore, a 54-element fuel bundle with smaller diameter of fuel elements compared to that of the 43-element bundle was designed and various nuclear fuels are examined for future use in a generic Pressure Tube (PT) SCWR. The 54-element bundle consists of 53 heated fuel elements with an outer diameter of 9.5 mm and one central unheated element of 20-mm outer diameter which contains burnable poison. The 54-element fuel bundle has an outer diameter of 103.45 mm, which is the same as the outer diameter of the 43-element fuel bundle. After developing the 54-element fuel bundle, one-dimensional heat-transfer analysis was conducted using MATLAB and NIST REFPROP programs. As a
result, the Heat Transfer Coefficient (HTC), bulk-fluid, sheath and fuel centerline temperature profiles were generated along the heated length of 5.772 m for a generic fuel channel. The fuel centerline and sheath temperature profiles have been determined at four Axial Heat Flux Profiles (AHFPs) using an average thermal power per channel of 8.5 MWth. The four examined AHFPs are the uniform, cosine, upstream-skewed and downstream-skewed profiles.

Additionally, this study focuses on investigating a possibility of using low, enhanced and high thermal-conductivity fuels. The low thermal-conductivity fuels, which have been examined in this study, are uranium dioxide (UO2), Mixed Oxide (MOX) and Thoria (ThO2) fuels. The examined enhanced thermal-conductivity fuels are uranium dioxide – silicon carbide (UO2 - SiC) and uranium dioxide - beryllium oxide (UO2 - BeO). Lastly, uranium carbide (UC), uranium dicarbide (UC2) and uranium nitride (UN) are the selected high thermal-conductivity fuels, which have been proposed for use in SCWRs.

A comparison has been made between the low, enhanced and high thermal-conductivity fuels in order to identify the fuel centerline temperature behaviour when different nuclear fuels are used. Also, in the process of conducting the sensitivity analysis, the HTC was calculated using the Mokry et al. correlation, which is the most accurate supercritical water heat-transfer correlation so far. The sheath and the fuel centerline temperature profiles were determined for two cases. In Case 1, the HTC was calculated based on the Mokry et al. correlation, while in Case 2, the HTC values calculated for Case 1 were multiplied by a factor of 2. This factor was used in order to identify the amount of decrease in temperatures if the heat transfer is enhanced with appendages.

Results of this analysis indicate that the use of the newly developed 54-element fuel bundle along with the proposed fuels is promising when compared with the
Variant-20 (43-element) fuel bundle. Overall, the fuel centerline and sheath temperatures were below the industry and design limits when most of the proposed fuels were examined in the 54-element fuel bundle, however, the fuel centerline temperature limit was exceeded while MOX fuel was examined.

Keywords: SCWRs, Fuel Centerline Temperature, Sheath Temperature, High Thermal Conductivity Fuels, Low Thermal Conductivity Fuels, HTC
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**NOMENCLATURE**

\[ A_{cs} \] cross sectional area, m\(^2\)

\[ A_{fl} \] flow area, m\(^2\)

\[ c_p \] specific heat at constant pressure, J/kg K

\[ \bar{c}_p \] averaged specific heat within the range of \((t_w - t_b)\); \(\left(\frac{H_w - H_b}{T_w - T_b}\right)\), J/kg K

\[ D \] diameter, m

\[ D_{hy} \] hydraulic-equivalent diameter; \(\left(\frac{4A_{fl}}{P_w}\right)\), m

\[ E \] Young’s modulus, MPa

\[ e_{gen} \] heat generation, W/m\(^3\)

\[ G \] mass flux, kg/m\(^2\)s; \(\left(\frac{m}{A_{fl}}\right)\)

\[ H \] specific enthalpy, J/kg

\[ h \] heat transfer coefficient, W/m\(^2\)K

\[ k \] thermal conductivity, W/m K

\[ L \] length, m

\[ N \] number

\[ P \] pressure, MPa

\[ Q \] heat transfer rate, W

\[ q \] heat flux, W/m\(^2\); \(\left(\frac{Q}{A_h}\right)\)

\[ q' \] linear element rating, W/m

\[ R, r \] radius, m

\[ T, t \] temperature, °C
\[ x \text{ axial coordinate, m} \]

**Greek Letters**

\[ \delta \text{ thickness, m} \]
\[ \pi \text{ pi number, 3.1416} \]
\[ \mu \text{ dynamic viscosity, Pa s} \]
\[ \nu \text{ Poisson’s ratio} \]
\[ \rho \text{ density, kg/m}^3 \]

**Non-dimensional Numbers**

\[ \text{Nu} \text{ Nusselt number; } \left( \frac{hD}{k} \right) \]
\[ \text{Pr} \text{ Prandtl number; } \left( \frac{\mu c_p}{k} \right) = \left( \frac{\nu}{\alpha} \right) \]
\[ \overline{\text{Pr}} \text{ averaged Prandtl number within the range of } (t_w - t_b); \left( \frac{\mu c_p}{k} \right) \]
\[ \text{Re} \text{ Reynolds number; } \left( \frac{GD}{\mu} \right) \]

**Subscripts or superscripts**

\[ \text{avg} \text{ average} \]
\[ \text{b} \text{ bulk} \]
\[ \text{c} \text{ center} \]
\[ \text{ch} \text{ channel} \]
\[ \text{cr} \text{ critical} \]
\[ \text{dht} \text{ deteriorated heat transfer} \]
\[ \text{e} \text{ elements} \]
el  electric
fl  flow
gen  generation
h  heated
hy  hydraulic-equivalent
i  inner
o  outer
pc  pseudocritical
sh  sheath
th  thermal
tot  total
w  wall

Abbreviations and Acronyms

AHFP                    Axial Heat Flux Profile
BOP                     Balance of Plant
BWR                     Boiling Water Reactor
CANDU                   CANada Deuterium Uranium (reactor)
CANFLEX                 CANdu FLEXible fuelling
DHT                     Deteriorate Heat Transfer
EBR-II                  Experimental Breeder Reactor
FCC                     Face Centered Cubic
FFTF                    Fast Flux Test Facility
GFR                     Gas-cooled Fast Reactor
GIF                     Generation-IV International Forum
<table>
<thead>
<tr>
<th>Acronym</th>
<th>Description</th>
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<tbody>
<tr>
<td>HEC</td>
<td>High Efficiency Channel</td>
</tr>
<tr>
<td>HPT</td>
<td>High Pressure Turbine</td>
</tr>
<tr>
<td>HTC</td>
<td>Heat Transfer Coefficient</td>
</tr>
<tr>
<td>IPT</td>
<td>Intermediate Pressure Turbine</td>
</tr>
<tr>
<td>LFR</td>
<td>Lead-cooled Fast Reactors</td>
</tr>
<tr>
<td>LMFBR</td>
<td>Light Metal Fast Breeder Reactor</td>
</tr>
<tr>
<td>LOCA</td>
<td>Loss of Coolant Accident</td>
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<tr>
<td>LPT</td>
<td>Low Pressure Turbine</td>
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<td>LWR</td>
<td>Liquid Water Reactors</td>
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<tr>
<td>MATLAB</td>
<td>MATrix LABoratory</td>
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<td>MSR</td>
<td>Molten Salt-cooled Reactors</td>
</tr>
<tr>
<td>Mo</td>
<td>Molybdenum</td>
</tr>
<tr>
<td>MOX</td>
<td>Mixed OXide Fuel</td>
</tr>
<tr>
<td>Nb</td>
<td>Niobium</td>
</tr>
<tr>
<td>NIST</td>
<td>National Institute of Standards and Technology</td>
</tr>
<tr>
<td>NPP</td>
<td>Nuclear Power Plant</td>
</tr>
<tr>
<td>R &amp; D</td>
<td>Research and Developement</td>
</tr>
<tr>
<td>PT</td>
<td>Pressure Tube</td>
</tr>
<tr>
<td>PuO₂</td>
<td>Plutonium Oxide</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurized Water Reactor</td>
</tr>
<tr>
<td>REC</td>
<td>Re-Entrant Channel</td>
</tr>
<tr>
<td>REFPROP</td>
<td>REFerence PROPerties</td>
</tr>
<tr>
<td>SCW</td>
<td>SuperCritical Water</td>
</tr>
<tr>
<td>SCWR</td>
<td>SuperCritical Water-cooled Reactors</td>
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<tr>
<td>Abbreviation</td>
<td>Description</td>
</tr>
<tr>
<td>--------------</td>
<td>----------------------------------</td>
</tr>
<tr>
<td>SFRs</td>
<td>Sodium-cooled Fast Reactors</td>
</tr>
<tr>
<td>Sn</td>
<td>Tin</td>
</tr>
<tr>
<td>SiC</td>
<td>Silicon Carbide</td>
</tr>
<tr>
<td>SRH</td>
<td>Steam Re-Heat</td>
</tr>
<tr>
<td>ThO₂</td>
<td>Thorium Oxide</td>
</tr>
<tr>
<td>UC</td>
<td>Uranium Carbide</td>
</tr>
<tr>
<td>UC₂</td>
<td>Uranium Dicarbide</td>
</tr>
<tr>
<td>UN</td>
<td>Uranium Nitride</td>
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<td>UO</td>
<td>University of Ottawa</td>
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<tr>
<td>UO₂</td>
<td>Uranium Dioxide</td>
</tr>
<tr>
<td>VHTRs</td>
<td>Very High-Temperature gas-cooled Reactors</td>
</tr>
<tr>
<td>YSZ</td>
<td>Yttria Stabilized Zirconia</td>
</tr>
<tr>
<td>Zr</td>
<td>Zirconium</td>
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</table>
1.0 INTRODUCTION

Nuclear reactors are one of the cleanest energy sources available worldwide. Currently, 436 Generation II, III and III + nuclear-power reactors have been implemented, and contribute to approximately 16% of the world electricity needs (Aswathanarayana et al. 2010). Most reactor designs developed in the past generations have proven to be reliable; however, they are not energy efficient due to the low operating pressures and temperatures. Starting from the first generation, nuclear reactor development has been progressing towards safer and more efficient nuclear power plants. Figure 1 shows the evolution of the different nuclear reactors from Generation I to Generation IV.

![Figure 1: Evolution of Nuclear Reactor Concepts (www.gen-4.org).](image)

In 2001, an international program for the development of future nuclear reactors called the Generation IV International Forum (GIF) was established. There are 13 countries involved in the GIF and their main goals are the development of
nuclear reactors by the year 2030 with improved sustainability, economics, safety, reliability, proliferation resistance, and physical protection (Pioro, 2010). Nuclear energy research and development (R&D) programs are working on six nuclear-reactor concepts under the GIF program. The six Gen IV nuclear-reactor concepts under research and development are:

- Gas-cooled Fast Reactors (GFRs)
- Very High-Temperature gas-cooled Reactors (VHTRs)
- Sodium-cooled Fast Reactors (SFRs)
- Lead-cooled Fast Reactors (LFRs)
- Molten Salt-cooled Reactors (MSRs)
- SuperCritical Water-cooled Reactors (SCWRs)

The above mentioned nuclear reactor concepts employ different designs and operating parameters (i.e., temperatures and pressures). In terms of neutron spectrum, these reactors can be classified into two categories: 1) thermal and 2) fast reactors. The Generation IV thermal neutron-spectrum reactors are VHTR, MSR and SCWR. Alternately, the three fast neutron-spectrum reactors are GFR, SFR and LFR. Table 1 summarizes the operating parameters of the six Gen-IV nuclear reactor concepts.
Since this study focuses on a generic Pressure Tube (PT) SCWR concept, further information will be provided on the design specifications of this concept. The PT SCWR will operate above the thermodynamic critical point of water, which is at a pressure of 22.1 MPa and a temperature of 374°C. Operating at such high temperatures and pressures, SCWRs will have high thermal efficiencies in the range of 45 – 50% as opposed to current conventional water-cooled Nuclear Power Plants (NPPs) that have thermal efficiencies of around 30 – 35%. There are two fuel cycle options proposed for the SCWRs; the open cycle with a thermal neutron-spectrum and the closed cycle with the fast neutron-spectrum (Weston, 2007).
The use of SCWRs can result in several advantages over Light Water Reactors (LWRs) such as the higher thermal efficiencies, reduced coolant pumps, piping, and lower pumping power due to the lower coolant mass flow rates and a smaller containment building. Additionally, some of the expensive equipment used in previous designs such as steam dryers, steam separators, recirculation pumps, and steam generators can be eliminated. Figure 2 shows a general schematic of a SCWR concept using the Rankine steam cycle.

Figure 2: Schematic of a PT type SCWR (US DOE, 2002).

The focus of this study is to conduct sensitivity analysis, which includes the calculation of the fuel centerline and sheath temperature profiles of a generic PT SCWR. The sensitivity analysis incorporates the use of different nuclear fuels, changing the Heat Transfer Coefficient (HTC) and the number of elements in the fuel bundle and examining the effect of changing these parameters on the fuel
centerline temperature and sheath temperature. The coolant temperatures (350 – 625°C) and pressure (~25 MPa) have been used to determine the fuel centerline temperature of several nuclear fuels. These fuels are enclosed in a newly developed 54-element fuel bundle design. A summary of the major parameters of the reference SCWR design is shown in Table 2, which indicates that the proposed PT SCWR will have a thermal power of around 2540 MW\textsubscript{th} and an electrical power ranging from 1143 - 1220 MW\textsubscript{el}. These parameters indicate that the thermal efficiency would range from 45 to 48%. However, since the SCWR is still in the conceptual design phase, these parameters change slightly from one source to another and the thermal efficiency can range from 40 – 50%.

Table 2: Major SCWR Design Parameters (Naidin et al. 2009)

<table>
<thead>
<tr>
<th>Parameters</th>
<th>Unit</th>
<th>Generic PT SCWR</th>
</tr>
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<tbody>
<tr>
<td>Thermal Power</td>
<td>MW\textsubscript{th}</td>
<td>2540</td>
</tr>
<tr>
<td>Electric Power</td>
<td>MW\textsubscript{el}</td>
<td>1143 – 1220</td>
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<tr>
<td>Thermal Efficiency</td>
<td>%</td>
<td>45 – 48</td>
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<tr>
<td>Coolant</td>
<td>-</td>
<td>H\textsubscript{2}O</td>
</tr>
<tr>
<td>Moderator</td>
<td>-</td>
<td>D\textsubscript{2}O</td>
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<tr>
<td>Pressure of SCW at Inlet/Outlet</td>
<td>MPa</td>
<td>25.8 – 25</td>
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<tr>
<td>Pressure of SHS at Inlet/Outlet</td>
<td>MPa</td>
<td>6.1 – 5.7</td>
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<tr>
<td>(T_{in}/T_{out}) Coolant (SCW)</td>
<td>°C</td>
<td>350 – 625</td>
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<tr>
<td>(T_{in}/T_{out}) Coolant (SHS)</td>
<td>°C</td>
<td>400 – 625</td>
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<td>Mass Flow Rate per SCW/SRH Channel</td>
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<tr>
<td>Thermal Power per SCW/SRH Channel</td>
<td>MW</td>
<td>8.5 – 5.5</td>
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<td># of SCW/SRH Channels</td>
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<td>220 – 80</td>
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<tr>
<td>Heat Flux in SCW/SRH Channel</td>
<td>kW/m\textsuperscript{2}</td>
<td>970 – 628</td>
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</tbody>
</table>
The following chapter will provide a literature survey related to different aspects of SCWRs such as the proposed thermal cycles, reactor core designs, fuel-channel options and explanations on general definitions related to critical and supercritical pressures. Afterwards, an explanation of several HTC correlations followed by sheath material selection and properties of the eight nuclear fuels presented in this study.

Next is the methodology and calculations chapter, which will explain in details the fuel-bundle geometry and fuel-centerline-temperature calculations. Afterwards, the results and discussion chapter will provide fuel-centerline-temperature profiles that are generated for the fuels proposed in this study. Some of these results were published at several conferences such as the 19th 20th International Conference On Nuclear Engineering (ICONE) at which the paper presented was awarded the best North American Student Paper certificate. A list of publications are listed in Appendix C.
2.0 LITERATURE SURVEY

2.1 SuperCritical Water-cooled Reactor (SCWRs)

The first concepts of SCWRs were developed by Russia and USA in the 1950s. Nowadays, SCWRs Research and Development (R&D) have gained momentum in various countries around the world including Canada. The interest in developing this type of reactor increased due to the great demand of introducing green energy sources that are environmentally friendly.

Currently, there are two thermal cycles proposed for use in SCWRs. These two cycles are the direct cycle with Moisture Separator and Reheat (MSR) and the direct single-reheat cycle. The PT-SCWR direct thermal cycle was introduced by Yetisir et al. (2011). This cycle takes advantage of layout of the Balance Of Plant (BOP) of the current modern SCW fossil-fired power plants.

Figure 3 shows a general scheme of such a BOP layout and thermal cycle. This cycle eliminates the use of steam generators such that the steam passes directly through the high pressure turbine. From the outlet of the high pressure turbine steam is directed to the intermediate pressure turbine. Then, the steam passes through the moisture separator to remove the moisture from the steam. Furthermore, flow of steam passes through the low pressure turbine and eventually to the condenser. In the condenser, the steam is condensed to liquid which is then passed through a series of low pressure and high pressure heaters to increase the temperature and pressure of the coolant to the required inlet values.
Figure 3: Canadian Direct-Cycle SuperCritical Water Nuclear Reactor (Yetisir et al. 2011).

Alternately, the single re-heat cycle uses fuel channels that are located at the periphery of the core in order to superheat steam at a subcritical pressure. The single re-heat cycle was introduced by Naidin et al. (2009) and is based on the already operating SuperCritical (SC) fossil-fired thermal power plants. Figure 4 shows the schematic diagram of the steam re-heat cycle.

Supercritical steam passes through a high-pressure turbine at which the temperature and pressure drop. Afterwards, the steam at a pressure of 5.7 MPa passes through the Steam Re-Heat (SRH) fuel channels through which the temperature increases to about 625°C. Then, the superheated steam flows through the intermediate-pressure turbine and eventually through the low-pressure turbines. The condenser cools down the output of the low-pressure turbine to liquid. The temperature and pressure of the feedwater is increased through a series of open and closed feed water heat exchangers. The steam,
which is extracted from the intermediate- and high-pressure turbines, is used as a heat source in order to increase the temperature of the feed water to the reactor inlet temperature and pressure. This cycle is promising due to high thermal efficiency and the use of a proven technology, which is already implemented in coal-fired power plants. One main disadvantage of implementing this cycle in NPPs is the complexity of introducing SRH channels to the reactor core.

![Figure 4: Single-reheat cycle for SCW NPP (Naidin et al. 2009)](image)

2.2 Core Designs

There are two pressure-channel concepts proposed for use in SCWRs. The two concepts are the on-power fuelled horizontal core and the batch fuelled vertical core. In both concepts, the High Efficiency Channel (HEC), which is discussed in Section 2.3.1, is considered as the primary option. Section 2.2.1 and 2.2.2 will give a brief description of the two core design concepts.
2.2.1 Horizontal Core PT Design

The horizontal PT design is an evolution of the traditional design used in CANDU reactors, however there are some safety related concerns associated with the on-line refuelling at SCW conditions. The horizontal PT design and the proposed generic 1200-MW$_{el}$ PT SCWR channel layout are shown in Figures 5 and 6.

![Diagram](image)

**Figure 5: General Concept of a Horizontal Core Pressurized-Channel SCWR (Pioro and Duffey, 2007).**

The reactor core of the proposed 1200-MW$_{el}$ SCWR consists of 300 fuel channels located inside a cylindrical vessel called calandria vessel. From these 300 fuel channels, there are 220 (SCW) fuel channels located at the center of the reactor core and 80 (SRH) fuel channels on the periphery of the core. The coolant inlet and outlet temperatures are 350 and 625°C, while the coolant pressure is at 25MPa. Consequently, the core consists of distributed pressure channels with a wall thickness of around 10 mm.
2.2.2 Vertical PT Core Design

The second and more recent PT SCWR design is a vertical core design. The vertical core design is intended to operate on the concept of batch refuelling. In general, the vertical batch-refuelled core design was introduced due to the different safety considerations associated with the horizontal PT design, which employs the on-line refuelling scheme. Therefore, to avoid connecting the fuelling machine to a pressure channel at the high temperatures and pressures of SCWRs, the vertical core design with off-power batch refuelling is being considered.
The presented vertical SCWR reactor-core design is shown in Figure 7. It can be observed from this figure that the vertical design uses an inlet plenum at which the light water coolant enters the core. The coolant enters the core at a pressure of 25 MPa and an inlet temperature of 350°C. After the coolant enters the inlet plenum, it flows downward and eventually it is heated to a temperature of 625°C.

![Figure 7: Preliminary Vertical SCWR Pressure-Tube Concept (Yetisir et al. 2011)](image)

There are several advantages obtained in the vertical PT SCWR concept. These advantages are the use of offline batch refuelling, which enhances safety, eliminating inlet feeders used in the horizontal designs, passive cooling can be implemented through natural convection of a heavy-water moderator. It is also possible to provide easy PT replacement by removing the inlet plenum head. The SCWR research is now moving towards the vertical core design due to the above mentioned safety related concerns. Furthermore, Table 1 in section 1.0 (Introduction) lists in details the operating parameters of a generic PT SCWR.
2.3 SCWR Fuel Channel Options

There are two fuel-channel designs that are being considered for future PT reactors: the High Efficiency Channel (HEC) and the Re-Entrant Channel (REC). These two design concepts and their characteristics are briefly described in Sections 2.3.1 and 2.3.2.

2.3.1 High Efficiency Channel (HEC)

The HEC design differs from the currently used CANDU-6 fuel channel in several ways. HEC does not employ a calandria tube to separate the generic pressure tube from the moderator. In fact, the pressure tube of each fuel channel is in direct contact with the moderator, which operates at an average temperature of 80°C. The pressure tube is thicker than that of the CANDU-6 fuel channel in order to withstand the high operating temperatures and pressures of the SCWR. The two materials: Excel (Zr-3.5wt% Sn-0.8wt% MO-0.8wt%Nb-1130 ppm O) and Zr 2.5 wt% Nb; are proposed for use as the PT material. These two materials were proposed due to their high creep resistance, low creep growth rates, and low thermal-neutron absorption cross-section (Chow et al. 2008).

On the other hand, in order to protect the pressure tube from being exposed to high temperatures and reduce the heat loss from the coolant to the moderator, it is then thermally insulated from the hot coolant by a ceramic insulator. The ceramic insulator must have good corrosion resistance and provide an effective thermal barrier that can withstand thermal stresses. A porous material must be selected to increase thermal resistance and improve thermal-shock resistance. The insulator must also have a low neutron-absorption cross-section and low thermal conductivity. The material proposed for use as the ceramic insulator is
the Porous Yttria Stabilized Zirconia (YSZ) due to its compliance with the previously mentioned requirements (Chow et al. 2008).

The ceramic insulator is then protected by a metal liner. This metal liner of about 1-mm thickness is inserted between the ceramic insulator and the fuel bundle. It is a perforated tube so that it protects the ceramic insulator from being damaged by the fuel bundles during operation, refuelling and from erosion due to coolant flow. The material of the liner should be able to stand up to wear and fretting. Also, coating may be required in order to reduce the oxidation of the liner material.

Finally, the material of the fuel sheath should be able to withstand the pressure differences between the coolant and the fuel sides. The sheath material has to have also high temperature resistance and low neutron absorption cross-section for thermal neutrons. Unlike CANDU reactors, zirconium alloys cannot be used in SCWRs due the high corrosion rates at temperatures between 400 and 800°C; therefore, nickel or stainless steel alloys are potential candidates for the sheath material. Figure 8 shows the HEC with a CANFLEX fuel bundle.
2.3.2 Re-Entrant Fuel Channel (REC)

The REC consists of a flow tube, a PT, and a calandria tube, which is in direct contact with the liquid moderator. The pressure tube is separated from the heavy-water moderator by an annular gap filled with gas. The coolant flows first between the PT and an inner tube called the flow tube. It then turns around and flows through the inner tube, where the fuel bundles reside. This concept allows the pressure tube to operate at lower temperatures, and, therefore zirconium alloys could be used as PT material. Since corrosion increases with temperature, a coating on the sheath would be required to reduce the corrosion rates to acceptable levels. The REC fuel channel is shown in Figure 9.
The REC was furthermore modified in order to maintain the mechanical integrity of the pressure tube and reduce the heat losses from the coolant to the moderator. In order to achieve this goal, a ceramic insulator was introduced in the new design shown in Figure 10. The implementation of a ceramic insulator as a thermal barrier reduces operating temperature and the thermal stresses on the pressure tube. This would allow the pressure tube to operate at lower temperatures and therefore, a wider range of materials can be investigated for use in the REC concept. However, for the purpose of this study, the HEC concept was used as the SCWR fuel channel concept and the calculations were based on parameters of the HEC.
2.4 Supercritical Fluids

2.4.1 Definitions and Expressions

General definitions and expressions related to critical and supercritical pressures are necessary for a complete understanding of SCWRs and the heat-transfer calculations associated with the development of such reactors. A list of the needed terms and expressions as presented in Pioro and Duffey (2007) are listed below. In Figure 11, a thermodynamic diagram for water is shown to illustrate these terms and expressions.

*Supercritical fluid* is a fluid at pressures and temperatures that are higher than the critical pressure and critical temperature. However, in the current paper, the term *supercritical fluid* includes both terms – *supercritical fluid* and *compressed fluid*.

*Deteriorated heat transfer (DHT)* is characterized by lower values of the wall heat transfer coefficient compared to those at the normal heat transfer regime.
and hence has higher values of wall temperature within some part of a test section or within the entire test section.

**Pseudocritical point** (characterized with $P_{pc}$ and $T_{pc}$) is a point at a pressure above the critical pressure and at a temperature ($T_{pc} > T_{cr}$) corresponding to the maximum value of the specific heat for this particular pressure.

![Figure 11: Pressure-Temperature Diagram of Water in the Critical Region (Mokry et al. 2009a).](image)

Properties of water change as they cross the critical point. The critical point is found at a temperature of 373.95°C and a pressure of 22.064 MPa. In the critical point and beyond a fluid is considered as a single-phase substance. Section 2.4.2 will discuss further the thermophysical properties of fluids at the critical and pseudocritical points.
2.4.2 Supercritical Fluid Properties

The thermophysical properties of fluids undergo significant changes at the critical and pseudocritical points. Consequently, it is necessary to capture such changes in the coolant properties of a SCWR for the purpose of calculating the fuel centerline temperature. The National Institute of Standards and Technology (NIST) Reference Fluid Properties (REFPROP) software was used to calculate the thermophysical properties of water. Figure 12 shows changes in water properties within the pseudocritical range at a pressure of 25 MPa. The most significant changes in specific heat, density, thermal conductivity and viscosity can be observed around ± 25°C from the pseudocritical temperature (384.9°C at 25 MPa).

![Figure 12: Selected Properties of SuperCritical Water within the Pseudocritical Range (Mokry et al. 2009a).](image-url)
2.5 Heat Transfer Correlations

Over the years, different heat transfer correlations have been developed in order to calculate the Heat Transfer Coefficient (HTC) in forced convection of several fluids. Some of these heat transfer correlations were developed based on water as the working fluid at supercritical pressures. However, these correlations show different results within the same temperature and pressure operating ranges.

At subcritical pressures, the Dittus-Boelter heat-transfer correlation (i.e., Eq.(1)) is widely used to calculate the HTC (Dittus and Boelter, 1930). This correlation was also proposed for use in calculating forced-convection heat-transfer in turbulent flows at subcritical pressures.

\[ Nu_b = 0.0243 \, Re_b^{0.8} \, Pr_b^{0.4} \]  \hspace{1cm} (1)

However, in 1976, Schnurr et al. realized that the Dittus-Boelter correlation can produce unrealistic results within the same flow conditions, especially, within the critical and pseudocritical regions. These unrealistic results are due to the fact that the correlation is highly sensitive to property variations, which occur at the above mentioned regions. Nonetheless, the Dittus-Boelter correlation was used afterwards as the basis in the development of various supercritical heat-transfer correlations.

In 1965, Bishop et al. conducted experiments on supercritical water flowing upward inside bare tubes and annuli. These experiments were performed over the in the of operating parameters: pressure 22.8 – 27.6 MPa, bulk-fluid temperature 282 – 527°C, mass flux – 3662 kg/m²s and heat flux 0.31 – 3.46MW/m² (Pioro and Duffey, 2007). Their data for heat transfer in tubes were generalized using the following correlation, which predicts the HTC within a ±15% uncertainty. Bishop et al. (1965) also used the cross-sectional averaged
Prandtl number and the last term in the correlation that accounts for the entrance-region effect.

\[ Nu_b = 0.0069 \text{Re}_b^{0.9} \text{Pr}_b^{0.66} \left( \frac{\rho_w}{\rho_b} \right)^{0.43} \left( 1 + 2.4 \frac{D}{x} \right) \]  \hspace{1cm} (2)

The last term in the Bishop et al. (1965) takes into account the entrance effects; however, in case of a fuel channel, the flow is considered to be turbulent from the beginning of the fuel channel due to the presence of the fuel bundles, end plates, and other appendages. As a result, the last term in Eq. (2) can be eliminated when calculating the fuel centerline temperature of an SCWR fuel channel. In 2010, Mokry et al. developed a new correlation, shown as Eq. (3), which takes into account such consideration.

\[ Nu_b = 0.0061 \text{Re}_b^{0.904} \text{Pr}_b^{0.684} \left( \frac{\rho_w}{\rho_b} \right)^{0.564} \]  \hspace{1cm} (3)

However, these correlations were developed for vertical bare tubes and intended to be used only within the normal and improved heat-transfer regimes. Due to this, an empirical correlation shown as Eq. (4) was proposed for the determination of the onset of deteriorated heat transfer regime (Gabaraev, 2007).

\[ q_{dht} = -58.97 + 0.745 G \]  \hspace{1cm} (4)

The Mokry et al. (2010) correlation was developed based on the most recently updated heat-transfer set of data and the latest thermophysical properties of water from NIST within the SCWRs operating range. In addition, an independent study conducted by Zahlan et al. (2010) showed that the Mokry et al. correlation results in the lowest root-mean-square error when compared with several well-known heat-transfer correlations. Therefore, the Mokry et al. correlation will be used to calculate the HTC in this study as a conservative approach, since, there are no HTC correlations developed for such a fuel bundle.
Figure 13 shows scatter plots of experimental HTC values versus calculated HTC values using the Mokry et al. correlation. As shown in Figure 13, the Mokry et al. correlation predicts the experimental HTC and wall temperature within ±25% and ±15% uncertainty, respectively. It is important to mention these uncertainties since they can have an impact on the calculated fuel centerline temperatures and its maximum values.

It is also important to mention that there is only one supercritical-water heat-transfer correlation for fuel bundles developed by Dyadyakin and Popov in 1977 (Pioro and Duffey, 2007). The Dyadyakin and Popov correlation was obtained using an experimental setup with a tight-lattice 7-element helically-finned bundle cooled with water and is shown in Figure 14. However, heat transfer correlations for fuel bundles are generally very sensitive to the bundle design. As a result, using this correlation to calculate the HTC of a different fuel bundle will lead to inaccurate results, therefore, this correlation was not used in the present study. The Dyadyakin and Popov HTC correlation is presented as Eq. (5):
\[ \text{Nu}_x = 0.021 \text{Re}_x^{0.8} \text{Pr}_x^{0.7} \left( \frac{\rho_w}{\rho_b} \right)_x^{0.45} \left( \frac{\mu_w}{\mu_b} \right)_x^{0.2} \left( \frac{\rho_b}{\rho_{in}} \right)_x^{0.1} (1 + 2.5 \frac{D_h y}{x}) \] (5)

Figure 14: Dyadyakin and Popov Test Bundle Configuration (courtesy of W. Peiman).

2.6 Fuel Bundle Geometry

Several fuel bundle designs were developed for use in pressure tube reactors. In this study, the primal objective of developing a new fuel bundle is to reduce the fuel centerline temperature and the sheath temperature in order to comply with the industry and design temperature limits. The general trend is to increase the number of fuel elements while reducing their diameter. Figure 15 shows several fuel bundle geometries such as the 37-element, CANFLEX, Variant-18 and Variant-20, respectively.

The 37-element fuel bundle consists of 37 fuel elements all with the same outer diameter of 13.06 mm. The CANFLEX fuel bundle consists of 43-elements with central and inner ring elements of 13.5 mm diameter, intermediate and outer ring elements of 11.5 mm diameter. Furthermore, the Variant-18 and Variant-20 fuel
bundles consist of 43 fuel elements with an outer diameter of 11.5 mm and a central unheated element with an outer diameter of 18 and 20 mm, respectively. The central elements are filled with burnable poison; therefore, these elements are considered as unheated.

![Figure 15: CANDU Fuel Bundle Designs (Leung et al. 2008)](image)

Previously, the Variant-20 (43-element) fuel bundle was analyzed at SCW conditions. When the fuel centerline temperature was calculated using UO$_2$ as a fuel, it was found that the fuel centerline temperature might exceed the industry accepted limit of 1850°C for UO$_2$ fuel. This indicates that the use of UO$_2$ fuel in the Variant-20 fuel bundle can result in high fuel centerline temperatures that would eventually result in fuel melting. To further illustrate the temperature profile, Figures 16 and 17 show the cosine and downstream skewed Axial Heat Flux Profiles (AHFP) of the UO$_2$ fuel enclosed in a Variant-20 fuel bundle. These temperature profiles indicate that the fuel centerline temperatures exceeded the
limit when UO$_2$ fuel was examined in the Variant-20 fuel bundle at SCW conditions. The maximum fuel centerline temperature in the downstream-skewed AHFP was noticed at around 2250°C, which is 400°C above the industry accepted limit of 1850°C and only 600°C below the UO$_2$ melting point of 2850°C.

![Cosine AHFP, Variant-20 Fuel Bundle with UO$_2$ Fuel (courtesy of W. Peiman).](image)

Figure 16: Cosine AHFP, Variant-20 Fuel Bundle with UO$_2$ Fuel (courtesy of W. Peiman).
The main causes of the noticed high fuel-centerline-temperature are the high operating parameters at SCW conditions, the low thermal-conductivity of UO$_2$ fuel and the Variant-20 fuel bundle geometry. For these reasons, the process of conducting the sensitivity analysis included developing different fuel-bundle designs including a 54-element fuel bundle. Also, in order to observe what mostly affects the fuel centerline temperature at SCW conditions, different nuclear fuels were examined as well as multiples of the HTC. The 54-element bundle design was examined using different nuclear fuels (see Section 2.7 for examined fuels) and four AHFPs.
2.7 Sheath Material

Material selection is one of the major challenges in the development of the SCWR and other high-temperature reactors. The material selection is concerned with in-core and out-of-core components of the reactor. One of the most important in-core components of the reactor is the fuel bundle. A candidate material for the fuel bundle must meet essential requirements in order to prevent fuel element failure. These requirements are mainly high corrosion resistance, low neutron absorption, high mechanical strength and high thermal conductivity. Zirconium alloys are one of the most popular sheath materials used in current nuclear reactors. However, these alloys cannot be used in SCWRs due to their low corrosion resistance at temperatures above 350°C (Duffey & Hedges, 1999).

In this study, three sheath material options were investigated and one material was selected for sheath temperature calculations. The three materials investigated are Inconel 600, Inconel 718 and Stainless Steel (SS) 304. These three materials meet the sheath requirements that are necessary in high temperature reactors.

Inconel 600 is a non magnetic alloy that is used in many applications that require high temperature and corrosion resistance. This alloy is applicable for such applications due to the high nickel and chromium content, which provide resistance to corrosion, immunity to chloride-ion-stress corrosion cracking and resistance to oxidizing conditions at high temperatures. Due to these properties, this material is used in applications such as mufflers, exhaust liner, turbine seals and furnace components (Special Metals, 2008).

Similarly, inconel 718 is also used in high temperature applications such as components in liquid-fuel rockets, casings and land-based turbine engines (Ahn et al., 2010). This alloy is used in such applications due to its good corrosion resistance, tensile, fatigue, creep and rupture strength in hostile environments. It
is also important to note that Inconel 718 is high strength corrosion resistant nickel-chromium material that is used in applications that involve temperatures in the range of -252 to 705°C (Special Metals, 2008). This indicates that this material could be a good candidate for use in SCWR since it is used in applications that have temperatures above the coolant outlet temperature of 625°C that can be achieved in SCWRs.

The third material proposed for use as sheath material in SCWRs is the SS 304. This grade of SS is an austenitic steel that contains 16-26% Cr, 8-24% Ni and up to 0.40% C (Kotecki, 2009). As mentioned before, these alloys provide good strength, toughness and oxidation resistance up to temperatures of around 540°C (Kotecki, 2009). However, there are two disadvantages of using SS 304 at high temperatures. First, sensitization can occur at welded heat affected zones which leads to corrosion. Second disadvantage is the possibility of hot cracking of the welded zones. These disadvantages can occur in the temperature range of 427 – 871°C. Table 3 lists the different thermophysical properties of the three proposed sheath materials.

Table 3: Thermophysical Properties of Proposed Sheath Material (Special Metals, 2008 and Technical Handbook of Stainless Steels)

<table>
<thead>
<tr>
<th>Property</th>
<th>Inconel 600</th>
<th>Inconel 718</th>
<th>Stainless Steel 304</th>
</tr>
</thead>
<tbody>
<tr>
<td>Melting Point (°C)</td>
<td>1354 – 1413</td>
<td>1260 – 1336</td>
<td>1399 – 1454</td>
</tr>
<tr>
<td>Density (kg/m³)</td>
<td>8470</td>
<td>8146</td>
<td>8000</td>
</tr>
<tr>
<td>Thermal Conductivity (W/m K)</td>
<td>@100°C – 15.9</td>
<td>@100°C – 12.5</td>
<td>@100°C – 16.2</td>
</tr>
<tr>
<td></td>
<td>@500°C – 22.1</td>
<td>@500°C – 15.5</td>
<td>@500°C – 21.5</td>
</tr>
<tr>
<td></td>
<td>@700°C – 25.7</td>
<td>@700°C – 21.5</td>
<td>-</td>
</tr>
<tr>
<td>Electrical Resistivity (µΩ-m)</td>
<td>1.03</td>
<td>1.03</td>
<td>0.72</td>
</tr>
</tbody>
</table>
The melting temperature is the highest for SS 304 (1399 – 1454°C) then for Inconel 600 (1354 – 1413°C) and for Inconel 718 (1260 – 1336°C). On the other hand, Inconel 600 has the highest thermal conductivity followed by SS 304 and Inconel 718. This indicates that from the heat transfer point of view, the safety margin is maximized while using Inconel 600 since the melting range is similar when comparing Inconel 600 to SS 304, while the thermal conductivity difference is significant between both materials. Also, the electrical resistivity of the Inconel alloys are much higher than SS 304 at approximately 1.03 and 0.72 µΩ-m respectively. Table 4 lists the limiting chemical compositions of the three proposed sheath materials.

**Table 4: Chemical Compositions of the Proposed Sheath Materials (Special Metals (2008), Technical Handbook of Stainless Steels)**

<table>
<thead>
<tr>
<th>Chemical Composition (%)</th>
<th>Inconel 600</th>
<th>Inconel 718</th>
<th>SS 304</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nickel</td>
<td>72 (min)</td>
<td>50-55</td>
<td>8-12</td>
</tr>
<tr>
<td>Chromium</td>
<td>14-17</td>
<td>17-21</td>
<td>18-20</td>
</tr>
<tr>
<td>Iron</td>
<td>6-10</td>
<td>Balance</td>
<td>Balance</td>
</tr>
<tr>
<td>Carbon</td>
<td>0.15 (max)</td>
<td>0.08 (max)</td>
<td>0.08 (max)</td>
</tr>
<tr>
<td>Manganese</td>
<td>1 (max)</td>
<td>0.35 (max)</td>
<td>2.00 (max)</td>
</tr>
<tr>
<td>Sulphur</td>
<td>0.015 (max)</td>
<td>0.015 (max)</td>
<td>0.03 (max)</td>
</tr>
<tr>
<td>Silicon</td>
<td>0.50 (max)</td>
<td>0.35 (max)</td>
<td>-</td>
</tr>
<tr>
<td>Copper</td>
<td>0.50 (max)</td>
<td>0.30 (max)</td>
<td>-</td>
</tr>
<tr>
<td>Niobium</td>
<td>-</td>
<td>4.75-5.50</td>
<td>-</td>
</tr>
<tr>
<td>Molybdenum</td>
<td>-</td>
<td>2.8-3.30</td>
<td>-</td>
</tr>
<tr>
<td>Titanium</td>
<td>-</td>
<td>0.65-1.15</td>
<td>-</td>
</tr>
<tr>
<td>Aluminum</td>
<td>-</td>
<td>0.2-0.8</td>
<td>-</td>
</tr>
<tr>
<td>Cobalt</td>
<td>-</td>
<td>1.00 (max)</td>
<td>-</td>
</tr>
<tr>
<td>Phosphorus</td>
<td>-</td>
<td>0.015 (max)</td>
<td>0.045</td>
</tr>
</tbody>
</table>
Other important considerations when choosing the sheath material are the shear and Young’s Modulus of the material. Figures 18 and 19 show the change in shear and Young’s Modulus of the three proposed sheath materials as the temperature increase.

![Graph showing shear modulus vs temperature for Inconel 600, Inconel 718, and SS 304.](image)

**Figure 18: Shear Modulus of Inconel 600, Inconel 718 and SS 304 vs. Temperature (Special Metals, Technical Handbook of Stainless Steels).**
Figure 19: Young’s Modulus of Inconel 600, Inconel 718 and SS 304 vs. Temperature. (Special Metals, Technical Handbook of Stainless Steels).

The shear and Young’s Modulus for Inconel 600 is the highest, followed by those for Inconel 718 and SS 304. Since materials chosen for SCWRs need to withstand high temperatures, pressures and thermal stresses, it is important to choose a material that has a higher probability of withstanding such conditions. With these conditions in mind, Inconel 600 would be a better candidate from this point of view, since, it has the highest Young’s Modulus, which indicates that it can withstand longer tensile or compressive loads more than the other two sheath-material candidates.

Amongst the three proposed sheath materials, Inconel 600 was chosen as the sheath material for this study. The main reasons of choosing this material over the other two proposed sheath materials are:

- Higher thermal conductivity
- Higher density,
- High melting point
- Higher shear and Young’s Modulus of elasticity.

Therefore, in the process of calculating the fuel centerline temperature, the sheath temperature was calculated using the Inconel 600 thermal conductivity correlation (Eq. 27).

2.7 Proposed Nuclear Fuels

There are several nuclear fuels that are currently being investigated for use in high temperature reactors such as SCWRs. In terms of heat-transfer capability, these nuclear fuels can be classified into three main categories: low thermal-conductivity, enhanced thermal-conductivity and high thermal-conductivity fuels. The low thermal conductivity fuels examined in this study are uranium dioxide (UO$_2$), thorium dioxide (ThO$_2$), and Mixed OXide (MOX) fuel.

The enhanced thermal conductivity fuels examined in this study are uranium dioxide plus silicon carbide (UO$_2$ - SiC) and Uranium Dioxide plus Beryllium Oxide (UO$_2$ - BeO). Also, the high thermal-conductivity fuels include uranium carbide (UC), uranium dicarbide (UC$_2$) and uranium nitride (UN). Some of these high thermal-conductivity fuels have superior properties such as thermal-conductivity, thermal-shock resistance and higher uranium-atom density compared to those of UO$_2$. Consequently, there is a high interest in examining these fuels, especially, for use in high-temperature applications mainly due to their high thermal conductivity, which results in low fuel centerline temperatures.

For the purpose of this study, the fuel centerline temperatures of the aforementioned nuclear fuels enclosed in the proposed 54-element fuel bundle have been calculated. Figure 20 shows the thermal conductivity of several
nuclear fuels as a function of temperature. The next sections will include a description of the properties of each fuel in more details.

As shown in Figure 20, the thermal conductivity of the presented fuels behave differently as the temperature changes. Starting with the high thermal-conductivity fuels, it can be observed that the UC thermal conductivity decreases as temperature increase to around 1000°C then starts to increase. On the other hand, the thermal conductivity of UC₂ and UN increases as temperature increases till up to 2850 – 3000°C. Other enhanced thermal conductivity fuels such as UO₂ - SiC and UO₂ - BeO have higher thermal conductivity, but it decreases significantly as the temperature increase.

Figure 20: Thermal Conductivity of Various Nuclear Fuels (Peiman et al. 2011).
In contrast, low thermal-conductivity fuels such as UO$_2$ and MOX behave similarly. These fuels have a thermal conductivity of about 9 to 10 W/mK, which follows a decreasing trend as temperature increases to approximately 1750°C. Beyond this temperature, the thermal conductivity increases. On the other hand, ThO$_2$ has a higher thermal conductivity than UO$_2$ and MOX; however, its thermal conductivity has a decreasing trend similar to those of UO$_2$ and MOX as a function of temperature to around 1 W/mK at a temperature of 3000°C.

2.7.1 Low Thermal-Conductivity Fuels

Table 5: Properties of UO$_2$, MOX, and ThO$_2$ at 0.1 MPa and 298K (Kirillov et al. 2007)

<table>
<thead>
<tr>
<th>Property</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>MOX</td>
</tr>
<tr>
<td>Thermal Conductivity (W/m K)</td>
<td>7.82$^1$</td>
</tr>
<tr>
<td>Melting Point (°C)</td>
<td>2750</td>
</tr>
<tr>
<td>Linear Expansion Coefficient, (K$^{-1}$)</td>
<td>9.43·10$^{-5}$</td>
</tr>
<tr>
<td>Heat Capacity (J/kg K)</td>
<td>240</td>
</tr>
<tr>
<td>Crystal Structure</td>
<td>FCC</td>
</tr>
</tbody>
</table>

$^1$at 95% density.

UO$_2$

UO$_2$ is the most widely used ceramic fuel in current commercial nuclear reactors such as PWRs, BWRs, and CANDU reactors. There are many advantages and disadvantages of using uranium fuel in nuclear reactors. There are mainly three advantages that motivate using UO$_2$ as a fuel; the low thermal-neutron absorption cross-section, chemical and structural stability. In addition to that, UO$_2$ has a high melting temperature of 2850°C, which provides a good operating and safety margin when UO$_2$ is used. On the other hand, UO$_2$ fuel has some
disadvantages such as the low thermal conductivity, grain growth at high temperatures, stresses and cracking of material which might lead to the release of gaseous fission products (Cochran & Tsoulfanidis, 1999). Several selected properties of UO$_2$ are listed in Table 5.

UO$_2$ fuel might encounter some difficulties if used in SCWRs due to the high operating parameters, which might lead to cracking of the material. This is due to the fact that UO$_2$ fuel is prone to thermal shocks during the power transients, because of its low thermal conductivity. The thermal conductivity of UO$_2$ experiences a trend at which it decreases as temperature increases to around 1750°C, and then starts to increase as shown in Figure 20. In the process of identifying the behaviour of the fuel centerline temperature along the fuel channel while using UO$_2$ as a fuel, the thermal conductivity at 95% Theoretical Density (TD) was calculated using the Frank et al. correlation based on Eq. (6), also, this correlation is valid over a wide range of temperatures of up to 2847°C (Carbajo et al., 2001).

$$k_{UO_2}(T) = \frac{100}{7.5408+17.692\times(10^{-3}T)+3.6142\times(10^{-3}T)^2} + \frac{6400}{(10^{-3}T)^{5/2}} \exp^{-16.35/(10^{-3}T)}$$ (6)

**MOX**

MOX fuel is a combination of plutonium dioxide (PuO$_2$) and UO$_2$ (natural, enriched or depleted depending on the application). The thermal conductivity of MOX fuel depends on mainly four factors; temperature, porosity, oxygen to metal ratio, and burn-up. This type of fuel was originally used in breeder reactors such as the Liquid-Metal Fast Breeder Reactors (LMFBRs), Experimental Breeder Reactor II (EBR-II), Fast Flux Test Facility (FFTF), PHENIX and SUPERPHENIX breeder reactors, and was also used to partially fuel PWR reactors in France (Kirillov et al., 2007).
The MOX fuel composition used in reactors mainly consists of 3 - 5% PuO$_2$ mixed with 95 - 97% UO$_2$ (Popov et al., 2000). Compared to UO$_2$, MOX fuel has lower thermal conductivity, melting temperature and heat of diffusion. Despite of that, the thermal conductivity of MOX fuel behaves in a similar manner as UO$_2$ at which the thermal conductivities of both fuels decrease until they reach approximately 1550 - 1750°C then starts increasing at higher temperatures as shown in Figure 20. The thermal-conductivity of MOX fuel is calculated using the following correlation shown as Eq. (7) (Carbajo et al., 2001). Also, some of the thermophysical properties of MOX fuel are summarized in Table 5:

$$k(T, x) = \frac{1}{A+C(10^{-3}T)} + \frac{6400}{(10^{-3}T)^{5/2}} \exp^{-16.35/(10^{-3}T)}$$  \hspace{1cm} (7)$$

Were x is a function of oxygen to heavy metal ratio ($x = 2 - O/M$)

$A(x)$ and $C(x)$ are calculated using Eqs. (8) and (9):

$$A(x) = 2.58x + 0.035 \ \left(\frac{\text{mK}}{\text{W}}\right)$$  \hspace{1cm} (8)$$

$$C(x) = (-0.715x + 0.286)(\text{m/W})$$  \hspace{1cm} (9)$$

The thermal conductivity correlation shown in Eq. (7) is valid over a wide range of temperatures ranging from 427 - 2827°C. This correlation was chosen to be the best available correlation for predicting the thermal-conductivity of MOX fuel based on analysis done by Fink et al (1999). It is also important to note that there is a ±10% uncertainty accompanied with using this correlation.

**ThO$_2$**

Thorium dioxide fuel or Thoria is proposed as an alternative fuel to be used in nuclear reactors. There are several reasons for choosing thorium fuel as an alternative; one of these reasons is the abundance of thorium which is approximately three times more abundant than uranium in the earth’s crust.
Countries such as India with large thorium reserves are looking forward to
develop nuclear reactors using thorium based fuels. In addition, the use of
thorium based fuels provides good proliferation resistance and is economically
favourable (Tulenko and Baney, 2007).

Thorium itself is not a fissile material; therefore, in order to achieve criticality
and maintain a chain reaction, a fissile material has to be used along with
thorium fuel (e.g. U-235 or Pu-239). In general, there are two options either use a
solid mixture of Thoria and a fissile material or physically separate in different
fuel elements or fuel bundles. When thorium fuel is irradiated, U-233 is
produced, which is a fissile material that can be used in thermal reactors
(Makhijani and Bode, 2009). ThO₂ is another fuel candidate under investigation
for use in SCWRs. Although thorium is a low thermal-conductivity fuel, its
thermal conductivity is slightly higher than that of the UO₂ fuel with a better
chemical stability, less fission product release and lower thermal expansion
coefficient (Carbajo et al., 2001).

Table 5 shows a comparison of selected properties of UO₂, MOX and ThO₂. In
addition, the melting point of ThO₂ is approximately 300°C higher than that of
UO₂. Therefore, the use of this fuel can result in a better operating margin.
Another property which may result in an increased operating margin is the
thermal conductivity. A high thermal conductivity results in a lower fuel
centerline temperature. The thermal conductivity correlation used in the process
of calculating the ThO₂ fuel centerline temperature is shown as Eq. (10).

\[
k = \frac{1}{0.0327 + (1.603 \cdot 10^{-4}T)}
\]  (10)
2.7.2 Enhanced Thermal Conductivity Fuels

**UO$_2$ - SiC**

Currently, UO$_2$ fuel is used in most commercial nuclear reactors with different enrichment levels. The addition of silicon carbide to UO$_2$ in certain percentages enhances the thermal conductivity of the fuel while keeping some of the important UO$_2$ properties such as the high melting temperature of 2850°C. Since SCWRs operate at high temperatures and pressures, a high thermal-conductivity fuel can be a better candidate from the heat transfer point of view. This fuel can be economically ideal since all uranium mining and manufacturing facilities already exist.

It is important to mention that SiC has some unique material properties such as corrosion resistance at high temperatures, low neutron absorption, dimensional stability, and is a non-porous material (Jamila et al., 2009). A study conducted by Tulenko and Baney (2007) showed the thermal conductivity behavior of SiC as a function of temperature. Figure 21 shows the thermal-conductivity behavior of SiC. It can be observed that the SiC can reach very high thermal conductivities of up to approximately 490 W/m·K (Tulenko and Baney, 2007). As shown in Figure 20, the thermal-conductivity of UO$_2$ is only around 8.7 W/m·K at room temperature, however, adding SiC to UO$_2$ fuel can significantly enhance its thermal conductivity to values of around 32 W/m·K at room temperature.
Figure 21: Thermal Conductivity of SiC vs. Temperature

(Tulenko and Baney, 2007).

The thermal conductivity correlation for UO$_2$-SiC fuel was developed by Jamila et al. (2009) for two fuel compositions. These two compositions of the UO$_2$-SiC are the 8% SiC+89% UO$_2$ and 12% SiC+85% UO$_2$. The thermal-conductivity of the 8% SiC+89% UO$_2$ fuel can be calculated using Eq. (11):

\[ k = -1.16 \cdot 10^{-8} \cdot T^3 + 5.03 \cdot 10^{-5} \cdot T^2 - 7.76 \cdot 10^{-2} \cdot T + 49.1 \]  \hspace{1cm} (11)

The second composition examined is the 12% SiC+85% UO$_2$ and is calculated using the following correlation shown as Eq. (12). In Eqs. (11) and (12), $T$ is the temperature in degrees K:

\[ k = -9.59 \cdot 10^{-9} \cdot T^3 + 4.29 \cdot 10^{-5} \cdot T^2 - 6.87 \cdot 10^{-2} \cdot T + 46.8 \]  \hspace{1cm} (12)
The thermal conductivity correlations of the two fuel compositions were calculated using UO$_2$ density of 97% TD and are valid between temperatures of 600 to 1600K (Jamila et al., 2009). The correlation of the 8%SiC+89%UO$_2$ fuel composition was the one used for this study as a conservative approach since it has a lower thermal conductivity compared to 12%SiC+85%UO$_2$ fuel composition and will experience a higher fuel centerline temperature.

**UO$_2$ - BeO**

Another type of enhanced thermal-conductivity fuel considered in this study was the UO$_2$ - BeO fuel, which was proposed by Solomon et al. (2005). The result of their research showed that the addition of 10 vol% of BeO to UO$_2$ resulted in approximately 50% increase in the thermal-conductivity of UO$_2$. Also, some studies have shown that BeO has some attractive properties such as negligible solubility and chemical compatibility with UO$_2$ and most sheath materials, compatibility with CO$_2$ at high temperatures, high burnup and lower fuel cycle costs (Mccoy and Mays, 2008; Solomon et al., 2005).

It is also important to note that BeO has the highest thermal conductivity amongst all oxides and a low neutron absorption cross section. Studies by Mccoy and Mays (2008) have also shown that there was in improvement in fuel temperature, internal rod pressure and fission gas release. Another advantage of using such fuel is the comparable requirements for industrial hygiene to that of UO$_2$ fuel in the processing stage. For the purpose of this study, the thermal-conductivity of UO$_2$-13.6 wt% BeO was used in the process of calculating the fuel centerline temperature. Figure 22 shows the thermal-conductivity of the BeO as a function of temperature. The thermal-conductivity of BeO is around 275 W/m·K at room temperature. Therefore, when looking at Figure 20, it can be seen that adding 13.6 wt% of BeO to UO$_2$ results in a fuel that has a thermal conductivity of around 65 W/m·K at room temperature.
Figure 22: Thermal Conductivity of BeO as a Function of Temperature (Ishimoto et al., 1996)

The study done by McCoy and Mays (2008) shows that UO$_2$ - BeO fuel has an end-of-life burnup of around 72.2 GWd/MTU as opposed to 65 GWd/MTU for UO$_2$ fuel. This indicates that more energy can be extracted from the UO$_2$ - BeO fuel, making it more efficient than UO$_2$ from that perspective.

2.7.3 High Thermal Conductivity Fuels at 1MPa and 298K

Table 6: Properties of UC, UC$_2$, and UN.

<table>
<thead>
<tr>
<th>Property</th>
<th>UC</th>
<th>UC$_2$</th>
<th>UN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Melting Point (°C)</td>
<td>2430$^1$</td>
<td>2375</td>
<td>2850±30</td>
</tr>
<tr>
<td></td>
<td>2532$^2$</td>
<td>2662</td>
<td></td>
</tr>
<tr>
<td>Theoretical Density</td>
<td>13630$^3$</td>
<td>11680</td>
<td>14420</td>
</tr>
<tr>
<td>Thermal Conductivity</td>
<td>21.2</td>
<td>11.57</td>
<td>14.6</td>
</tr>
<tr>
<td>Linear Expansion Coefficient (K$^{-1}$)</td>
<td>$10.1 \times 10^{-6}$</td>
<td>18.1</td>
<td>7.52</td>
</tr>
</tbody>
</table>
Uranium carbide is one of the fuel options proposed for use in high temperature reactors due to its good thermal conductivity and high fissile density. Using a fuel with such characteristics helps in reducing the fuel temperature, hence providing a greater operating margin at high temperatures imposed by such reactors. UC has a face-centered cubic (FCC) structure and a high melting temperature of around 2532°C (De Coninck et al., 1975); therefore, the fuel centerline temperature limit could be taken 1000°C below the melting point at around 1532°C. In addition to the melting point, the thermal conductivity of a fuel is another important parameter, which has a significant impact on the fuel centerline temperature and temperature gradient across the fuel pellets. Therefore, it is necessary to determine and use the thermal conductivity of the fuel as a function of temperature and percent porosity.

Experiments were conducted in order to predict the irradiation behaviour of UC fuel (Arai et al., 1987). These experiments studied the hypostoichiometric, stoichiometric and hyperstoichiometric UC fuels. It was concluded that the when the hypo-stoichiometric carbide fuel is irradiated some precipitating metals in grain boundaries exist, which can affect the fuel integrity.

On the other hand, studies on stoichiometric UC at a temperature range of 1100 – 2250°C and hypo-stoichiometric UC fuel at temperature range of 570-2000°C were conducted (De Coninck et al, 1975). This study measured the thermal diffusivity between temperatures of 600 to 2250°C. Afterwards, the thermal conductivity correlation of UC fuel was derived from the thermal diffusivity.
measurements. The thermal conductivity correlations of stoichiometric and hypo-stoichiometric UC fuels are presented in Eqs. (13) and (14) respectively. In this study, Eq (13) was used to calculate the thermal conductivity of UC.

\[
k = [1.95 \cdot 10^{-1} + 3.57 \cdot 10^{-8} (T - 1123.15)^2]
\]

(13)

\[
k = 100 \cdot [2.04 \cdot 10^{-1} + 2.836 \cdot 10^{-8} (T - 843.15)^2]
\]

(14)

UC₂

Uranium Dicrabide (UC₂) is another fuel candidate that has a relatively high melting temperature at about 2375°C. UC₂ also has a high thermal conductivity and density which allows for better heat transfer capability and mechanical stability. In general, UC₂ is found in the hypo-stoichiometric composition of UC₁.₈ (Frost, 1963). UC₂ fuel was determined to have a body centered tetragonal (BCT) structures up to temperatures of approximately 1760°C. Above this temperature, UC₂ transforms into a face centered cubic (FCC) structure (Tagawa et al., 1971). This variation in lattice parameter can be due to the different compositions that are experienced as the temperature increases.

Due to the above mentioned characteristics of this type of fuel, research was conducted in order to identify the validity of using such fuel in high temperature reactors. Experiments were done using UC₂ in order to identify some of its characteristics such as the thermal diffusivity, thermal conductivity and emissivity as they change with temperature. In the process of calculating the thermal conductivity, the thermal diffusivity was first found using a method called the modulated electron beam technique (De Coninck et al., 1976). The thermal diffusivity is then used with density and specific heat of UC₂ in order to determine the thermal conductivity correlation as a function of temperature. The thermal conductivity correlations found for 5% porosity of the stoichiometric, slightly hypo-stoichiometric and hypo-stoichiometric UC₂ are shown in Eqs. (15) through (17), respectively.
**Stoichiometric**

\[ k = [0.115 + 2.7 \cdot 10^{-5}(T - 273.15) + 2.8 \cdot 10^{-10}(T - 273.15)^2 + 3.035 \cdot 10^{-12}(T - 273.15), \text{ for } 873 < T < 2013 \text{ K}] \]  

(15)

**Slightly Hypo-stoichiometric**

\[ k = 100 \cdot [0.1182 + 2.895 \cdot 10^{-5}(T - 273.15) + 3.8 \cdot 10^{-9}(T - 273.15)^2] + 1.9 \cdot 10^{-12}(T - 273.15)^3, \text{ for } 873 < T < 1993 \text{ K} \]  

(16)

**Hypo-stoichiometric**

\[ k = 100 \cdot [0.132 + 1.9 \cdot 10^{-5}(T - 273.15) + 4.3 \cdot 10^{-9}(T - 273.15)^2], \text{ for } 873 < T < 1993 \text{ K} \]  

(17)

**UN**

Uranium mononitride (UN) is another nuclear fuel that was investigated in this study. This fuel has a high thermal conductivity, high melting temperature at around 2850°C and good irradiation properties. However, some sources indicate that UN changes its composition and free uranium occurs at high temperatures (Balankin et al., 1978). Therefore, it is important to conduct more experimental analysis on UN fuel at high temperatures in order to understand its chemical behaviour if used in high temperature reactors such as SCWRs. It is also important to investigate its thermodynamic properties, specifically thermal conductivity, in an attempt to study the temperature behaviour of such fuel if used in SCWRs.

It can be observed that UN thermal conductivity is around 15 W/m · K at room temperature then starts to increase as temperature increases to about 32 W/m · K at temperatures up to 3000 °C.
Steven et al (1987) developed a correlation, shown as Eq. (18), which predicts the thermal-conductivity of UN fuel. This correlation is valid in the temperature range of 10 to 1923 K with a ± 10% error. This 10% error is due to the deviation between the experimental data and the predictions of the correlation (Steven et al., 1987). In Eq. (18), T is the sheath temperature, and $P_o$ is the percent porosity

$$k = 1.37 \cdot T^{0.41} \left(\frac{1-P_o}{1+P_o}\right)$$  \hspace{1cm} (18)
3.0 METHODOLOGY AND CALCULATIONS

3.1 Geometrical Calculations

The geometrical calculations utilized throughout the process of developing a new fuel bundle consists of angle calculations along with calculating distances between each ring and the elements within each ring. The angles of each element in the first ring was determined by dividing 360° by the number of elements desired in each ring. This process was done while taking into account a minimum gap of 1.5 mm between adjacent elements. The minimum gap of 1.5 mm was chosen in order to provide enough space to insert spacers and prevent direct contact of fuel elements within the bundle.

Furthermore, the distance between each element in different rings was calculated using Eq. (19), in order to define the minimum distance between elements in the different rings.

\[ R = R_1 + \text{Gap}_{\text{ring}1} + R_e + \text{Gap}_{\text{ring}2} + R_e + R_n \]  

Where \( R_1 \) is the radius of the central unheated element, \( \text{Gap}_{\text{ring}1} \) is the gap between the central unheated element and the elements of the first ring, \( R_e \) is the outside radius of the heated elements in the first ring and \( R_n \) is the outside radius of elements in consecutive rings.

The thickness of fuel elements was then identified using Eq. (20):

\[ \delta = D_{c,o} \frac{3 \cdot p \cdot (1-v^2)}{2 \cdot E} \]  

Where, \( v \) is Poisson’s ratio, \( E \) is the Young’s Modulus, and \( p \) is the collapse pressure of the bundle.
After the wall thickness is calculated, the inner diameter of the fuel elements can be found by subtracting twice the wall thickness from the outer diameter. It is notable that the fuel-sheath gap was considered to be negligible.

In the process of designing the fuel bundle, several fuel bundles were developed. Initially, bundles including 54, and 64 elements were designed before choosing the 54-element fuel bundle for this study. Figures 23 and 24 show the initially developed 54 and 64 element fuel bundles.

![Figure 23: 54 Element Fuel Bundle Design](image-url)
Figure 24: 64 Element Fuel Bundle Design

The two fuel bundles were developed while taking into account the pressure tube inner diameter of 103.45 and also a 1.5 mm space between each element within the fuel bundle. Table 7 shows the different parameters used in thermodynamic calculations for the 43, 54 and 64 element fuel bundles.

Table 7: 43, 54 and 64 Element Bundle Parameters.

<table>
<thead>
<tr>
<th></th>
<th>Symbol</th>
<th>Unit</th>
<th>PT - ID 103.45 mm</th>
<th>PT - ID 103.45 mm</th>
<th>PT - ID 103.45 mm</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>OD - 11.5</td>
<td>OD - 9.5</td>
<td>OD - 9.127</td>
</tr>
<tr>
<td>Total Number of Elements</td>
<td>$N_e$</td>
<td>-</td>
<td>43</td>
<td>54</td>
<td>64</td>
</tr>
<tr>
<td>Number of Heated Elements</td>
<td>$N_h$</td>
<td>-</td>
<td>42</td>
<td>53</td>
<td>63</td>
</tr>
<tr>
<td></td>
<td>Nu</td>
<td>-</td>
<td>1</td>
<td>1</td>
<td>1</td>
</tr>
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<td>--------------------------------</td>
<td>-----</td>
<td>-----</td>
<td>----</td>
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<td>Number of Bundles</td>
<td>Nb</td>
<td>-</td>
<td>12</td>
<td>12</td>
<td>12</td>
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<tr>
<td>Wall Thickness</td>
<td>δ</td>
<td>mm</td>
<td>0.4</td>
<td>0.31</td>
<td>0.30</td>
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<tr>
<td>Gap</td>
<td>δ_{gap}</td>
<td>mm</td>
<td>1.5-4.0</td>
<td>1.4-2.0</td>
<td>1.5-1.8</td>
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<td>Pressure</td>
<td>P</td>
<td>MPa</td>
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<td>25</td>
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<td>OD of the Elements</td>
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<td>ID of the Elements</td>
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<td>mm</td>
<td>103.45</td>
<td>103.45</td>
<td>103.45</td>
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<tr>
<td>Bundle Heated Length</td>
<td>L_h</td>
<td>m</td>
<td>0.481</td>
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<tr>
<td>Bundle Fuel Volume</td>
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<td>cm³</td>
<td>1816.57</td>
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<td>1731.9</td>
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<td>Flow Area</td>
<td>A_{flow}</td>
<td>mm²</td>
<td>3728.60</td>
<td>4334.3</td>
<td>3969.3</td>
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<tr>
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<td>P_{wet}</td>
<td>m</td>
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<td>1.97</td>
<td>2.19</td>
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<tr>
<td>Hydraulic- Equivalent Diameter</td>
<td>D_{hy}</td>
<td>mm</td>
<td>7.83</td>
<td>8.80</td>
<td>7.24</td>
</tr>
<tr>
<td>Heated Area Per Bundle</td>
<td>A_{heated}</td>
<td>m²</td>
<td>0.73</td>
<td>0.76</td>
<td>0.87</td>
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<td>Heated Area Per Bundle String</td>
<td>A_{tot}</td>
<td>m²</td>
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<td>9.13</td>
<td>10.43</td>
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<td>Heat Flux</td>
<td>q</td>
<td>kW/m²</td>
<td>970.50</td>
<td>930.99</td>
<td>815.22</td>
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<tr>
<td>Heated Perimeter</td>
<td>P_h</td>
<td>m</td>
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<td>D_h</td>
<td>mm</td>
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<td>10.96</td>
<td>8.79</td>
</tr>
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<td>Heat Generation</td>
<td>q_{gen}</td>
<td>MW/m³</td>
<td>389.93</td>
<td>448.72</td>
<td>408.98</td>
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<td>Mass Flow Rate</td>
<td>m</td>
<td>kg/s</td>
<td>4.37</td>
<td>4.37</td>
<td>4.37</td>
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<td>Mass Flux</td>
<td>G</td>
<td>kg/m²s</td>
<td>1172.02</td>
<td>1008.2</td>
<td>1100.9</td>
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<td>Power Per Element</td>
<td>P</td>
<td>kW/el</td>
<td>16.87</td>
<td>13.36</td>
<td>11.24</td>
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<tr>
<td>Linear Element Rating</td>
<td>q'</td>
<td>kW/m</td>
<td>35.06</td>
<td>27.79</td>
<td>23.38</td>
</tr>
</tbody>
</table>
The fuel bundle used in examining the fuel centerline temperature is a modification of the Variant-20 fuel bundle. The newly developed fuel bundle is a 54-element bundle with a central unheated element of 20 mm in diameter, and outer fuel elements with an outer diameter of 9.5 mm. The geometric orientation of the 54-element fuel bundle is shown in Figure 25.

![Figure 25: 54-Element Fuel Bundle.](image)

The outer diameter of the fuel bundle is kept the same as that of the Variant-20. In other words, this fuel bundle is designed such to fit in a fuel channel with an inner diameter of 103.45 mm. A 3D model of the high efficiency fuel channel containing the 54-element fuel bundle is shown in Figure 26.
3.2 Fuel centerline temperature calculations

The following analysis includes the determination of different parameters such as the bulk-fluid temperature, outer- and inner-sheath surface temperatures, HTC, and the fuel centerline temperature along the heated length (5.772 m) of the fuel channel. MATLAB codes were developed in order to calculate these parameters using an iterative technique. While calculating these parameters, NIST REFPROP software was used to determine the thermophysical properties of water at supercritical conditions. Eq. (21) was used in order to calculate the outer-sheath temperatures along the heated length of the fuel bundle.

\[ T_{o,sh} = \frac{q_{avg}}{HTC} + T_b \]  \hspace{1cm} (21)

Where \( q_{avg} \) is the average uniform heat flux and is calculated using the Eq. (22).
\[ q_{avg} = \frac{Q_{ch}}{\pi D_{e,o} L_h N} \]  \hspace{1cm} (22)

\( Q_{ch} \) is the average channel power of 8.5 MW, \( N \) is the number of fuel elements in a fuel bundle, \( L_h \) is the heated length of 5.772 m and \( D_{e,o} \) is the outside diameter of heated fuel elements. In the process of calculating the outer-sheath temperature, the HTC has to be calculated.

The HTC was calculated using the Mokry et al. correlation (Eq. (3)), were the average Prandtl number (\( \overline{Pr} \)) and the average specific heat (\( \overline{C_p} \)) are calculated using the following Eqs. (23) and (24).

\[ \overline{Pr} = \frac{\overline{C_p} \cdot \mu_b}{k_b} \]  \hspace{1cm} (23)

\[ \overline{C_p} = \frac{H_w - H_b}{\tau_w - \tau_b} \]  \hspace{1cm} (24)

Where \( k_b \) and \( \mu_b \) are the thermal conductivity and the dynamic viscosity of the fluid respectively and are determined using NIST REFPROP. Alternately, there are two unknowns while calculating the HTC correlation, the sheath temperature (\( T_w \)) and the HTC. Therefore, iterations are needed in order to determine the HTC value. To start these iterations, the sheath temperature was guessed to be 50°C above the bulk-fluid temperature as an initial guess. Afterwards, iterations took place using a MATLAB code until the difference between the bulk fluid temperature and the outer sheath temperature is less than 1°C. Also, \( H_b \) is the bulk fluid enthalpy (J/kg) and is calculated using Eq. (25). Where \( p \) is the heated parameter (m), \( q \) is the heat flux (W/m²), \( \dot{m} \) is the mass flow rate (kg/s) and \( \Delta x \) is the length increment.

\[ h_b = h_{b-1} + \frac{p \cdot q}{\dot{m}} \cdot \Delta x \]  \hspace{1cm} (25)
After determining the outer sheath temperature, the inner-sheath temperature \((T_{i,sh})\) is calculated using Eq. (26):

\[
T_{i,sh,x} = \frac{q_{avg} D_{e,o} \ln\left(\frac{r_o}{r_i}\right)}{2 \cdot k_{sh,x}} + T_{o,sh}
\]  

(26)

Where \(r_i\) and \(r_o\) are the inner and outer radii of the fuel rod (i.e., sheath), \(D_{e,o}\) is the outer diameter of the fuel rod, \(q_{avg}\) is the average heat flux and is calculated using Eq. (22). In addition, the thermal conductivity of the sheath \((k)\) is calculated using Eq. (27) assuming Inconel 600 as the sheath material, the outer sheath temperature \((T_{o,sh})\) calculated using Eq. (21) was used to calculate the thermal conductivity of the sheath material along the heated length of the fuel bundle (Kirillov et al., 2005).

\[
k = 8.116 + 0.0176 \cdot T_{o,sh}
\]  

(27)

After calculating the inner sheath temperature, the fuel centerline temperature can be then calculated using Eq. (28).

\[
T_{b+1,fuel} = T_{b,fuel} + \frac{\dot{e}_{gen}(r_b^2 - r_{b+2}^2)}{4k_{fuel}}
\]  

(28)

Where, the heat generation \((\dot{e}_{gen})\) is calculated using Eq. (29).

\[
\dot{e}_{gen} = \frac{4Q_{ch}}{V_{fuel}}
\]  

(29)

\(V_{fuel}\) is the volume of fuel in the bundle string \((m^3)\), \(Q_{ch}\) is assumed to be the average channel power of 8.5 MW in all fuel channels, \(D_{i,e}\) is the inner diameter of the fuel rod \((m)\), and \(q_{x}\) is the heat flux \((W/m^2)\) calculated using Eq. (30).
\[ q_x = q_{av_g} \cdot (b_0 + b_1(x) + b_2(x)^2 + b_3(x)^3 + b_4(x)^4 + b_5(x)^5 + b_6(x)^6 + b_7(x)^7) \]  

(30)

Where \( b_1 - b_7 \) are the power coefficients used to generate different AHFPs. Non-uniform cosine type AHFPs power ratios are based on the coefficients obtained from the Figure 27 based on the results of Leung et al. (2008).

![Figure 27: AHFPs at 8.5 MW (Average Channel Power) (based on paper by Leung et al. 2008).](image)

The fuel centerline temperatures for the different fuels are then calculated iteratively at 50 points along the radius of a fuel pellet using MATLAB programming. Also, a perfect contact between fuel pellets and the sheath was
assumed during the process of calculating the fuel centerline temperature. A summary of the methodology is shown in Figure 28.

\[
h_{b+1} = h_b + \frac{p \cdot q_x}{m} \Delta x
\]

\[
T_{\text{wall,o}} = T_{\text{coolant}} + \frac{q}{H}
\]

\[
\text{Nu} = \frac{H \cdot D_{hy}}{k}
\]

\[
Q = \frac{T_{\text{wall,i}} - T_{\text{wall,o}}}{R}
\]

\[
R = \frac{\ln\left(\frac{r_o}{r_i}\right)}{2\pi L k}
\]

\[
T_{r,b+1} = T_{r,b} + \frac{Q_{\text{gen}}(r_b^2 - r_{b+1}^2)}{4k_{\text{avg}}}
\]

Figure 28: Methodology of Fuel Centerline Temperature Calculations.
4.0 RESULTS AND DISCUSSION

A steady-state one-dimensional heat transfer analysis was conducted using MATLAB to calculate the fuel centerline temperature. The heat transfer coefficient, sheath temperature, and centerline temperatures of various nuclear fuels were calculated at four Axial Heat Flux Profiles (AHFPs). The four AHFPs are the uniform, cosine, downstream-skewed, upstream-skewed profiles. Calculations were based on the following SCW parameters: pressure of 25 MPa, coolant inlet temperature of 350°C, average mass flow rate of 4.4 kg/s and the 54-element bundle geometry shown in Figure 25.

The results obtained in this section do not take into account the effect of the gap between the fuel and sheath. In other words, it was assumed that there was no gap between the fuel pellets and the sheath material. It was also assumed that all SCW fuel channels in the reactor core have a channel power of 8.5 MWth. Therefore, the results were obtained for one SCW fuel channel and it was assumed that other fuel channels have the same average channel power.

There are eight nuclear fuels presented in this study. The chosen nuclear fuels are a variety of low, enhanced and high thermal conductivity fuels. The low thermal conductivity fuels examined in this study are UO2, MOX and ThO2. The enhanced thermal conductivity fuels are UO2 - SiC and UO2 - BeO, while the high thermal conductivity fuels are UC, UC2, and UN. The analysis for this study were obtained for all of the above mentioned AHFPs, however, this section will only include the downstream-skewed AHFPs since it showed the highest fuel centerline temperatures amongst all the other temperature profiles. The uniform, cosine, and upstream-skewed AHFPs can be found in Appendix B.

The Mokry et al. correlation was used in the process of calculating the HTC. This correlation was proven by the University of Ottawa to be the correlation that best predicts the behaviour of SCW, however, there is a 25% error in this calculation.
Therefore, a sensitivity analysis was conducted, where the fuel centerline and sheath temperatures were calculated based on the HTC found by the original Mokry et al. correlation and when the HTC values were doubled. This process was followed in order to find the amount of reduction in the sheath and fuel centerline temperatures if the HTC calculated value is doubled.

Figures 29 through 36 show the HTC, bulk-fluid temperature, sheath temperature, and the fuel centerline temperature of the examined fuels starting from low to high thermal-conductivity fuels.
Figure 29: HTC, Sheath Temperature, and Fuel Centerline, UO₂ Fuel.

Figure 30: HTC, Sheath Temperature, and Fuel Centerline, MOX Fuel.
Figure 31: HTC, Sheath Temperature, and Fuel Centerline, Thoria fuel.

Figure 32: HTC, Sheath Temperature, and Fuel Centerline, UO$_2$ - SiC Fuel.
**Figure 33**: HTC, Sheath Temperature, and Fuel Centerline, UO$_2$-BeO Fuel.

**Figure 34**: HTC, Sheath Temperature, and Fuel Centerline, UC$_2$ Fuel.
Heated Length, m
Temperature, °C
0
300
600
900
1200
1500
1800
2100
2400
HTC, kW/m²K
5
10
15
20
25
1
... MPa G =1025 kg/m²s
Qch = 8.5 MW, qavg = 914 kW/m²
UC, Downstream-Skewed AHFP
Dhy =8.5 mm

Figure 35: HTC, Sheath Temperature, and Fuel Centerline, UC Fuel.

Heated Length, m
Temperature, °C
0
300
500
700
900
1100
1300
1500
1700
1900
HTC, kW/m²K
5
10
15
20
25
1
... MPa G =1025 kg/m²s
Qch = 8.5 MW, qavg = 914 kW/m²
UN, Down Skewed Cosine AHFP
Dhy =8.5 mm
HTC ( Mokry et al. Corr.)

Figure 36: HTC, Sheath Temperature, and Fuel Centerline, UN Fuel.
As previously mentioned, the HTC, sheath temperature, and fuel centerline temperature were calculated for the 54-element fuel bundle at supercritical conditions (outlet temperature of 625°C and 25 MPa pressure). In all cases, the sheath temperature was calculated assuming Inconel 600 as the sheath material. Furthermore, the maximum sheath temperatures found were around 800°C while using the downstream skewed AHFP. Therefore, the design limit of 850°C was not exceeded in any of the AHFPs, however, the maximum sheath temperature was only 50°C below the design limit of Inconel 600.

In regards to the fuel centerline temperatures, the industry accepted limit was exceeded while using MOX as a fuel enclosed in the 54-element fuel bundle and examining the downstream-skewed AHFP. This was due to the very low thermal-conductivity of the MOX fuel. Therefore, according to the analysis done in this study, MOX fuel might not be a viable candidate for use in SCWR since the maximum fuel centerline temperature exceeds the limit of 1850°C.

Consequently, all the examined fuels (excluding MOX) experienced fuel centerline temperatures that were predicted to be below the industry accepted limits. The fuel centerline temperature results are promising when high thermal conductivity fuels were examined, however, they increase as the thermal conductivity of fuels decrease. The fuel centerline temperature limits are taken to be around 1000°C less than the fuel melting temperature. Therefore, UO₂, MOX, ThO₂, UO₂ - SiC and UO₂ - BeO are considered in this study to have a temperature limit of 1850°C, while UC, UC₂ and UN had a temperature limit of 1500°C since they experience a lower melting temperature.

It was also observed from Figures 29 through 36 and Table 8 that MOX fuel has the highest fuel centerline temperature (1869°C), which was the only fuel that exceeded the fuel centerline temperature limit. The second highest fuel centerline temperature observed was 1758°C for UO₂ fuel, followed by Thoria at
temperature of 1350°C. The enhanced thermal conductivity fuels such as the UO₂ - BeO and UO₂ - SiC experienced lower centerline temperatures of around 1150°C and 1100°C respectively.

Alternately, high thermal conductivity fuels experienced fuel centerline temperatures that were significantly lower than the previously mentioned fuels, however, as mentioned previously, the fuel centerline temperature limits for such fuels are less than the low and enhanced thermal conductivity fuels. For example, using UN as a fuel results in the lowest fuel centerline temperature at around 945°C, followed by UC and UC₂ fuels at around 1000 and 1045°C respectively, but their fuel centerline temperature limit was taken to be 1500°C.

**Table 8: Maximum Fuel Centerline Temperatures of Different Nuclear Fuels**

<table>
<thead>
<tr>
<th>AHFP</th>
<th>Uniform</th>
<th>Cosine</th>
<th>Upstream-skewed</th>
<th>Downstream-skewed</th>
</tr>
</thead>
<tbody>
<tr>
<td>Max. Fuel CLT °C</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>UO₂</td>
<td>1553</td>
<td>1641</td>
<td>1468</td>
<td>1758</td>
</tr>
<tr>
<td>MOX</td>
<td>1636</td>
<td>1744</td>
<td>1558</td>
<td>1869</td>
</tr>
<tr>
<td>ThO₂</td>
<td>1328</td>
<td>1362</td>
<td>1228</td>
<td>1459</td>
</tr>
<tr>
<td>UO₂ - SiC</td>
<td>1050</td>
<td>1035</td>
<td>890</td>
<td>1125</td>
</tr>
<tr>
<td>UO₂ - BeO</td>
<td>1082</td>
<td>1070</td>
<td>991</td>
<td>1145</td>
</tr>
<tr>
<td>UC</td>
<td>957</td>
<td>920</td>
<td>873</td>
<td>1000</td>
</tr>
<tr>
<td>UC₂</td>
<td>1010</td>
<td>990</td>
<td>929</td>
<td>1046</td>
</tr>
<tr>
<td>UN</td>
<td>937</td>
<td>894</td>
<td>850</td>
<td>945</td>
</tr>
</tbody>
</table>

The heat-transfer analysis presented in this study concluded that the fuel centerline temperatures and the sheath temperatures do not exceed their industry and design accepted limits when enclosed in the proposed 54-element fuel bundle except while using MOX fuel in the Downstream-skewed AHFP.
Despite of the chemical restrictions on each one of the fuels, it can also be concluded that almost all the fuels can withstand the high operating parameters of the SCWRs from a heat transfer point of view. The only difference would be the increased operating margin when high thermal conductivity fuels are used as opposed to the low thermal conductivity ones. Furthermore, Figures (37) to (44) will be used to analyse the change in fuel centerline and sheath temperatures as the HTC is doubled. The following sensitivity analysis would help in understanding how changes in the HTC impact the fuel centerline and sheath temperatures.
Figure 37: HTC*2, Sheath Temperature, and Fuel Centerline, UO₂ Fuel.

Figure 38: HTC*2, Sheath Temperature, and Fuel Centerline, MOX Fuel.
Figure 39: HTC*2, Sheath Temperature, and Fuel Centerline, ThO2 Fuel.

Figure 40: HTC*2, Sheath Temperature, and Fuel Centerline, UO2 - SiC Fuel.
Figure 41: HTC*2, Sheath Temperature, and Fuel Centerline, UO₂-BeO Fuel.

Figure 42: HTC*2, Sheath Temperature, and Fuel Centerline, UC₂ Fuel.
Figure 43: HTC*2, Sheath Temperature, and Fuel Centerline, UC Fuel.

Figure 44: HTC*2, Sheath Temperature, and Fuel Centerline, UN Fuel.
Figures 37 – 44 represent the downstream-skewed AHFPs for the eight nuclear fuels proposed in this study. In this case, the downstream-skewed profiles were generated while the calculated HTC value was doubled. Doubling the HTC resulted in decreasing the fuel centerline temperature by different increments as the fuels change. Table 9 shows the calculated maximum fuel centerline temperatures for the different nuclear fuels as the HTC value was doubled.

Table 9: Maximum Fuel Centerline Temperatures of Different Nuclear Fuels at HTC*2

<table>
<thead>
<tr>
<th>AHFP</th>
<th>Uniform</th>
<th>Cosine</th>
<th>Upstream-skewed</th>
<th>Downstream-skewed</th>
</tr>
</thead>
<tbody>
<tr>
<td>Max. Fuel CLT °C (@HTC x 2)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>UO₂</td>
<td>1381</td>
<td>1345</td>
<td>1249</td>
<td>1463</td>
</tr>
<tr>
<td>MOX</td>
<td>1455</td>
<td>1424</td>
<td>1314</td>
<td>1547</td>
</tr>
<tr>
<td>ThO₂</td>
<td>1177</td>
<td>1114</td>
<td>1050</td>
<td>1199</td>
</tr>
<tr>
<td>UO₂ - SiC</td>
<td>942</td>
<td>849</td>
<td>824</td>
<td>884</td>
</tr>
<tr>
<td>UO₂ - BeO</td>
<td>961</td>
<td>864</td>
<td>939</td>
<td>936</td>
</tr>
<tr>
<td>UC</td>
<td>848</td>
<td>775</td>
<td>756</td>
<td>818</td>
</tr>
<tr>
<td>UC₂</td>
<td>906</td>
<td>843</td>
<td>816</td>
<td>890</td>
</tr>
<tr>
<td>UN</td>
<td>832</td>
<td>748</td>
<td>754</td>
<td>798</td>
</tr>
</tbody>
</table>

While comparing the results from Table 8 and Table 9, it was concluded that the fuel centerline temperature was significantly smaller in most cases. For example, MOX fuel was affected the most since the maximum fuel centerline temperature decreased from 1869°C to 1547 resulting in a temperature difference of around 322°C while both examined using the downstream-skewed AHFP. The least affected fuels were the composite fuels (e.g. UO₂ - SiC and UO₂ - BeO) where the temperature difference in the upstream-skewed AHFP was found to be 66 and
54°C respectively. Table 10 shows the temperature difference between the nuclear fuels as a result of changing the HTC.

**Table 10: Maximum Fuel Centerline Temperature Difference Between Results Obtained While Using HTC and HTC x 2.**

<table>
<thead>
<tr>
<th>AHFP</th>
<th>Uniform</th>
<th>Cosine</th>
<th>Upstream</th>
<th>Downstream</th>
</tr>
</thead>
<tbody>
<tr>
<td>UO₂</td>
<td>172</td>
<td>296</td>
<td>219</td>
<td>295</td>
</tr>
<tr>
<td>MOX</td>
<td>181</td>
<td>320</td>
<td>244</td>
<td>322</td>
</tr>
<tr>
<td>ThO₂</td>
<td>151</td>
<td>248</td>
<td>178</td>
<td>260</td>
</tr>
<tr>
<td>UO₂ - SiC</td>
<td>108</td>
<td>186</td>
<td>66</td>
<td>241</td>
</tr>
<tr>
<td>UO₂ - BeO</td>
<td>121</td>
<td>206</td>
<td>52</td>
<td>209</td>
</tr>
<tr>
<td>UC</td>
<td>109</td>
<td>145</td>
<td>117</td>
<td>182</td>
</tr>
<tr>
<td>UC₂</td>
<td>104</td>
<td>147</td>
<td>113</td>
<td>156</td>
</tr>
<tr>
<td>UN</td>
<td>105</td>
<td>146</td>
<td>96</td>
<td>147</td>
</tr>
</tbody>
</table>
5.0 CONCLUDING REMARKS

SuperCritical Water-Cooled Reactors operate at high temperatures and pressures compared to those of current nuclear reactors; therefore, there is a need to develop new fuel bundle designs and study different nuclear fuels. It is essential to investigate new designs and fuels in order to prevent over heating of the fuel and the fuel bundle at the SCWR operating conditions. A new 54-element fuel bundle design has been developed in an attempt to decrease the sheath and the fuel centerline temperature at supercritical water conditions. This design contains 53 heated elements and 1 central un-heated element with diameters of 9.5 mm and 20 mm respectively.

The fuels examined in this study are UO₂, MOX, ThO₂, UO₂ - SiC, UO₂ - BeO, UC, UC₂ and UN. Temperature profiles were generated for each of these fuels in order to identify their sheath and fuel centerline temperature behaviour when enclosed in the 54-element fuel bundle at SCW conditions. It was noticed that the sheath temperature did not exceed the design limit of 850°C for Inconel 600. Alternately, the fuel centerline temperature was examined for each of the above mentioned fuels at four Axial Heat Flux Profiles (AHFPs). The four AHFPs examined are the uniform, cosine, upstream-skewed, and downstream-skewed profiles. When the four AHFPs were examined, the MOX fuel was the only fuel that exceeded the fuel centerline temperature limit of 1850°C. Other low thermal-conductivity fuels such as UO₂ and ThO₂ experienced high fuel centerline temperature but are still below the industry accepted limit. Also, high thermal-conductivity fuels (i.e., UC, UC₂, and UN) and enhanced thermal conductivity fuels (i.e., UO₂ - SiC, and UO₂ - BeO), temperature limits were set to be 1000°C below their melting temperatures. Nevertheless, the established limits were not exceeded in this case. Also, the HTC value obtained from the Mokry et al. correlation was doubled in order to investigate the amount of decrease in fuel
centerline temperature that would result from such change. It was noticed that
the low thermal conductivity fuels were affected the most by changing the HTC.
For example MOX fuel, which has the lowest thermal conductivity, experienced
a fuel centerline temperature that was 320°C less after doubling the HTC value.
In contrast, a high thermal conductivity fuel such as UN, experienced a change of
only 147°C after HTC values were doubled.

Finally, the 54-element fuel bundle showed promising sheath temperature
profiles and the proposed fuels (excluding MOX fuel) experienced fuel centerline
temperatures that did not exceed the established temperature limits. The
enhanced and high thermal-conductivity fuels result in lower fuel centerline
temperatures, therefore, from this point of view; the operating margin is
increased when using enhanced and high thermal conductivity fuels enclosed in
the 54-element fuel bundle. Thus, the 54-element fuel bundle and the proposed
fuels can be promising candidates for use in SCWRs.
6.0 FUTURE WORK

Future work may include examining different fuels at maximum channel power. Also, incorporating changes in coolant pressure and taking into account changes in sheath and fuel centerline temperatures due to neutron bombardment. Also, some of the assumptions made in this study can be taken into consideration such as the fuel sheath gap.

Additionally, it is also important to develop a neutronic code. This neutronic code is necessary in order to identify the in-core power distribution. Alternately, fuel bundles with higher number of elements or different geometrical representations can be investigated along with different sheath material and fuel compositions.
References


Zahlan, H., Groeneveld, D.C., Tavoularis, S., “Look-up table for trans-critical heat transfer”, the 2nd Canada-China Joint Workshop on Supercritical Water-Cooled Reactors (CCSC-2010), Toronto, Canada, April 25-28, 2010, 18 pages
Appendix A – Uniform, Cosine and Upstream-skewed AHFP Graphs

UO$_2$

Figure 45: HTC, Sheath Temperature, and Fuel Centerline, UO$_2$ Fuel.

Figure 46: HTC*2, Sheath Temperature, and Fuel Centerline, UO$_2$ Fuel.
Figure 47: HTC, Sheath Temperature, and Fuel Centerline, UO₂ Fuel.

Figure 48: HTC*2, Sheath Temperature, and Fuel Centerline, UO₂ Fuel.
Figure 49: HTC, Sheath Temperature, and Fuel Centerline, UO$_2$ Fuel.

Figure 50: HTC*2, Sheath Temperature, and Fuel Centerline, UO$_2$ Fuel.
MOX

Figure 51: HTC, Sheath Temperature, and Fuel Centerline, MOX Fuel.

Figure 52: HTC*2, Sheath Temperature, and Fuel Centerline, MOX Fuel.
Heated Length, m
Temperature, °C
300
500
700
900
1100
1300
1500
1700
1900

HTC, kW/m²K
5
10
15
20
25

Figure 53: HTC, Sheath Temperature, and Fuel Centerline, MOX Fuel.

Figure 54: HTC*2, Sheath Temperature, and Fuel Centerline, MOX Fuel.
Figure 55: HTC, Sheath Temperature, and Fuel Centerline, MOX Fuel.

Figure 56: HTC*2, Sheath Temperature, and Fuel Centerline, MOX Fuel.
ThO$_2$

Figure 57: HTC, Sheath Temperature, and Fuel Centerline, ThO$_2$ Fuel.

Figure 58: HTC*2, Sheath Temperature, and Fuel Centerline, ThO$_2$ Fuel
Figure 59: HTC, Sheath Temperature, and Fuel Centerline, ThO₂ Fuel

Figure 60: HTC*2, Sheath Temperature, and Fuel Centerline, ThO₂ Fuel
Figure 61: HTC, Sheath Temperature, and Fuel Centerline, ThO$_2$ Fuel

Figure 62: HTC*2, Sheath Temperature, and Fuel Centerline, ThO$_2$ Fuel
UO₂ - SiC

![Graph of HTC, Sheath Temperature, and Fuel Centerline, UO₂ - SiC Fuel](image)

**Figure 63:** HTC, Sheath Temperature, and Fuel Centerline, UO₂ - SiC Fuel

![Graph of HTC*2, Sheath Temperature, and Fuel Centerline, UO₂ - SiC Fuel](image)

**Figure 64:** HTC*2, Sheath Temperature, and Fuel Centerline, UO₂ - SiC Fuel
Figure 65: HTC, Sheath Temperature, and Fuel Centerline, UO₂ - SiC Fuel

Figure 66: HTC*2, Sheath Temperature, and Fuel Centerline, UO₂ - SiC Fuel
Figure 67: HTC, Sheath Temperature, and Fuel Centerline, UO₂ - SiC Fuel

Figure 68: HTC*2, Sheath Temperature, and Fuel Centerline, UO₂ - SiC Fuel
Figure 69: HTC, Sheath Temperature, and Fuel Centerline, UO$_2$ - BeO Fuel

Figure 70: HTC*2, Sheath Temperature, and Fuel Centerline, UO$_2$ - BeO Fuel
Figure 71: HTC, Sheath Temperature, and Fuel Centerline, UO₂ - BeO Fuel

Figure 72: HTC*2, Sheath Temperature, and Fuel Centerline, UO₂ - BeO Fuel
Figure 73: HTC, Sheath Temperature, and Fuel Centerline, UO₂ - BeO Fuel

Figure 74: HTC*2, Sheath Temperature, and Fuel Centerline, UO₂ - BeO Fuel
**Figure 75: HTC, Sheath Temperature, and Fuel Centerline, UC Fuel**

**Figure 76: HTC*2, Sheath Temperature, and Fuel Centerline, UC Fuel**
**Figure 77:** HTC, Sheath Temperature, and Fuel Centerline, UC Fuel

**Figure 78:** HTC*2, Sheath Temperature, and Fuel Centerline, UC Fuel
Figure 79: HTC, Sheath Temperature, and Fuel Centerline, UC Fuel

Figure 80: HTC*2, Sheath Temperature, and Fuel Centerline, UC Fuel
**Figure 81: HTC, Sheath Temperature, and Fuel Centerline, UC₂ Fuel**

**Figure 82: HTC*2, Sheath Temperature, and Fuel Centerline, UC₂ Fuel**
Figure 83: HTC, Sheath Temperature, and Fuel Centerline, UC₂ Fuel

Figure 84: HTC*2, Sheath Temperature, and Fuel Centerline, UC₂ Fuel
Figure 85: HTC, Sheath Temperature, and Fuel Centerline, UC₂ Fuel

Figure 86: HTC*2, Sheath Temperature, and Fuel Centerline, UC₂ Fuel
Figure 87: HTC, Sheath Temperature, and Fuel Centerline, UN Fuel

Figure 88: HTC*2, Sheath Temperature, and Fuel Centerline, UN Fuel
Figure 89: HTC, Sheath Temperature, and Fuel Centerline, UN Fuel

Figure 90: HTC*2, Sheath Temperature, and Fuel Centerline, UN Fuel
Figure 91: HTC, Sheath Temperature, and Fuel Centerline, UN Fuel

Figure 92: HTC*2, Sheath Temperature, and Fuel Centerline, UN Fuel
Appendix B – MATLAB Codes

Automated Bundle Design Code:

gapring=input('please input a value for the gap between the core element and the first ring in millimeters, greater than 1.5 mm: '); %This allows the user to input the gap between the unheated center element and the first ring of elements
if gapring<1.5; %An if statement that runs only if gapring is less than 1.5
    error 'gap must be greater than 1.5mm' %If the above statement is true, then an error message is shown since the gap must be between 1.5 and 1.8
elseif gapring>1.8; %An elseif statement that only runs if gapring is greater than 1.8
    error 'gap must be between 1.5mm and 1.8 mm' %If the above statement is true, an error message is shown stating that the gap must be between 1.5 and 1.8
else %else statement corresponding to the above if statement
    gapring=gapring; %if gapring is between 1.5 and 1.8, set gapring equal to gapring
end

gap_e=input('please input a value for the gap between elements in millimeters, greater than 1.5 mm: '); %This allows the user to input the gap that they require between elements
if gap_e<1.5; %An if statement that runs only if gap_e is less than 1.5
    error 'gap must be greater than 1.5mm' %If the above statement is true, then an error message is shown since the gap must be between 1.5 and 1.8
elseif gap_e>1.8; %An elseif statement that only runs if gap_e is greater than 1.8
    error 'gap must be between 1.5mm and 1.8 mm' %If the above statement is true, an error message is shown stating that the gap must be between 1.5 and 1.8
else %else statement corresponding to the above if statement
    gap_e=gap_e; %if gap_e is between 1.5 and 1.8, set gap_e equal to gap_e
D1=20; %Diameter of the center unheated element
D2=103.45; %Diameter of the pressure tube
R1=D1/2; %Radius of the center element
R2=D2/2; %Radius of the pressure tube
R3=(R2-(gap_e/2)); %Radius of the pressure tube buffer
R4=(R1)+(gapring/2); %Radius of the buffer of the center element
D_e=input('please specify element diameter in millimeters greater than 7.5mm: ');
if D_e>80.4
    error 'elements will not fit inside the pressure tube!' %If the above if statement is true, and error message is displayed stating that the elements will not fit inside the pressure tube
elseif D_e<7.5
    error 'elements are smaller than 7.5 mm')
else
    D_e=D_e; %If the above if statements are false, then set D_e equal to D_e
end
[x,y,z] = cylinder(R2,200);
plot(x(1,:),y(1,:)); %Plots the pressure tube
hold on
[x,y,z]=cylinder(R1,200);
plot(x(1,:),y(1,:)); %Plots the center element
hold on
[x,y,z]=cylinder(R3,200);
plot(x(1,:),y(1,:),'r'); %Plots the pressure tube buffer
hold on
[x,y,z]=cylinder(R4,200);
plot(x(1,:),y(1,:),'r') %Plots the buffer of the center element
hold on
R_e=D_e/2; %Defines the radius of the elements
R_ring1=R1+gapring+R_e; %Defines the radius of the first ring
ringradius(1)=R_ring1;
[x,y,z]=cylinder(R_ring1,200);
plot(x(1,:),y(1,:),'g') %Plots the first ring
N=3; %Initially set N, the number of elements per ring, equal to 3
test=0; %Set test equal to 0
while test==0 %This is a while loop that only runs while the test is equal to 0
    phi=(2*pi/N); %The angle increment of the elements
    ex=sqrt((2*(R_ring1^2))-(2*(R_ring1^2)*cos(phi))); %The straight line distance between two elements
    if ex<(D_e+gap_e) %If the straight line distance between two elements is less than the diameter of an element plus a gap, execute the following lines
        N=N-1; %Set the number of elements, N, equal to the number of elements minus 1
        test=1; %Set test equal to 1
    else %If the above if statement is false, execute the following lines
        N=N+1; %Set the number of elements, N, equal to the number of elements plus 1
    end
end
inc=2*pi/N; %Increment of the angle between elements
for c=1:N %A for loop for when c is from 1 to N
    if c==1 %When c is equal to 1, execute the following 3 lines of code
        [x,y,z]=cylinder(R_e,200);
        plot(x(1,:)+R_ring1,y(1,:)) %Plots the first element in the ring
        hold on
else % When c is equal to everything else but 1
    [x,y,z]=cylinder(R_e,200);
    plot(x(1,:)+(R_ring1*cos((c-1)*inc)),y(1,:)+(R_ring1*sin((c-1)*inc))) % Plots the rest of the elements in that ring taking into account the angle increment
    hold on
end
end

for c=1:N % A for loop for when c is from 1 to N
    if c==1 % When c is equal to 1, execute the following 3 lines of code
        [x,y,z]=cylinder(R_e+(gap_e/2),200);
        plot(x(1,:)+R_ring1,y(1,:),r) % Plots the first elements buffer in the ring
        hold on
    else
        [x,y,z]=cylinder((R_e+(gap_e/2)),200); % When c is equal to everything else but 1
        plot(x(1,:)+(R_ring1*cos((c-1)*inc)),y(1,:)+(R_ring1*sin((c-1)*inc)),r) % Plots the rest of the elements buffers in that ring taking into account the angle increment
        hold on
    end
end

R_ring=R_ring1; % Sets R_ring equal to R_ring1
R_ringold=R1; % Defines R_ringold to be R1, which is equal to the radius of the unheated center element
R_ringold2=R1; % Defines R_ringold2 to be R1, which is equal to the radius of the unheated center element
R_ringold3=R1; % Defines R_ringold3 to be R1, which is equal to the radius of the unheated center element

stop=15; % Sets the stop condition to 15
\[ u = D_e + \text{gap}_e; \] \% Sets \( u \) equal to \( D_e + \text{gap}_e \), where \( u \) is the smallest distance possible between elements of different rings

\[ n = 0; \] \% Sets \( n \) equal to 0

\textbf{for} ring = 2:stop \% A for loops that runs while the variable \( \text{ring} \) goes from 2 to the stop condition which was set above as 15

\[ \beta = \arcsin(R_{\text{ring}} \sin(\pi/N)/u); \] \% Defines an angle using the sine law

\[ \gamma = \pi - \beta - \pi/N; \] \% Defines an angle using the law that states that all angles in a triangle add to 180 degrees, or \( \pi \)

\[ R_{\text{ring}} = (u \sin(\gamma)/\sin(\pi/N)); \] \% Defines the radius of the most recent ring, using the sine law

\textbf{if} (R_{\text{ring}} - R_{\text{ringold2}}) < (D_e + \text{gap}_e) \% If the radius of the most recent ring minus the radius of two rings before it is less than a diameter of an element plus a gap, execute the following line

\[ R_{\text{ring}} = R_{\text{ringold2}} + D_e + \text{gap}_e; \] \% If the above if statement is true, set the radius of the most recent ring equal to the radius of two rings before it plus the diameter of an element plus a gap

\textbf{else}

\[ R_{\text{ring}} = R_{\text{ring}}; \] \% If the above if statement is false, set the radius of the most recent ring equal to the radius of the most recent ring that was defined above the above if statement

\textbf{end}

\textbf{if} ring == 3 \% And if statement for when the counter 'ring' is equal to 3

\textbf{if} (R_{\text{ring}} - R_{\text{ring}1}) < (D_e + \text{gap}_e) \% If the radius of the most recent ring minus the radius of the first ring is less than \( D_e \) plus \( \text{gap}_e \), execute the statement below

\[ R_{\text{ring}} = R_{\text{ring}1} + D_e + \text{gap}_e; \] \% If the above if statement is true, set the radius of the most recent ring equal to the radius of the first ring plus the diameter of an element, plus the gap

\textbf{end}
end
if sqrt(2*R_ringold^2-(2*R_ringold^2*cos(2*pi/N)))>(3*D_e) %If the distance between two elements side by side in the same ring is greater than the diameter of an element multiplied by 3
    m=1; %set m equal to 1
else
    m=0; % if it is less than or equal to the diameter of an element multiplied by 3, set m equal to 0
end
if ~n
    if m
        h=0.0001;
        while
            R_ringold2+((R_e+gap_e)/tan(asin((R_e+gap_e)/(D_e+gap_e))))+((R_e+gap_e)*tan(h))<(R2-gap_e-R_e)
                h=h+0.0001;
                R_ring=R_ringold2+((R_e+gap_e)/tan(asin((R_e+gap_e)/(D_e+gap_e))))+((R_e+gap_e)*tan(h));
                ang=asin((R_e+gap_e/2)/R_ring);
            ang1=ang;
            if R_ring>=sqrt(((D_e+gap_e)^2)+((R_ringold2)^2)-(2*(D_e+gap_e)*(R_ringold2)))*cos(5*pi/6) && sqrt(((R_ring)^2)+((R_ringold)^2)-(2*(R_ring)*(R_ringold)))>=D_e+gap_e;
                break
            end
        end
    end
else
    ang=0;
end
else
    ang=0;
    $R_{\text{ring}} = R_{\text{ringold}} + D_e + \text{gap}_e$;
end

$\text{ringradius(ring)} = R_{\text{ring}}$;

if $(R_{\text{ring}} + R_e + (\text{gapring}/2)) > R_3$ %If the radius of the most recent ring plus
the radius of an element plus half of a gap is greater than the pressure tubes
buffer
    $\text{ring}$;
    if m
        o=1;
    else o=0;
    end
    $\text{ringradius} = \text{ringradius}(:,1:(\text{ring}-1))$;
    break %break, stop making rings
end

if mod(ring,2)
    for c=1:N
        if c==1
            $[x,y,z]=\text{cylinder}(R_e,200)$;
            plot($x(1,:)+R_{\text{ring}}*\cos(inc+ang),y(1,:)+R_{\text{ring}}*\sin(inc+ang)$)
            hold on
            $[x,y,z]=\text{cylinder}((R_e+\text{gap}_e/2),200)$;
            plot($x(1,:)+R_{\text{ring}}*\cos(inc+ang),y(1,:)+R_{\text{ring}}*\sin(inc+ang)$,'r')
            hold on
        else
            $[x,y,z]=\text{cylinder}(R_e,200)$;
            plot($x(1,:)+(R_{\text{ring}}*\cos((c)*inc+ang)),y(1,:)+(R_{\text{ring}}*\sin((c)*inc+ang))$)
            hold on
[x,y,z]=cylinder((R_e+gap_e/2),200);
plot(x(1,:)+(R_ring*cos((c)*inc+ang)),y(1,:)+(R_ring*sin((c)*inc+ang)),,'r')
hold on
for c=1:N
if c==1
  [x,y,z]=cylinder(R_e,200);
  plot(x(1,:)+R_ring*cos(inc/2+ang),y(1,:)+R_ring*sin(inc/2+ang))
  hold on
  [x,y,z]=cylinder((R_e+gap_e/2),200);
  plot(x(1,:)+R_ring*cos(inc/2+ang),y(1,:)+R_ring*sin(inc/2+ang),,'r')
  hold on
else
  [x,y,z]=cylinder(R_e,200);
  plot(x(1,:)+(R_ring*cos((c-1)*inc+inc/2+ang)),y(1,:)+(R_ring*sin((c-1)*inc+inc/2+ang)))
  hold on
  [x,y,z]=cylinder((R_e+gap_e/2),200);
  plot(x(1,:)+(R_ring*cos((c-1)*inc+inc/2+ang)),y(1,:)+(R_ring*sin((c-1)*inc+inc/2+ang)),,'r')
  hold on
end
end
[x,y,z]=cylinder(R_ring,200);
plot(x(1,:),y(1,:),'g')
hold on
end
if mod(ring,2)
    for c=1:N
        if c==1
            [x,y,z]=cylinder(R_e,200);
            plot(x(1,:)+R_ring*cos(inc-ang),y(1,:)+R_ring*sin(inc-ang))
            hold on
            [x,y,z]=cylinder((R_e+gap_e/2),200);
            plot(x(1,:)+R_ring*cos(inc-ang),y(1,:)+R_ring*sin(inc-ang),'r')
            hold on
        else
            [x,y,z]=cylinder(R_e,200);
            plot(x(1,:)+(R_ring*cos((c)*inc-ang)),y(1,:)+(R_ring*sin((c)*inc-ang)))
            hold on
            [x,y,z]=cylinder((R_e+gap_e/2),200);
            plot(x(1,:)+(R_ring*cos((c)*inc-ang)),y(1,:)+(R_ring*sin((c)*inc-ang)),'r')
            hold on
        end
    end
end
[x,y,z]=cylinder(R_ring,200);
plot(x(1,:),y(1,:),'g')
hold on
else
    for c=1:N
        if c==1
            [x,y,z]=cylinder(R_e,200);
            plot(x(1,:)+R_ring*cos(inc/2-ang),y(1,:)+R_ring*sin(inc/2-ang))
hold on
[x,y,z]=cylinder((R_e+gap_e/2),200);
plot(x(1,:)+R_ring*cos(inc/2-ang),y(1,:)+R_ring*sin(inc/2-ang),'r')
hold on
else
[x,y,z]=cylinder(R_e,200);
plot(x(1,:)+(R_ring*cos((c-1)*inc+inc/2-ang)),y(1,:)+(R_ring*sin((c-1)*inc+inc/2-ang)),'r')
hold on
[x,y,z]=cylinder((R_e+gap_e/2),200);
plot(x(1,:)+(R_ring*cos((c-1)*inc+inc/2-ang)),y(1,:)+(R_ring*sin((c-1)*inc+inc/2-ang)),'r')
hold on
end
end
[x,y,z]=cylinder(R_ring,200);
plot(x(1,:),y(1,:),'g') %Plots the circumference of the last ring calculated
hold on
end
R_ringold3=abs(R_ringold2); %Reassigns R_ringold2 to be R_ringold3
R_ringold2=abs(R_ringold); %Reassigns R_ringold to be R_ringold2
R_ringold=abs(R_ring); %Reassigns R_ring to be R_ringold
if m
  n=1;
  m=0;
end
end
ringradii=ringradius';
hold off
axis equal

CenterLineTemperature2
Fuel Centerline Temperature Code

%%% Center Line Temperature Calculations

% Quantities that need to be entered by the user

clear
clc

P=input ('please input your pressure in kPa: ');
T=input ('please input your inlet temperature in K: ');
L=input ('please input the maximum heated length in m: ');
inclength=input ('please input the incremental length in m: ');
fluid=input ('please input the fluid name (water, D2O, etc): ','s');
type=input ('please select your required heat flux (uniform, cosine, upstream, downstream): ','s');

D_e= 9.5;
ri=zeros;
ro=zeros;
H_bulk=zeros;
L_bulk=zeros;
T_cl=zeros;
R=zeros;
Q=zeros;
L_clad=zeros;
T_shi=zeros;

n=(L/inclength)+1; % Calculates the number of points along the channel

m_flowrate=4.37; % Mass flow rate in kg/s
do=D_e/1000; % Outside diameter of the elements in m
ro(1)=do/2; % Outside radius of the elements in m
dt=do*(((31250*(1-(0.8264^2)))/(2*142024000))^(1/3)); %Wall thickness of the elements in m
di=do-(2*dt); %Inside diameter of the elements in m
ri(1)=di/2; %Inside radius of the elements
L_heated=5.772; %Heated length in m
%if m&&~o
  % elnumb=(ring+1)*N;
%else
  % elnumb= ring*N;
%end
elnumb=54;
R_e=D_e/2;
D1=20; %Diameter of the center unheated element
D2=103.45; %Diameter of the pressure tube
R1=D1/2; %Radius of the center element
R2=D2/2; %Radius of the pressure tube

P_heated=pi*do*elnumb; %Heated perimeter in m
Pwet=(pi*do*elnumb)+((D2/1000)*pi)+((D1/1000)*pi); %Wetted perimeter in m
A_flow=(pi*((R2/1000)^2))-(elnumb*pi*((R_e/1000)^2))-(pi*((R1/1000)^2));
%Flow area in m^2
HeatTransferCoefficient2
T_sho=Temp;
HTC=HTCm;
i=1;
% %
for j=0:inclength:L %Iterations for every point along the channel
  L_clad(i)=0.0176*T_sho(i)+8.116; %Thermal conductivity of the cladding (Inconel 600) in kW/m-K
%L_clad(i)=(1.357*(10^-5)*((T_sho(i))^2)+((19.46/T_sho(i))^1-1)+(-7.135*(10^-5))*((T_sho(i))^2)+(0.06959*(T_sho(i)))); %Thermal conductivity of cladding (Stainless Steels) in kW/mK

%L_clad(i)=(11.45+(1.156*(10^-2)*T_sho(i))+(7.72*(10^-6)*(T_sho(i))^2)); %Thermal conductivity of the cladding(Inconel 718) in kW/mK

%L_clad(i)=(12.767-(5.4348*(10^-4)*(T_sho(i)))+(8.9818*(10^-6)*(T_sho(i))^2)); %Thermal conductivity of the cladding(Zircalloy2 and 4) in kW/mK

Power=8500; %Power per bundle string in kW

R(i)=(log(ro(1)/ri(1)))/(2*pi*L_clad(i)*L_heated); %cladding

Fuel_volume=((pi/4)*(di^2)*L_heated*elnumb); %Fuel Volume in m^3

%e_gen=Power/Fuel_volume; %Heat Generation in kW/m^3

T_shi(i)=T_sho(i)+(qx(i)*do*log(ro(1)/ri(1))/(2*L_clad(i))); %Temperature of the cladding in K

%L_ther = (((-9.59*(10^-9) * (T_shi(i))^3 + 4.29*(10^-5) * (T_shi(i))^-2 - 6.87*(10^-2) * T_shi(i)) + 46.8)/1000; % UO2 + SiC ( 12% SiC + 85% UO2)

%L_ther = (((-1.16*(10^-8) * T_shi(i))^3) + (5.03*(10^-5) * T_shi(i)^2) -(7.76*(10^-2) * T_shi(i)) + 49.1)/1000; % UO2 + SiC (8% SiC + 89%UO2)(use this)

%L_ther = (((1-0.05)/(1+0.05))*(1/(0.0327 + 1.603*10^-4 * T_shi(i))))/1000; % ThO2

%L_ther = (((1-0.05)/(1+2*0.05))*1.158*((1/(0.035+0.286*(10^-3)*T_shi(i)))+(6400/(((10^-3))*T_shi(i))^5/2)))*(exp((-16.35)/(10^-3)*T_shi(i))))/1000; % MOX

%L_ther=(100/(7.5408+(17.692*(10^-3)*T_shi(i)))+3.6142*((10^-3)*T_shi(i))^2)+6400/(((10^-3)*T_shi(i))^5/2))exp(-16.35/((10^-3)*T_shi(i))))/1000; %kW/m-K UO2

%L_ther = (1.37*(T_shi(i)^0.41)*((1-0.05)/(1+0.05))) UN(use this)

%L_ther = (1.864*(exp(-2.14*0.05))*(T_shi(i)^0.361))/1000; % UN
\%L_{\text{ther}} = \frac{((1-0.05)/(1+0.05)) \times 100 \times (1.95 \times 10^{-8} \times (T_{\text{shi}(i)}-273.15-850)^2)}{1000}; \text{ \% UC}

\%L_{\text{ther}} = \frac{((1-0.05)/(1+0.05)) \times 100 \times (0.115+2.7 \times 10^{-5} \times (T_{\text{shi}(i)}-273.15)+2.8 \times 10^{-10} \times (T_{\text{shi}(i)}-273.15)^2+3.035 \times 10^{-12} \times (T_{\text{shi}(i)}-273.15)^3)}{1000}; \text{ \% UO2-BeO}

L_{\text{ther}} = \frac{(1-0.05)/(1+0.05) \times (1+0.01)/(1-0.01) \times (6 \times 10^{-11} \times (T_{\text{shi}(i)}-273.15)^4+(-1*2 \times 10^{-7}) \times ((T_{\text{shi}(i)}-273.15)^3)+0.0003 \times ((T_{\text{shi}(i)}-273.15)^2)+(-1*0.2204) \times (T_{\text{shi}(i)}-273.15)+71.977)}{1000}; \text{ \% UO2-BeO}

Ttest=zeros;

for k=1:1:50
    if k==1
        Ttest(k)=(e_gen(i)*((ro(k)^2)-(ri(k)^2))/(4*mean(L_ther)))+T_shi(i); %Fuel Temperature in K
    else
        ro(k)=ri(k-1);
        ri(k)=ri(k-1)-(ri(1)/50);
        Ttest(k)=(e_gen(i)*((ro(k)^2)-(ri(k)^2))/(4*mean(L_ther)))+Ttest(k-1);
    end
end
T_{\text{cl}}(i)=Ttest(50);
i=i+1;
end
plot(xx,T_{\text{cl}}-273,'r'); hold on; plot(xx,T_{\text{shi}}-273); plot(xx,T_{\text{bulk}}-273);
Heat Transfer Coefficient Code

%%%% Heat Transfer Calculations
%

%%%% Objective: To calculate the HTC using the Mokry Correlation

%%% Quantities that need to be entered by the user

D_hy=4*A_flow/Pwet; %hydraulic diameter in m
G=m_flowrate/A_flow; %kg/m^2-s
n=(L/inclength)+1; %Calculates the number of points along the channel
i=1;

HTCm=zeros; %Pre-assign the matrix for all HTC values
T_bulk=zeros;
Temp=zeros;
t_bulk=zeros;
H_bulk=zeros;
T_wall(i)=zeros;
V_bulk=zeros;
L_bulk=zeros;
H_wall=zeros;
D_bulk=zeros;
D_wall=zeros;
HTC=zeros;
er_T_wall=zeros;
qx=zeros;
Cp_avg=zeros;
Pr_avg=zeros;
q_avg=(8500)/(pi*do*L_heated*elnumb); %uniform heat flux in kW/m^2
if strcmp(type,'uniform') == 1
  %Uniform
  b0=1;  b1=0;  b2=0;  b3=0;  b4=0;  b5=0;  b6=0;  b7=0; %q_avg is the uniform heat flux
elseif strcmp(type,'cosine') == 1
  %Cosine
  b0=0.0826674395;  b1=0.870995913;  b2=0.1768749998;
  b3=-0.3226217824;  b4=0.12207148461;  b5=-0.0207148461;
  b6=1.4036013137e-3;  b7=-1.6731615192e-5;
elseif strcmp(type,'downstream') == 1
  %Down-stream skewed
  b0=0.0921920266;  b1=0.7308422355;  b2=0.3752256758;
  b3=-0.5331702337;  b4=0.206326024;  b5=-0.0332278541;
  b6=1.8826011775e-3;  b7=0;
else
  %Up-stream skewed
  b0=0.0131929303;  b1=1.5915371326;  b2=-0.2119130363;
  b3=-0.3695601674;  b4=0.1795787014;  b5=-0.0313491961;
  b6=1.9800082669e-3;  b7=-1.2660441982e-5;
end
xx=0:inclength:5.772;
nn=length(xx);
for j=0:inclength:L  %Iterations for every point along the channel
  if j==0  %Runs this loop only at the initial condition (i.e. point 0)
t\_bulk(i)=T; \text{ %K}
H\_bulk(i)=\text{refpropm('H','T',t\_bulk(i),'P',P,fluid)}; \text{ %J/kg}
T\_wall(i)=(t\_bulk(i)+5); \text{ %Assumption in K}

\text{else %Runs this loop for points 1 to n}
\quad t\_bulk(i)=\text{refpropm('T','P',P,'H',H\_bulk(i),fluid)}; \text{ %K}
\quad T\_wall(i)=(t\_bulk(i)+5); \text{ %Assumption in K}
\text{end}

a=0;
\text{while(a==0) %Iterates until you are within a tenth of a degree of the actual wall temperature}

qx(i)=q\_avg*(b0+b1*((xx(i))^1)+b2*((xx(i))^2)+b3*((xx(i))^3)+b4*((xx(i))^4)+b5*((xx(i))^5)+b6*((xx(i))^6)+b7*((xx(i))^7)); \text{ %kW/m}^2
\text{e\_gen(i)= 4*qx(i)/(di);}

V\_bulk(i)=\text{refpropm('V','T',t\_bulk(i),'P',P,fluid)}; \text{ %Dynamic Viscosity from NIST in Pa-s}
L\_bulk(i)=\text{refpropm('L','T',t\_bulk(i),'P',P,fluid)}; \text{ %Thermal Conductivity from NIST in W/m\cdot k}
H\_bulk(i)=\text{refpropm('H','T',t\_bulk(i),'P',P,fluid)}; \text{ %Bulk Enthalpy from NIST in J/kg}
H\_wall(i)=\text{refpropm('H','T',T\_wall(i),'P',P,fluid)}; \text{ %Wall Enthalpy from NIST in J/kg}
D\_bulk(i)=\text{refpropm('D','T',t\_bulk(i),'P',P,fluid)}; \text{ %Bulk Density from NIST in kg/m}^3
D\_wall(i)=\text{refpropm('D','T',T\_wall(i),'P',P,fluid)}; \text{ %Wall Density from NIST in kg/m}^3
Cp\_avg(i)=((H\_wall(i)-H\_bulk(i))/(T\_wall(i)-t\_bulk(i))); \text{ %Specific Heat in J/kg-K}
\[ \text{Pr}_{\text{avg}}(i) = \frac{((\text{Cp}_{\text{avg}}(i)) \cdot V_{\text{bulk}}(i))}{L_{\text{bulk}}(i)}; \] % Average Prandtl number, no units

\[ \text{HTC}(i) = \frac{((L_{\text{bulk}}(i)/1000)/D_{\text{hy}}) \cdot 0.0061 \cdot ((G \cdot D_{\text{hy}})/V_{\text{bulk}}(i))^0.904 \cdot ((\text{Pr}_{\text{avg}}(i))^0.684) \cdot ((D_{\text{wall}}(i)/D_{\text{bulk}}(i))^0.564))^2; \] % Heat Transfer Coefficient (Mokry Corr.) kW/m^2K

\[ \text{Temp}(i) = \frac{\text{qx}(i)}{\text{HTC}(i)} + t_{\text{bulk}}(i); \]
\[ \text{ea2} = \text{abs}(\text{Temp}(i) - T_{\text{wall}}(i)); \]
\[ \text{if} \quad \text{ea2} > 0.1 \] % If the temperature is more than one tenth of a degree away, iterate again

\[ T_{\text{wall}}(i) = (\text{Temp}(i) + T_{\text{wall}}(i))/2; \]
\[ \text{else} \] % If it is within one tenth of a degree, end conditions are met

\[ a = 1; \]
\[ \text{er}_T_{\text{wall}}(i) = \text{ea2}; \]
\[ \text{HTC}_m(i) = \text{HTC}(i); \]
\[ \text{end} \]

\[ H_{\text{bulk}}(i+1) = \frac{(\text{qx}(i) \cdot \text{inclength} \cdot P_{\text{heated}} \cdot 1000)}{m_{\text{flowrate}}} + H_{\text{bulk}}(i); \]
% Bulk enthalpy of the next point in J/kg
\[ \text{end} \]

\[ T_{\text{bulk}}(i) = t_{\text{bulk}}(i); \]
\[ i = i + 1; \]
\[ \text{end} \]
Appendix C - List of Publications by Ayman Abdalla


5. Qureshi, A., Draper, Sh., Abdalla, A., King, K., Joel, J., Peiman, W., Pioro,

Appendix D - Awards

Best North American student paper award for paper # 43692 in the 19th International Conference On Nuclear Engineering (ICONE-19), Makuhari, Japan, May 16-19

Appendix E - List of Conferences Attended

The 20th International Conference On Nuclear Engineering (ICONE-20) – ASME 2012 POWER Conference, July 30 - August 3, Anaheim, California, USA.

The 19th International Conference On Nuclear Engineering (ICONE-19), May 16-19, Makuhari, Japan.

The 14th International Topical Meeting on Nuclear Reactor Thermalhydraulics (NURETH-14), September 25-30, Toronto, Canada.