March 2018

A CFD Assisted Closed-loop Control System Design for the 37-element Canadian SCWR under Full Load Condition

Huirui Han  
*The University of Western Ontario*

Supervisor  
Prof. Zhang, Chao  
*The University of Western Ontario*

Graduate Program in Mechanical and Materials Engineering

A thesis submitted in partial fulfillment of the requirements for the degree in Master of Engineering Science

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Abstract

A method to design the closed-loop control system for the 37-element vertical Canadian supercritical water-cooled reactor (SCWR) fuel bundle has been proposed in this study. The dynamic models used in the controller design are obtained based on the computational fluid dynamics (CFD) simulations of the fluid flow and heat transfer of the supercritical water in the SCWR fuel bundle. The Reynolds Stress Model is used in the CFD simulations. Comparisons of the supercritical water flow behaviors and heat transfer phenomenon in the single-rod channel and multi-rod channel are also carried out. The results show that there are secondary flows in the single rod channel. The maximum cladding surface temperatures in the single rod channel and the multi-rod channel are different. Therefore, the multi-rod channel is used to conduct the numerical simulations of the fluid flow and heat transfer in the Canadian SCWR and the CFD data are used to construct the control system for the SCWR.

To construct the dynamic control model, the transient thermo-hydraulic behaviors of the supercritical water in the SCWR fuel bundles are predicted numerically. Step perturbances are used to generate the dynamic responses between the inputs and outputs. The linear dynamic control models are constructed by the system identification technique based on the CFD simulation data. Then, the linear dynamic models are validated by comparing the results with those from full-scale CFD simulations. Based on the linear dynamic models, the control system with two PID controllers is designed to make the SCWR return to the design condition when perturbances occur. The performance evaluation of the proposed control system is carried out by using it in a closed-loop control system for the Canadian SCWR.
Keywords: SCWR, CFD, cladding surface temperature, linear dynamic models, system identification technique, controllers
Authorship Statement

Chapters 3 and 4 of this thesis will be submitted for publications.

All papers are drafted by Huirui Han and modified under the supervision of Prof. Chao Zhang and in consultation with Prof. Jin Jiang, Mr. Binggang Cui in Prof. Jin Jiang’s research group.
Acknowledgement

I would like to express my sincere gratitude to Prof. Chao Zhang, Prof. Jin Jiang and Mr. Binggang Cui for their support and encouragement through my whole research work.

Then I would like to thank to members of our Computational Fluid Dynamics Research Laboratory, especially Hao Luo, Zeneng Sun, Yunfeng Liu, who provided help in both academics and daily life.

Finally, I would also thank to my parents for their love during my whole master's process.
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**Nomenclature**

**Acronyms**

<table>
<thead>
<tr>
<th>Acronym</th>
<th>Description</th>
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</thead>
<tbody>
<tr>
<td>AHWR</td>
<td>Advance Heavy Water Reactor</td>
</tr>
<tr>
<td>BWR</td>
<td>Boiling Water Reactor</td>
</tr>
<tr>
<td>CANDU</td>
<td>CANada Deuterium Uranium</td>
</tr>
<tr>
<td>CFD</td>
<td>Computational Fluid Dynamics</td>
</tr>
<tr>
<td>HTF-SCWR</td>
<td>High Temperature Fast Supercritical Water-cooled Reactor</td>
</tr>
<tr>
<td>LPV</td>
<td>Linear Parameter-Varying</td>
</tr>
<tr>
<td>LWR</td>
<td>Light Water Reactor</td>
</tr>
<tr>
<td>PHWR</td>
<td>Pressurized Heavy Water Reactor</td>
</tr>
<tr>
<td>RSM</td>
<td>Reynolds Stress Model</td>
</tr>
<tr>
<td>SCLWR</td>
<td>Supercritical Light Water-Cooled Reactor</td>
</tr>
<tr>
<td>SCWR</td>
<td>Supercritical Water-Cooled Reactor</td>
</tr>
<tr>
<td>SCWFR</td>
<td>Supercritical Water-Cooled Fast Reactor</td>
</tr>
<tr>
<td>SISO</td>
<td>Single Input Single Output</td>
</tr>
</tbody>
</table>

**Symbols**

- $c_p$: Specific heat capacity
- $C_{ij}$: Convection term
- $D_{L,ij}$: Molecular diffusion
- $D_{T,ij}$: Turbulent diffusion
- $D$: Outer diameter of the channel
- $d$: Inner diameter of the channel
G \hspace{1cm} \text{Transfer function}

g \hspace{1cm} \text{Gravitational acceleration}

\( G_{ij} \) \hspace{1cm} \text{Buoyancy production}

k \hspace{1cm} \text{Turbulent kinetic energy}

\( K_D \) \hspace{1cm} \text{Derivative parameter}

\( K_I \) \hspace{1cm} \text{Integral parameter}

\( K_P \) \hspace{1cm} \text{Proportional parameter}

p \hspace{1cm} \text{Pressure}

\( P_{ij} \) \hspace{1cm} \text{Stress production}

Pr \hspace{1cm} \text{Prandtl number}

Re \hspace{1cm} \text{Reynolds number}

T \hspace{1cm} \text{Temperature}

u \hspace{1cm} \text{Velocity}

y \hspace{1cm} \text{Wall distance}

\( y^+ \) \hspace{1cm} \text{Dimensionless wall distance}

\( Y_M \) \hspace{1cm} \text{Additional dilatation dissipation}

\textbf{Subscripts}

w \hspace{1cm} \text{Wall}

t \hspace{1cm} \text{Turbulent}

cr \hspace{1cm} \text{Value at the critical point}

pc \hspace{1cm} \text{Pseudo-critical point}

\textbf{Greek Letters}

\( \phi \) \hspace{1cm} \text{Heat source}
<table>
<thead>
<tr>
<th>Symbol</th>
<th>Description</th>
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<tbody>
<tr>
<td>$\phi_{ij}$</td>
<td>Pressure strain</td>
</tr>
<tr>
<td>$\phi_{ij,1}$</td>
<td>Slow pressure strain term</td>
</tr>
<tr>
<td>$\phi_{ij,2}$</td>
<td>Rapid pressure strain term</td>
</tr>
<tr>
<td>$\phi_{ij,w}$</td>
<td>Wall reflection term</td>
</tr>
<tr>
<td>$\varepsilon_{ij}$</td>
<td>Dissipation</td>
</tr>
<tr>
<td>$\lambda$</td>
<td>Thermal conductivity</td>
</tr>
<tr>
<td>$\mu$</td>
<td>Dynamic viscosity</td>
</tr>
<tr>
<td>$\rho$</td>
<td>Density</td>
</tr>
</tbody>
</table>
Glossary

**Critical point** - is the point where the liquid and its vapor can coexist. This point is defined by $T_{cr}$ and $p_{cr}$ for each substance.

**Supercritical fluid** - is the fluid whose temperature and pressure are higher than the critical values.

**Pseudo-critical point** - is the point at which the pressure is above the critical value and the specific heat value is the maximum under this pressure and temperature ($T_{pc} > T_{cr}$).
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Chapter 1  Introduction

1.1 Nuclear Reactors

Nowadays, nearly 20% of electricity is produced by nuclear power plants. As the population on the earth grows, the demand for energy will also become urgent. It is obvious that nuclear power plants will play a more important role in the future because of its high power-source ratio and sustainability. Figure 1.1 shows the evolution of the nuclear technology and also the roadmap for the future (USNERAC, 2002). The first nuclear reactors were developed as the prototype reactors in early 1950s-1960s. Then nuclear reactors were put into commercial use in the next decade, some of which are still operating today, such as BWR and CANDU. In the 1990s, Generation III nuclear reactors were developed with improvements on economics and safety. These techniques are still actively under development and used in nowadays nuclear power plants. Generation IV nuclear reactors were proposed recently for further reactor performance improvement. The goals are mainly focusing on higher economics, safety enhancement, waste disposal and higher proliferation resistance.

The Generation IV forum selected six nuclear reactor systems, Gas-Cooled Fast Reactor System, Lead-Cooled Fast Reactor System, Molten Salt Reactor System, Sodium-Cooled Fast Reactor System, Supercritical-Water-Cooled Reactor (SCWR) System, and Very-High-Temperature Reactor System. They are expected to be deployable in 2030 (USNERAC, 2002). Among these six nuclear reactor types, the SCWR system uses the supercritical water as the coolant and operates under the supercritical pressure. The operating condition of the SCWR is similar to the supercritical-water-cooled fossil power plants. Thus, some technologies from supercritical-water-cooled thermal power
plants can be used for reference. Several countries have started to work in this direction, such as Canada, USA, and Russia. Canadian SCWRs is proposed to be based on CANDU reactors in Canada. The moderator is still heavy water, but the coolant is light water. Such kind of SCWR is investigated in this work.

![Figure 1.1 Overview of the generations of Nuclear Energy Systems (USNESC, 2002)](image)

### 1.2 Supercritical Water

The critical point separates the liquid region from the vapor region of the fluid. Supercritical fluids have been used in industries for a long time. In the late 1800s, scientists used hydrothermal process to get high quality crystals (Pioro, Khartabil & Duffey, 2004). Then, supercritical fluids were applied in many heat transfer processes, such as cooling system for turbine blades in jet engines, refrigerants in air-conditioning or refrigerating systems, coolant and fuel for supersonic transport, transforming geothermal energy into electricity, supercritical fluid extraction, supercritical fluid chromatography and polymer processing in chemical and pharmaceutical industries (Pioro et al., 2004). Recently, the supercritical water has been widely used in thermal power plants. Since the water has no liquid-vapor transition feature at the supercritical pressure, the thermal efficiency of power plants can be higher. This thermal process is
going to be used in SCWRs.

Figure 1.2 shows the pressure-temperature diagram of the water. The pseudo-critical line shown in Fig.1.2 consists of pseudo-critical points at different pressures and temperatures under which the specific heat value is the maximum. Figures 1.3-1.6 show the thermal properties variations of the supercritical water for specific heat, thermal conductivity, dynamic viscosity, and density respectively (Pioro & Mokry, 2011). It can be seen that all these properties change dramatically in the region of ± 25°C from the critical point, which is called the pseudo-critical region. The specific heat of the water reaches its peak value in this region and this may cause the heat transfer deterioration. Therefore, it is necessary to investigate whether the deteriorated heat transfer phenomenon will cause the temperature in the fuel bundles to exceed the limit.

![Figure 1.2 Pressure-temperature diagram of water](image-url)
Figure 1.3 Specific heat-temperature diagram of water

Figure 1.4 Thermal conductivity-temperature diagram of water
Figure 1.5 Dynamic viscosity-temperature diagram of water

Figure 1.6 Density-temperature diagram of water
1.3 SCWR

Figure 1.7 shows the configuration of a typical SCWR system. The water needs to be heated to steam, which then drives the turbine to produce the mechanical energy. Finally, the generator converts the mechanical energy to electricity. When the subcritical water is used as the coolant, steam generators and steam separators are not needed since there is no phase change for the supercritical water, which can result in a simpler configuration and higher thermal efficiency for the nuclear reactor power plant. Actually, the efficiency of the supercritical water-cooled reactors can reach to about 44%, compared to 33%-35% for LWRs (USNERAC, 2002).

![SCWR Diagram](image-url)
1.4 Methodology

If some disturbances occur during the operation of the supercritical water-cooled reactor system, the fuel rod cladding surface temperature may increase sharply because the thermo-physical properties of the supercritical water change sharply around the pseudo-critical region. This results in challenges for the design of the control system used to bring the system back to the desired operating point when it is subjected to disturbances. CFD method can be useful in predicting the heat transfer phenomenon and flow stability behaviors of the supercritical water in SCWR fuel channels. Also, since the SCWR is still at the design stage, there are no experimental data available, which are needed for the design of the control system. CFD simulations are used to generate the required input and output data for the SCWR. Based on the input and output data, the dynamic control models are constructed. Then, a closed-loop control system is designed based on the dynamic control models to bring the system back to its operating point quickly after being disturbed.

1.5 Organization of the Thesis

The thesis is organized in the following order:

Literature reviews of CFD simulations of the fluid flow and heat transfer of supercritical fluids, and the control system design methods for the nuclear power plants are shown in Chapter 2. In Chapter 3, the numerical simulation results of the heat transfer phenomenon and flow structure of supercritical water in different fuel channels are compared. Then, Chapter 4 presents a closed-loop control system for the Canadian
SCWR designed based on the transient numerical simulation of the fluid flow and heat transfer in the Canadian SCWR as well as the performance evaluations of the proposed control system. Finally, the conclusions and future works are shown in Chapter 5.
References


USNERAC. (2002). *A Technology Roadmap for Generation IV Nuclear Energy Systems*. Retrieved from The U.S.DOE nuclear energy research advisory committee and the generation IV international forum:
Chapter 2  Literature Review

A literature review of the flow dynamics and heat transfer for supercritical fluids is shown at first. Then studies of SCWRs and fuel bundles are presented. In this chapter, some previous experimental and numerical studies on supercritical fluid flow and heat transfer are introduced. Besides, recent studies on the control system designs for SCWR are also discussed. Then, the motivations and objectives of this study are presented.

2.1 Heat Transfer and Flow Characteristics Research on Supercritical Fluids

2.1.1 Supercritical Fluid Properties

When a fluid changes from a liquid-like state to a vapor-like state at the supercritical pressure, the fluid properties change significantly. This process happens at about ± 25°C from the pseudo-critical temperature. Since supercritical fluids, such as water, are widely used nowadays, it is very important to have an in-depth knowledge about their thermo-hydraulic behavior at the supercritical condition to make sure safe and efficient industrial use (Pioro & Duffey, 2007).

2.1.2 Experimental Studies of Supercritical Fluids

The experimental studies of supercritical fluids have already been conducted since 1960s. Focus of these experiments can be summarized in two aspects:

(1) The flow and heat transfer characteristics of supercritical fluids

(2) The correlations to describe supercritical fluids heat transfer phenomenon
CO\textsubscript{2} and water were used in most experimental investigations on supercritical fluid flow and heat transfer. Swenson et al. (1965) systematically studied heat transfer characteristics of the supercritical water in smooth bare tubes under different flow conditions. The results showed that the heat transfer coefficient is strongly affected by the heat flux. The peak value occurred around the pseudo-critical temperature. And higher maximum value was seen at lower heat fluxes. This was also observed in the work by Yamagata et al. (1972). Ackerman (1970) also did similar experiments. It was found that the pseudo-boiling phenomenon could occur at the supercritical pressure.

Yamagata et al. (1972) conducted experiments of the supercritical water in both horizontal and vertical tubes under low and high heat fluxes. The heat transfer coefficient when the heat flux approaches zero was considered as the reference value. The results showed that the heat transfer deterioration started when the specific heat ratio=0.3. Glushchendo et al. (1972) did experiments of upward and downward flows in tubes under different flow conditions. The occurrence of the heat transfer deterioration was observed in both upward and downward flows under the same boundary conditions. However, Krasyakova et al. (1977) discovered that the wall temperature variation along the tube was much smoother in the downward flow tube than that in the upward flow tube.

Several experiments were performed to show the velocity and temperature profiles of supercritical fluids in tubes. Kurganov & Kaptil’ny. (1992) did the experiments of the supercritical CO\textsubscript{2} flow through a heated vertical circular tube. The flow structure in both the normal and deteriorated heat transfer conditions were compared. Pioro & Khartabil
(2005) also investigated similar CO₂ upward flows. Different wall and fluid bulk temperatures around the pseudo-critical temperature were used. And the wall temperatures and bulk fluid temperatures under different operating conditions were presented.

The experiments of supercritical fluids in different channel geometries were also conducted by several researchers. Kim et al. (2005) did experiments of supercritical CO₂ flows in various vertical tubes. The cross-sections of the tube were circle, triangular, and square. The heat transfer characteristics in the tubes with different geometries were compared. Wardana et al. (1999) studied on air flow in annulus with strongly heated inner cylinder. With the increase in the wall temperature, the turbulent fluctuations close to and within the viscous sub-layer were suppressed due to the increase in the kinematic viscosity. Similar experiments were also conducted by Kang et al. (2001) using liquid R-113 and Licht et al. (2009) using water but in square annular tubes.

Although there is no consensus on exact heat transfer correlations for describing deteriorated or improved heat transfer of supercritical fluids in tubes, many researchers have found different correlations under different operating conditions. Pioro et al. (2004) and Yoo (2013) presented a detailed summary of previous research works on heat transfer correlations. The majority of those correlations can be expressed in Dittus-Boelter form (Maitri, 2014):

\[ Nu = CRe^{\alpha}Pr^{\beta}F \] (2.1)
Here, \( F = \frac{P_f^n}{P_{\infty}^{n_1}} \left( \frac{\mu_f}{\mu_{\infty}} \right)^{n_2} \left( \frac{\lambda_f}{\lambda_{\infty}} \right)^{n_3} \left( \frac{c_p}{c_p_{\infty}} \right)^{n_4} \), which is based on the ratio of different thermal properties of bulk fluid and fluid at the wall. The summary of the heat transfer correlations for supercritical water flow in circular tubes based on the experimental data by different researchers are shown in Table 2.1.

### 2.1.3 Numerical Studies of Supercritical Fluid Flows

Experiments using supercritical fluids can be very expensive. Thus, computational fluid dynamics method is introduced to investigate the heat transfer and flow characteristics of supercritical fluids. Several turbulence models have been used in the numerical simulations, such as the \( k-\varepsilon \) model, \( k-\omega \) model, RSM model, etc.

The \( k-\varepsilon \) model was widely used for simulating supercritical fluids at first. Jones & Launder (1972) presented the \( k-\varepsilon \) model, in which the local turbulent viscosity is determined from the solution of the transport equations for the turbulence kinetic energy and the its dissipation rate. The model was validated by comparing with the experimental data. Renz & Bellinghausen (1986) included the gravity influence in this model and simulated the heat transfer in a vertical pipe at a supercritical pressure. The results indicated that the turbulent structure variation near the wall and the turbulent damping effect leaded to the heat transfer deterioration. Then, Bellinghausen & Renz (1990) validated Renz's simulation results with the experimental data. The results showed a good agreement. Wilcox (1988) developed the \( k-\omega \) model. The model gave more accurate results for flows with an adverse pressure gradient than any other similar models.
Menter (1993) combined the $k-\varepsilon$ and $k-\omega$ models, called $k-\omega_{SST}$ model, in which the $k-\omega$ model was used in the boundary layer region, and the $k-\varepsilon$ model was used for the free stream region. This model performs well in flows with adverse pressure gradients and separation. This model was further developed and used later by other researchers. Kim et al. (2004) simulated the vertical upward flow of the water in a heated tube under supercritical pressure. And the experimental data by Yamagata et al. (1972) were used for evaluating the accuracy of several selected turbulence models: $k-\omega$ model, SST $k-\omega$ model, standard $k-\varepsilon$ model, RNG $k-\varepsilon$ model, and realizable $k-\varepsilon$ model. Although RNG $k-\varepsilon$ model with enhanced wall treatment gives better predictions generally, a turbulence transport equation is still required to overcome the effect of the sharp changes of the fluid properties on the prediction of the wall temperature.

Sharabi et al. (2008) also did simulations of the heat transfer of CO$_2$ at supercritical pressures in the upward heated square and triangular tubes. The low-Reynolds number $k-\varepsilon$ model and RNG $k-\varepsilon$ model with a two-layer near wall treatment were used. The experimental data were from Kim et al. (2005). The numerical results obtained using the low-Reynolds number model were able to reproduce the trend of the heat transfer deterioration due to the buoyancy influence.

The two-equation turbulence models, $k-\varepsilon$ and $k-\omega$, are based on the hypothesis of isotropic turbulent viscosity, which means the ratio between Reynolds stress and mean rate of deformations is the same in all directions. For complicated flows, these models may not be accurate enough. Cheng et al. (2007) and Zhang et al. (2011) did simulations with anisotropic turbulence models. Zhang et al. (2011) used the Reynolds
Stress Model (RSM) in the simulation and compared the accuracy of the numerical results from the RSM and other isotropic models with experimental data. The results showed that the RSM performed better than others. Maitri et al. (2014) also did similar studies, and compared the heat transfer deterioration phenomenon using different turbulence models.

Table 2.1 Summary of the parameters in the heat transfer correlations for the supercritical water

<table>
<thead>
<tr>
<th>Researchers</th>
<th>C</th>
<th>θ</th>
<th>n_1</th>
<th>n_2</th>
<th>n_3</th>
<th>n_4</th>
<th>n_5</th>
<th>n_6</th>
<th>n_7</th>
<th>φ</th>
</tr>
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<tr>
<td>Dittus-Boelter (2007)</td>
<td>0.023</td>
<td>0.80</td>
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<td>0.0</td>
<td>0.0</td>
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<td>0.0</td>
<td>0.0</td>
<td>1.0</td>
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<td>-0.12</td>
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<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>1.0</td>
</tr>
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<td>Jackson (2002)</td>
<td>0.0183</td>
<td>0.82</td>
<td>0.50</td>
<td>0.0</td>
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<tr>
<td>Swenson et al. (1965)</td>
<td>0.00459</td>
<td>w</td>
<td>0.923</td>
<td>0.0</td>
<td>0.613</td>
<td>0.231</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Watts &amp; Chou (1982)</td>
<td>0.021</td>
<td>0.80</td>
<td>0.0</td>
<td>0.55</td>
<td>0.350</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>0.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Yamagata et al.(1972)</td>
<td>0.0135</td>
<td>0.85</td>
<td>0.8</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

where

\[ n_j = 0.4 \text{ for } T_\infty < T_w < T_{pc} \text{ and } 1.2 T_{pc} < T_\infty < T_w \]

\[ n_j = 0.4 + 0.2 \left( \frac{T_w}{T_{pc}} - 1 \right) \text{ for } T_\infty < T_{pc} < T_w \]

\[ n_j = 0.4 + 0.2 \left( \frac{T_w}{T_{pc}} - 1 \right) \left[ 1 - 5 \left( \frac{T_\infty}{T_{pc}} - 1 \right) \right] \text{ for } T_{pc} < T_\infty < 1.2 T_{pc} \text{ and } T_\infty < T_w \]

\[ \phi_w = 1 \text{ for } B_u^* > 10^{-5} \]

\[ \phi_w = \left[ 1 - 3000 B_u^* \right]^{0.295} \text{ for } 10^{-5} < B_u^* < 10^{-4} \]

\[ \phi_w = \left[ 7000 B_u^* \right]^{0.295} \text{ for } B_u^* > 10^{-4} \]

\[ B_u^* = \frac{G_{R_w}}{R_{e_w}^{0.50}} \text{ and } G_{R_w} = g \left( 1 - \frac{\rho_w}{\rho_\infty} \right) \frac{D^3}{2} \]

\[ F = \frac{C_p}{C_p} \text{ for } E > 1 \]

\[ F = 0.67 + P_{R_{pc}}^{-0.05 \left( C_p / C_p \right)^n} \text{ for } 0 \leq E \leq 1 \]

\[ F = \left( C_p / C_p \right)^n \text{ for } E < 0 \]

\[ n_1 = -0.77 (1 + 1 / P_{R_{pc}}) + 1.49 \]

\[ n_2 = 1.44 (1 + 1 / P_{R_{pc}}) - 0.53 \]
2.2 Experiments for flows in fuel bundles

Kjellstrom (1971) did the flow profiles measurements of air flows in a triangular rod bundle. Additional experiments using normal fluids in rod bundles were conducted later. Trupp & Azad (1975) changed the pitch-to-diameter ratio of hexagonal lattices between 1.2 and 1.5 and measured detailed turbulence profiles of the air flow. The eddy viscosity showed strong anisotropy. Carajilesco & Todreas (1976) used laser doppler anemometry to measure the water flow characteristics in triangular subchannels. Vonka (1988) also did similar research, but the fluid material was chosen for the same refractive index with wall. They both found that the secondary flow velocity was less than 1% of the mean flow velocity.

Turbulent mixing rate of fluids in channels have been studied by many researchers for a long time. Table 2.2 gives a summary of the turbulent mixing coefficient from different researchers. Jeong et al. (2007) defined a new mixing factor to evaluate all experimental data from different researchers about turbulent mixing. It was found that the ratio of the distance between the center of two adjacent subchannels and hydraulic diameter of a sub-channel can better describe the turbulent mixing. Xi et al. (2014) did an investigation about the supercritical water flow between the two heated parallel channels. The oscillations of the inlet mass flow rate and the outlet temperature were both observed. Verma et al. (2017) did experiments on a scaled test facility of an advanced heavy water reactor rod bundle to establish the effect of the spacer on the mixing rate in subchannels. The results indicated that the turbulent mixing rate increased with the increase in the average Reynolds number.
2.3 Numerical studies of the SCWR

It is obvious that anisotropic turbulence models are more accurate in the simulation of the supercritical water in fuel bundles. Gu et al. (2010) conducted the simulation of supercritical water in a typical SCWR fuel bundle by CFX. The results showed that the turbulent mixing was sensitive to the asymmetric boundary condition. Mukohara et al. (2000) did sub-channel analysis in high temperature fast supercritical water cooled reactor. They found that the cladding temperature was sensitive to the local power peak value and the sub-channel cross-section area. Yu et al. (2007) developed a sub-channel thermal-hydraulic analysis code, SUBCHAN, to analyze the thermo-hydraulic behavior in the CANDU-SCWR. The results showed that this code can simulate steady state flows successfully.

2.4 Development of the control system for the SCWR

Most studies of SCWRs until now are about numerical simulations of the fluid flow behaviors or heat transfer characteristics in tubes or fuel channels. Since the complexity of nuclear reactors and supercritical water properties, only a few control strategies for SCWR were developed.

Nakatsuka et al. (1998) did the first study on the control system for a supercritical water reactor. They experimentally analyzed the dynamic behaviors of the supercritical fast cooled reactor by adding perturbances on the control rod position, the feedwater flow rate, and the turbine control valve opening, respectively. Then the input and output pairings for the reactor were obtained: the main steam pressure was controlled by the turbine control valves, the main steam temperature was controlled by the feedwater flow
rate, and the core power was controlled by the control rods. Parameters of the control system are adjusted to satisfy both fast convergence and stability criteria. Then those parameters were optimized for reducing the overshoot with fast response. The results showed that the control system can maintain the supercritical water fast cooled reactor at a stable condition when perturbances were introduced on the input parameters. Later, Ishitawari et al. (2003) used Nakatsuka's method in the control system design for the supercritical high temperature light water reactors.

Because of strong cross-coupling features of the supercritical water-cooled reactor, it is difficult to design the control system. Sun (2012) simulated thermo-hydraulic behaviors of Canadian SCWR by a simple 1D model. Since the system identification technique method can only be used on single-input-single-output (SISO) systems, the direct Nquist array method was used to decouple the system and pre-compensators were used to convert the system to the diagonally dominant form. Although the control system design was able to keep the system at a stable condition when there were perturbances, the simulation method used for the reactor was too simple. Then, Sun et al. (2015) used a feed-forward control method in the control system to suppress the steam temperature variation when the reactor power is subjected to disturbances. Later, Sun et al. (2017) found that the response magnitude of the steam temperature to the same amount of the feedwater flow rate disturbance at a high power level was smaller than that at a low power level. This will lead to the original control system design not effective when working conditions were changed. A linear parameter-varying strategy was proposed to solve such problems. The results showed that the linear parameter-varying controller can stabilize the steam temperature under different operation conditions. Maitri et al. (2017) adopted the results from CFD simulations to derive the linear dynamic models
around the design operating condition using the system identification technique. Then, the linear dynamic models were validated by the full-scale CFD simulation data and the agreement was good. Although the simulation of the supercritical water flow in fuel bundles was simplified as a 2D tube flow, this confirmed that CFD simulations can be used for the development of the linear dynamic models for Canadian SCWR. Recently, Sun & Zhang (2017) investigated the relationship between the reactor and the turbine of SCWRs. A dynamic model of the Canadian SCWR turbine was developed and coupled with the reactor model to obtain the nonlinear dynamic model for the entire plant. Based on the nonlinear dynamic model, three control strategies were studied and the control performances were evaluated by introducing a typical load pattern. The test results showed that the proposed control strategy was effective in maintaining the Canadian SCWR under required operating conditions.

Table 2.2 Summary of the turbulent mixing coefficient

<table>
<thead>
<tr>
<th>Experimenter</th>
<th>Fluid</th>
<th>S/d</th>
<th>Turbulent Mixing Coefficient β</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gabraith &amp; Knudsen (1971)</td>
<td>Water</td>
<td>0.011</td>
<td>$\beta = 2.8365 \times 10^{-13} \cdot Re^{2.43}$</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.028</td>
<td>$\beta = 0.001571 \times Re^{0.23}$</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.063</td>
<td>$\beta = 0.002871 \times Re^{0.12}$</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.127</td>
<td>$\beta = 0.002277 \times Re^{0.12}$</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.228</td>
<td>$\beta = 0.005999 \times Re^{0.01}$</td>
</tr>
<tr>
<td>Rogers &amp; Rosehart (1972)</td>
<td>--</td>
<td>--</td>
<td>$\beta = 0.004 \cdot \left( \frac{D_h}{S} \right) Re^{-0.1}$</td>
</tr>
<tr>
<td>Castellana et al. (1974)</td>
<td>Water</td>
<td>0.334</td>
<td>$\beta = 0.027 \cdot Re^{-0.1}$</td>
</tr>
<tr>
<td>Rogers &amp; Tahir (1975)</td>
<td>Air</td>
<td>0.4</td>
<td>$\beta = 0.005 \cdot \left( \frac{D_h}{S} \right) \left( \frac{S}{d} \right)^{0.106} Re^{-0.1}$</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>$\beta = 0.007479 \cdot Re^{-0.1}$</td>
</tr>
<tr>
<td>Rowe &amp; Angle (1976)</td>
<td>Water</td>
<td>0.036</td>
<td>$\beta = 0.063 \cdot Re^{-0.1}$</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.149</td>
<td>$\beta = 0.021 \cdot Re^{-0.1}$</td>
</tr>
<tr>
<td>Kelly &amp; Todreas (1979)</td>
<td>Water</td>
<td>0.1</td>
<td>$\beta = 0.0070 \cdot Re^{-0.065}$</td>
</tr>
<tr>
<td>Seale (1979)</td>
<td>Air</td>
<td>0.1</td>
<td>$\beta = 0.02968 \cdot Re^{-0.1}$</td>
</tr>
</tbody>
</table>
\[ \beta = 0.01683 \cdot \text{Re}^{-0.1} \]
\[ \beta = 0.009225 \cdot \text{Re}^{-0.1} \]

### 2.5 Motivations

Previous experimental and numerical studies on the SCWR are mainly focused on the flow and heat transfer phenomenon of supercritical fluids in circle channels. However, the fuel bundle used in the SCWR has multiple fuel rods. So, the flow behavior is different from the channel flows. The experimental research for supercritical water in the fuel bundle is both costly and time consuming. CFD method is a useful tool to predict the thermo-hydraulic behaviors of the supercritical water in the Canadian SCWR. Therefore, a transient three-dimensional full-scale CFD simulation for the supercritical water in the vertical fuel bundle of the 37-element Canadian SCWR will be carried out.

Several researchers constructed control systems for the supercritical water-cooled reactors based on numerical simulations. But most of the numerical studies only consider supercritical water in circle channels or the fuel bundles with only one fuel rod. It cannot represent the accurate thermo-hydraulic behaviors of the supercritical water in the Canadian SCWR fuel bundles. In this work, a control system is constructed based on the full-scale numerical simulations of the 37-element Canadian SCWR.

### 2.6 Objectives

The main objectives of this thesis can be summarized as follows:
(1) Compare the heat transfer phenomenon of supercritical water in the fuel bundle with a single fuel rod and multiple fuel rods.

(2) Conduct the transient simulations of fluid flow and heat transfer of supercritical water in the 37-element Canadian SCWR fuel bundle. Step disturbances are applied in the numerical simulation to obtain the dynamic relationship between the inputs and outputs, which will be used to construct the linear dynamic control models for the Canadian SCWR. The linear dynamic control models are validated by comparing them with the nonlinear CFD simulations.

(3) Design the control system for the Canadian SCWR to maintain the reactor at the design point when it is subjected to perturbances and validate the performance of the proposed control system.
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Maitri, R. V. (2014). A CFD assisted control system design for supercritical water cooled reactor. (Master of Engineering Science), The University of Western Ontario, Electronic Thesis and Dissertations Repository. (2169)


Maitri, R. V. (2014). A CFD assisted control system design for supercritical water cooled reactor. (Master of Engineering Science), The University of Western Ontario, Electronic Thesis and Dissertations Repository. (2169)


Rowe, D. S., & Angle, C. W. Crossflow mixing between parallel flow channels during boiling, Part II: Measurement of flow and enthalpy in two parallel channels. Retrieved from United States:


Chapter 3  Numerical Simulation of Fluid Flow and Heat Transfer of the Supercritical Water in Different Fuel Rod Channels

Abstract

The supercritical water-cooled reactor was proposed as one of the Generation IV nuclear systems. Although many research works on the fluid flow and heat transfer of supercritical water in circular channels have been conducted, there is still lack of research on the fluid flow and heat transfer process in fuel bundles used in supercritical water-cooled nuclear reactors. Besides, fuel bundles have multiple fuel rods, the flow is an external flow, not internal flow as that in circle channels, which will cause the difference in the fluid flow phenomenon and heat transfer on the fuel rod cladding surface. In this work, the heat transfer and fluid flow characteristics of the supercritical water in the single-rod channel and the multi-rod channel are simulated numerically. The results show that there are secondary flows in both channels. The circumferential cladding surface temperature variation is large and should be considered in the future fuel rod design. With the same flow rate and heat flux input, the maximum cladding surface temperature in the multi-rod channel is much higher than that in the single-rod channel. Since the maximum cladding surface temperature is an important parameter for the safety of the nuclear reactor operation, it is recommended to use the multi-rod channel model to conduct numerical simulations for the fluid flow and heat transfer of the supercritical water in the Canadian SCWR.
Keywords: SCWR, heat transfer, single rod, multi-rod, cladding surface temperature
3.1 Introduction

With the rapid growing population, the electricity generation amount is in high demand. Nuclear reactor power plants can provide higher power-to-sources rate, compared with thermal power plants using coals or natural gas. Canada has a long history in the development of the CANDU-PHWR, which has been operating for about half century. The Supercritical Water Cooled Reactor (SCWR) is proposed as one of the six selected Generation IV reactor systems in the world since 2002 (USNERAC, 2002), which has unique advantages, such as higher thermal efficiency, lower coolant mass flow to the thermal power rate, and simpler components.

The studies for the flow in rod bundles have been conducted by several researchers. Kjellstrom (1971) did the flow profile measurements of air flows in a triangular rod bundle. Trupp & Azad (1975) changed the pitch-to-diameter ratios of hexagonal lattices between 1.2 and 1.5 and measured detailed turbulence profiles of the air flow. The eddy viscosity showed strong anisotropy. Carajileskov & Todreas (1976) and Vonka (1988) used Laser Doppler Anemometry to measure the water flow characteristics in triangular subchannels. They found that the secondary flow velocity was less than 1% of the mean flow velocity.

Many researchers have investigated the turbulent mixing rate of the fluids in channels. Jeong et al. (2007) defined a new mixing factor and then evaluated all the experimental data from previous researchers on the turbulent mixing. It was found that the turbulent mixing of fluids depends strongly on the ratio of the distance between the center of two adjacent sub-channels and the hydraulic diameter of a sub-channel. Only very few experimental studies were carried out for the heat transfer and flow phenomenon of
supercritical fluids because of the experiment environment restrictions. Xi et al. (2014) did an investigation on the supercritical water flow between two heated parallel channels. Both inlet mass flow rate and outlet temperature oscillations were observed. Verma et al. (2017) carried out the experiments using a scaled test facility of AHWR (advanced heavy water reactor) rod bundle. The effect of the spacer on the turbulent mixing rate in subchannels was investigated. The results showed that the turbulent mixing rate increased with the increase in the average Reynolds number.

The simulation results for the flow of the supercritical water in fuel bundles showed that the anisotropic turbulence models are more accurate. Gu et al. (2010) simulated the supercritical water flow in a SCWR fuel bundle using the sub-channel method. The simulation results demonstrated that the turbulent mixing rate was sensitive to the asymmetric boundary condition. Mukohara et al. (2000) conducted the sub-channel analysis in HTF-SCWR (high temperature fast supercritical water cooled reactor). It was found that the cladding surface temperature was sensitive to both the local power peak value and the sub-channel area. Yu et al. (2007) developed a sub-channel analysis code to analyze the thermo-hydraulic behavior of the CANDU-SCWR. The simulation results showed that this code can successfully simulate the steady state flows in sub-channels.

The Canadian SCWR concept is based on CANDU. Because of the sharp variation of the supercritical water properties around the pseudo-critical point, it is important to use appropriate anisotropic turbulence models for the simulations of the supercritical water flow behaviors and heat transfer phenomenon in fuel bundles. Previous researchers Cheng et al. (2007) and Zhang et al. (2011) have proved that the anisotropic model RSM
(Reynolds Stress Model) can give a better agreement with the experimental results for the supercritical water flow in channels compared with the isotropic two-equation turbulence models.

Previous numerical studies mainly focused on the flow and heat transfer phenomenon of the supercritical water in circle channels. However, the fuel bundle used in the SCWR has multiple fuel rods. It is time-consuming to simulate the fluid flow and heat transfer in the channel with multiple rods. Therefore, some researchers conducted simulations using simplified geometries, such as the work by Sun et al. (2012), where the multiple fuel rod system was simplified as a single-rod system. Therefore, in this study, the CFD simulations are carried out for the fluid flow and heat transfer of the supercritical water in both the single-rod channel and multi-rod channel under the same operating conditions in order to compare the difference in the results between them. The CFD simulations are conducted with the RSM using ANSYS FLUENT 15.0.

3.2 Governing Equations and Numerical Models

The governing equations for 3D steady flow and heat transfer are conservations of mass equation, momentum equation and energy equation, which is shown as follows in the Cartesian tensor (ANSYS, 2011):

\[
\frac{\partial}{\partial x_i} (\rho u_i) = 0 \quad (3.1)
\]

\[
\frac{\partial}{\partial x_j} (\rho u_i u_j) = -\frac{\partial p}{\partial x_i} + \frac{\partial}{\partial x_j} (\mu \frac{\partial u_i}{\partial x_j} - \rho u_i u_j) + \rho g_i \quad (3.2)
\]

\[
\frac{\partial}{\partial x_i} (u_i \rho c_p T) = \frac{\partial}{\partial x_i} \left[ (\lambda + \frac{c_p \mu}{P_{rt}}) \frac{\partial T}{\partial x_i} \right] + \phi \quad (3.3)
\]
Here, \( u \) is the velocity, \( T \) is the temperature, \( \mu \) is the dynamic viscosity, \( \rho \) is the density, \( \lambda \) is the thermal conductivity, \( c_p \) is the specific heat, \( \mu_t \) is the turbulent viscosity, and \( Pr \) is the turbulent Prandtl number. The RSM with the enhanced wall treatment is chosen in this study based on the recommendations from the previous studies (Zhang et al., 2011). The enhanced wall treatment is a near-wall modeling method, which combines a two-layer model with enhanced wall functions. If the near-wall mesh is fine enough to resolve the viscous sub-layer (typically with the first near-wall node paced at \( y^+ \approx 1 \)), the enhanced wall treatment will be the same as the traditional two-layer model (ANSYS, 2011). The two-layer model is used to specify both \( \varepsilon \) and the turbulent viscosity in the near-wall cells. The computational domain is subdivided into a viscosity-affected region and a fully-turbulent region. And the boundary of the two regions is determined by a wall-distance-based, turbulent Reynolds number, \( Re_y \). It can be defined as (ANSYS, 2011):

\[
Re_y = \frac{\rho y \sqrt{k}}{\mu}
\]  

(3.4)

Here \( k \) is the turbulence kinetic energy, \( y \) is the wall-normal distance between the cell center and the nearest wall, and it is calculated by the following equation (ANSYS, 2011):

\[
y = \min_{r_w \in \Gamma_w} \left\| r - r_w \right\|
\]  

(3.5)

Here \( r \) is the position vector at a point in the flow field, \( r_w \) is the position vector of the wall boundary, \( \Gamma_w \) is the union of all the wall boundaries involved. Thus, \( y \) can be defined in a flow domain with complex shape involving multiple walls. In the fully
turbulent region \((Re_y > Re_y^* = 200)\), the RSM is employed. In the viscosity-affected near-wall region \((Re_y < Re_y^* = 200)\), the one equation model proposed by Wolfshtein (1969) is used. In this one-equation model, \(\mu_t\) is computed from:

\[
\mu_{t,2layer} = \rho C_{\mu} l_{\mu} \sqrt{k}
\]

(3.6)

Here the length scale \(l_{\mu}\) is determined by the equation from Chen & Patel (1988):

\[
l_{\mu} = C_l (1 - e^{-Re_y/A_y})
\]

(3.7)

Jongen (1998) proposed a two-layer formulation for the turbulent viscosity, which can smoothly blend with the turbulent viscosity from the outer region:

\[
\mu_{t,enh} = \lambda_c \mu_t + (1 - \lambda_c) \mu_{t,2layer}
\]

(3.8)

Here, \(\lambda_c\) is 1 in the region far from wall, and 0 in the region near the wall.

ANSYS Fluent 15.0 is used to solve the governing equations. The SIMPLE scheme is selected for pressure correction, and QUICK method is used for the spatial discretization. The convergence criteria for continuum is \(10^{-3}\), for the momentum and turbulence parameters are \(10^{-5}\), and for the energy equations is \(10^{-6}\).

### 3.3 Configurations of the Channels and Grid Independent Tests

The single-rod channel and the multi-rod channel are shown in Figures 3.1 and 3.2, respectively. The working fluid is the supercritical water, and its properties are from Wagner (1998). The length of the channel is 1.5 m. For the single-rod channel, the channel diameter is 9mm and the rod diameter is 4mm. And for the multi-rod channel, the outer diameter is 9 mm. There are 5 rods in the channel and their diameter is 1.788mm, so, the total cross-section areas of all 5 rods is equal to the cross-section area
of the rod in the single-rod channel. The supercritical water flow in these channels is upward based on the configuration of the proposed SCWR (Leung, 2013). The reference pressure is 25MPa (Leung, 2013).

Figure 3.1 The single-rod channel

Figure 3.2 The multi-rod channel

Boundary conditions are as follows:
Inlet: The inlet velocity for each channel is 3m/s, and the inlet temperature is 623.15K. Turbulence intensity is set as 5%, and the hydraulic diameters are specified based on the geometrical shapes of the channels.

Outlet: Outflow is selected for each channel.

Walls: They are all smooth walls with the no-slip condition. The heat flux on the fuel rod surface is $10^6 \text{ W/m}^2$ based on the operating condition of the SCWR (Leung, 2013).

The cross-section views of the meshes used for these two channels simulations are shown in Figures 3.3 and 3.4, respectively. The mesh refinement near the wall is performed so that the non-dimensional distance to the wall $y^+$ is approximately 1.

![Cross-section view of the mesh for the single-rod channel](image-url)
The grid independent tests are performed for both the single-rod channel and multi-rod channel. Three mesh sizes are used for each channel for the grid independent tests. The mesh qualities for the two channels are shown in Tables 3.1 and 3.2, respectively. The outlet bulk temperature and the outlet mass flow rate are selected for assessing the grid independence. The differences in the results using the three mesh sizes for each channel type are shown in Table 3.3 and Table 3.4, respectively. The differences between the medium mesh and fine mesh are both very small. Thus, the fine meshes are chosen for the simulations.

Table 3.1 Mesh qualities for the single-rod channel

<table>
<thead>
<tr>
<th>Mesh Size</th>
<th>Size</th>
<th>Coarse</th>
<th>Medium</th>
<th>Fine</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Nodes</td>
<td>29234</td>
<td>133293</td>
<td>217942</td>
</tr>
<tr>
<td></td>
<td>Cells</td>
<td>134991</td>
<td>570765</td>
<td>1040826</td>
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<tr>
<td></td>
<td>Faces</td>
<td>281311</td>
<td>1203328</td>
<td>2170681</td>
</tr>
</tbody>
</table>
Table 3.2 Mesh qualities for the multi-rod channel

<table>
<thead>
<tr>
<th>Mesh Size</th>
<th>Size</th>
<th>Coarse</th>
<th>Medium</th>
<th>Fine</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nodes</td>
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<td>210000</td>
<td>508491</td>
<td></td>
</tr>
<tr>
<td>Cells</td>
<td>552720</td>
<td>1989577</td>
<td>2491526</td>
<td></td>
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<tr>
<td>Faces</td>
<td>1701278</td>
<td>3194031</td>
<td>5182536</td>
<td></td>
</tr>
</tbody>
</table>

Table 3.3 Grid independence tests for single-rod channel

<table>
<thead>
<tr>
<th>Mesh</th>
<th>Outlet Bulk Temperature/K</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coarse</td>
<td>644.57824</td>
<td></td>
</tr>
<tr>
<td>Medium</td>
<td>647.28470</td>
<td>0.418%</td>
</tr>
<tr>
<td>Fine</td>
<td>645.30579</td>
<td>0.307%</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Mesh</th>
<th>Outlet Mass Flow Rate/kg/s</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coarse</td>
<td>0.0933kg/s</td>
<td></td>
</tr>
<tr>
<td>Medium</td>
<td>0.0972kg/s</td>
<td>4.01%</td>
</tr>
<tr>
<td>Fine</td>
<td>0.0956kg/s</td>
<td>1.67%</td>
</tr>
</tbody>
</table>

Table 3.4 Grid independence tests for multi-rod channel

<table>
<thead>
<tr>
<th>Mesh</th>
<th>Outlet Bulk Temperature/K</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coarse</td>
<td>656.13019</td>
<td></td>
</tr>
<tr>
<td>Medium</td>
<td>667.18932</td>
<td>1.66%</td>
</tr>
<tr>
<td>Fine</td>
<td>661.45416</td>
<td>0.867%</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Mesh</th>
<th>Outlet Mass Flow Rate/kg/s</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coarse</td>
<td>0.0963kg/s</td>
<td></td>
</tr>
<tr>
<td>Medium</td>
<td>0.0939kg/s</td>
<td>2.56%</td>
</tr>
<tr>
<td>Fine</td>
<td>0.0957kg/s</td>
<td>1.88%</td>
</tr>
</tbody>
</table>

3.4 Results and Discussions

Figure 3.5 shows the outlet velocity vectors colored by the velocity magnitude of the supercritical water in the single-rod channel and the multi-rod channel. It is shown that there are secondary flows at the outlet plane in the single-rod channel and the multi-rod channel, especially near the fuel rod cladding surfaces. Figure 3.6 shows the contours of the outlet velocity magnitudes in the single-rod channel and multi-rod channel. The velocity magnitudes at the outlets of these two channels are quite different. The velocity
magnitude in the single-rod channel is much lower than that in the multi-rod channel. The maximum velocity at the outlet is 4.5 m/s in the single-rod channel and 7 m/s in the multi-rod channel. The velocity field will affect the heat transfer in the channel. Therefore, using a single-rod channel to replace the multi-rod channel used in the Canadian SCWR will cause inaccurate results.

The cladding surface temperature distribution in the single-rod channel is shown in Figure 3.7. The difference in the cladding surface temperature along the circumference in the single-rod channel is less than 50K. Figure 3.8 shows the cladding surface temperature distribution in the multi-rod channel. It can be seen that the cladding surface temperature difference along the circumference in the multi-rod channel can reach to about 150K.

Figures 3.9 to 3.11 show the maximum and minimum cladding surface temperatures of each fuel rod at z=0.1 m, z=0.8 m, and z=1.5 m in the multi-rod channel. It can be seen that the maximum cladding surface temperature at these three planes all occur at the fuel rod #4. And the minimum cladding surface temperature at these three planes occur at the fuel rods #1, #5, #5, respectively. The largest difference of maximum and minimum cladding surface temperatures occurs at the fuel rod #4 at the outlet plane z=1.5 m, which is 78.537K. The maximum cladding surface temperature is 670K for the single-rod channel and 780K for the multi-rod channel. The difference is 110K. Therefore, the multi-channel model should be used in the simulation to generate more accurate data used for the control system designs since the maximum cladding surface temperature is an important parameter for the safety of nuclear reactors.
3.5 Conclusions

In this study, the fluid flow and heat transfer characteristics of the supercritical water in the single-rod channel and the multi-rod channel are compared. The results show that there are secondary flows in the single-rod and multi-rod channels. The maximum cladding surface temperature in the multi-rod channel is about 110K higher than that in the single-rod channel. Besides, the cladding surface temperature distributions are also not same between the single-rod channel and multi-rod channel. The difference of the circumference cladding surface temperature for the multi-rod channel can be up to 78.537K. The large circumferential temperature difference should be considered in the Canadian SCWR fuel bundle design. Since the heat transfer characteristics for the single rod channel and the multi-rod channel are not similar, the numerical simulations of the Canadian SCWR should be performed for the multi-rod channel in order to obtain more accurate results.
Figure 3.5 Outlet velocity vectors colored by the velocity magnitude in the single-rod and multi-rod channels (m/s)
Figure 3.6 Outlet velocity magnitude contours in the single-rod and multi-rod channels (m/s)
Figure 3.7 Cladding surface temperature distribution in the single-rod channel (K)

Figure 3.8 Cladding surface temperature distribution in the multi-rod channel (K)
Figure 3.9 Cladding surface temperatures at $z=0.1\, \text{m}$ (K)
Figure 3.10 Cladding surface temperatures at z=0.8m (K)
Figure 3.11 Cladding surface temperatures at z=1.5m (K)
References

USNERAC. (2002). A Technology Roadmap for Generation IV Nuclear Energy Systems. Retrieved from The U.S.DOE nuclear energy research advisory committee and the generation IV international forum:


Chapter 4  Closed-loop Control System Design for the Canadian SCWR

Abstract

The supercritical water-cooled reactor (SCWR) is one of the proposed six Generation IV nuclear reactors. The proposed Canadian SCWR is based on the concept of the CANDU reactor. However, the dynamic characteristics of the SCWR significantly differ from the CANDU reactor because of the supercritical water special heat transfer phenomenon in the fuel bundle. Thus, it is important to design a control system to maintain the SCWR operating at the required operating point when it is subjected to disturbances. Although the Canadian SCWR is a multiple-input and multiple-output system, the system can be treated as a single-input and single-output system at one operating point. Firstly, the transient CFD simulations of the Canadian SCWR using FLUENT are conducted in order to obtain the dynamic relationship between the inputs and outputs of the SCWR. Then the linear dynamic control models are constructed by the system identification technique based on the on dynamic relationship between the inputs and outputs obtained from the CFD simulations. The linear dynamic models are validated using the results from the full scale non-linear CFD simulations. Based on the linear dynamic models, a control system, which consists two PID controllers, is designed. And the performance evaluation of the proposed control system is carried in this work.
Keywords: SCWR, CFD, linear dynamic models, PID controllers
4.1 Introduction
The Supercritical Water-Cooled Reactor (SCWR) was proposed as one of the six Generation IV nuclear reactor in 2002 (USNERAC, 2002). Compared with the existing reactors, the size of the SCWR system is smaller. Also, since the water is heated by the reactor core to the supercritical steam, the SCWR can have the thermal efficiency up to 44% (Chow & Khartabil, 2008). There is no boiling crisis and steam generators or dryers, so the SCWR nuclear power plant can be simpler with fewer major components (USNERAC, 2002). The proposed Canadian SCWR is based on the CANDU reactor, which is a thermal spectrum reactor and the reactor is one of the pressure tube type, i.e., the tube arrangement is surrounded by the moderator. The moderator is a heavy water while the coolant is a light water.

Because of the abrupt properties change of the supercritical water around the pseudo-critical point, the heat transfer deterioration might occur, which will result in a higher cladding surface temperature. This will affect the safety of the reactor system. Therefore, it is important to have a reliable control system for the SCWR. In order to design a control system, the dynamic relationship between the system inputs and outputs is required. Since the Canadian SCWR is still in conceptual design stage, there are no experimental data available. Numerical method can be used to predict the fluid flow and heat transfer of the supercritical water in the fuel bundle under different input conditions. So, the input and output relationship can be obtained from the numerical simulations. Previous numerical studies on the SCWR are mainly focused on the flow and heat transfer phenomenon of supercritical fluids in circle channels. Zeng et al. (2013) conducted the computational fluid dynamics simulation of supercritical flow and heat transfer in a circular channel. Zhang et al. (2015) proposed a new supercritical fluid
flow model to better deal with the physical instability for the supercritical fluid around the pseudo-critical point. Maitri et al. (2014) also simulated the heat transfer of the supercritical water in circular channels using different turbulence models. And the results were compared with the experimental data to validate the numerical models. Zhang et al. (2011) simulated the heat transfer and flow of the supercritical water in a 37-element horizontal arranged SCWR under steady state conditions using the CFD models validated from the circular channel flows. In this work, the study of the fluid flow and heat transfer of the supercritical water in vertical fuel bundle used in the SCWR under the transient condition is carried to generate the dynamic relationship between the inputs and outputs, which will be used for the control system design.

In the previous studies on the control system for the SCWR by Nakatsuka et al. (1998) and Ishiwatari et al. (2003), the dynamic behaviors of the supercritical fast cooled reactor were analyzed by adding perturbances on three selected parameters: the control rod position, feedwater flow rate, and turbine control valve opening, respectively. Then, the input and output pairings for the reactor were found: the main steam pressure was controlled by the turbine control valves, the main steam temperature was controlled by the feedwater flow rate, and the core power was controlled by the fuel rods. Parameters of the control system are adjusted to satisfy both fast convergence and stability criteria. Then those parameters were optimized for reducing the overshoot with fast response. The results showed that the control system can maintain the supercritical water fast cooled reactor operating at the design point when the system is subjected to perturbances. Later, Ishitawari et al. (2003) used Nakatsuka's method in the control system design for the supercritical high temperature light water reactors.
Sun et al. (2012) simulated the thermo-hydraulic behavior of the Canadian SCWR using a simple 1D model. The direct Nquist array method was used to decouple the system and pre-compensators were used to convert the system to a diagonally dominant form. However, although the control system design was able to keep the system at stable condition when there were perturbances, the model used for the simulation of the reactor was too simple. Then, Sun et al. (2015) included a feed-forward control method in the control system design to reduce the effect of the reactor power on the steam temperature. The results showed that the steam temperature variation due to the disturbances on the reactor power could be significantly suppressed. Later, Sun et al. (2017) found that the response magnitude of the steam temperature to the same amount of feedwater flow rate disturbance at a high power level was smaller than that at a low power level. Then, the original control system design will be not effective when the working conditions changed. A linear parameter-varying strategy was proposed to solve such problems. The results showed that the linear parameter-varying (LPV) controller not only stabilized the steam temperature under different disturbances but also efficiently suppressed the steam temperature variation at different power levels. Maitri et al. (2017) used the results from CFD simulations of the supercritical water flow in a circular cube by FLUENT to derive the linear dynamic models around the operating point based on the system identification techniques. Then the linear dynamic models were validated with the CFD results in the 2D tube flows.

To design a control system for the SCWR, accurate models are needed to describe dynamic behaviors of the Canadian SCWR. Heat transfer and fluid flow behaviors of supercritical water are complicated in reactor fuel bundles. Although the Canadian SCWR is a multi-input and multi-output system, the coupling of inputs and outputs can
be ignored when SCWRs work around the desired operating point. An effective approach for the controller design is to linearize the nonlinear process. There are many different methods to linearize a nonlinear process. The approach used in this work is "small signal models", i.e. A low amplitude disturbance is applied to the normal input in order to obtain the respective output. Then a linear dynamic model of the input and the output can be obtained by system identification techniques. Based on the linear dynamic model, controllers can be designed.

In this work, the linear dynamic models are derived from inputs and outputs data from full-scale transient CFD simulations where small perturbances are applied due to lack of the experimental data. The CFD models used in this work for the simulations of the supercritical fluid flow and heat transfer were validated in the previous works by Zhang et al. (2011).

4.2 Configuration and Operating Conditions of the Canadian SCWR

The Canadian SCWR system is shown in Figure 4.1. The moderator for the reactor is the heavy water and the coolant is the light water, which is the supercritical water. The specifications for Canadian SCWR (Leung, 2013) is shown in Table 4.1. Based on the current Canadian SCWR design (Heavy Water Reactors, 2002), the fuel bundle consists of 37 fuel rods, which are evenly distributed, as shown in Figure 4.2. The fuel bundle is vertically arranged. The geometrical parameters for the fuel bundle are given in Table 4.2 (Zhang et al., 2011). The first, second, and third rings of the fuel rods shown in Table 4.2 are marked in the cross-section view in Figure 4.2. The properties of supercritical water, such as density, specific heat, thermal conductivity, viscosity, are from Wagner

Figure 4.1 Canadian SCWR system (Leung, 2013)

Table 4.1 Operating parameters of the Canadian SCWR (Zhang et al., 2011)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant</td>
<td>Light water</td>
</tr>
<tr>
<td>Heat Flux</td>
<td>1000kw/m²</td>
</tr>
<tr>
<td>Inlet Velocity</td>
<td>1.18m/s</td>
</tr>
<tr>
<td>Number of fuel rods</td>
<td>37</td>
</tr>
<tr>
<td>Efficiency</td>
<td>48%</td>
</tr>
<tr>
<td>Fuel</td>
<td>UO2/TH</td>
</tr>
<tr>
<td>Inlet temperature</td>
<td>350°C</td>
</tr>
<tr>
<td>Cladding temperature</td>
<td>&lt;850°C</td>
</tr>
<tr>
<td>Heated length</td>
<td>1m</td>
</tr>
<tr>
<td>Operating pressure</td>
<td>25MPa</td>
</tr>
</tbody>
</table>
4.3 Governing Equations

The fluid flow and heat transfer of the supercritical water in the Canadian SCWR fuel bundle can be described by the conservation equation of mass, conservation equation of momentum, and conservation equation of energy as shown below (ANSYS, 2011):

\[
\frac{\partial (\rho u_i)}{\partial x_i} = 0 \tag{4.1}
\]

\[
\frac{\partial (\rho u_i u_j)}{\partial x_j} = -\frac{\partial p}{\partial x_i} + \frac{\partial}{\partial x_j} (\mu \frac{\partial u_i}{\partial x_j} - \rho u_i u_j) \tag{4.2}
\]

\[
\frac{\partial}{\partial x_i} (u_i \rho C_p T) = \frac{\partial}{\partial x_j} [(\lambda + \frac{C_p \mu_t}{P_r}) \frac{\partial T}{\partial x_j}] + S \tag{4.3}
\]

where \( u \) is the velocity, \( T \) is the temperature, \( \mu \) is the dynamic viscosity, \( \rho \) is the density, \( \lambda \) is the thermal conductivity, \( C_p \) is the specific heat, \( \mu_t \) is the turbulent viscosity, \( P_r \) is the turbulent Prandtl number. Zhang et al. (2011) found the numerical results obtained by Reynolds Stress Model (RSM) agree well with experimental data for the supercritical water flows in vertical tubes. Thus, the RSM with the enhancement
wall function is used for solving the Reynolds stress term in this simulation which includes the thermal and full buoyancy effects, and the viscous heating. The transportation equation of the RSM is shown as follows (ANSYS, 2011):

\[
\frac{\partial}{\partial t} (\rho u_i u_j) + \frac{\partial}{\partial x_k} \left[ \rho u_j u_k - u_i u_k \right] = -\frac{\partial}{\partial x_k} \left[ \rho u_i u_k + \rho \left( \delta_{ij} + \delta_{ik} u_j \right) + \frac{\partial}{\partial x_j} \left[ \mu \frac{\partial u_i}{\partial x_k} \right] \right] + \frac{\partial}{\partial x_k} \left[ \mu \frac{\partial u_i}{\partial x_k} \right]
\]

\[
-\rho \left( \frac{\partial u_i}{\partial x_k} + \frac{\partial u_j}{\partial x_j} \right) - \rho \beta (g \mu_i \theta + g \mu \theta) + \frac{p}{\rho} \left( \frac{\partial u_i}{\partial x_j} + \frac{\partial u_j}{\partial x_i} \right) - 2 \mu \frac{\partial u_i}{\partial x_j} \frac{\partial u_j}{\partial x_k} + S_{\text{user}}
\]  

(4.4)

4.4 Full-scale CFD Simulations of the Canadian SCWR

In the CFD simulation, the governing equations are solved by discretization method based on the control volume concept. ANSYS FLUENT 15.0 is used to solve those equations to produce inputs and outputs of the system, which show the dynamic characteristics of the fluid flow and heat transfer process of the supercritical water in the Canadian SCWR fuel bundle. The SIMPLEC scheme is selected for pressure correction, and QUICK method is considered for conducting spatial discretization. The convergence criteria for continuum is \(10^{-3}\), for the momentum and turbulence equations are both \(10^{-5}\), and for the energy equation is \(10^{-6}\). The supercritical water flows upward along the fuel bundle. And the heated length is reduced to 1m to save simulation time.

Boundary conditions are as follows:
Inlet: The inlet velocity is 1.18m/s and the inlet temperature is 623.15K. Turbulence intensity at the inlet assumed to be 6%. The hydraulic diameter is 7.416mm based on
the fuel bundle parameters shown in Table 4.2.

Outlet: Outflow is selected.

Walls: No-slip smooth wall condition is used.

The simulation is carried out for a quarter of the fuel bundle due to the symmetrical geometry of the fuel bundle. The computational domain and the coarse mesh are shown in Figure 4.3 and Figure 4.4, respectively.

The steady simulation for the fuel bundle is conducted for the grid independent tests. And the comparison of the mesh qualities and the grid independent test results are shown in Table 4.3 and Table 4.4. It can be seen from Table 4.4 that the difference in the results from the medium mesh and fine mesh is less than 1%. Therefore, the medium mesh with 334,290 nodes is chosen for transient simulations in this work. The simulation outputs for time step independent tests are shown in Figures 4.5 and 4.6. It can be seen the differences of results from three tests are small. Thus, time step size 0.01s is employed in the transient simulation. And the max iteration for each time step is 20.

Table 4.2 Geometrical parameters for the fuel bundle simulation (Zhang et al., 2011)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Rod Diameter/mm</td>
<td>13.081</td>
</tr>
<tr>
<td>Fuel Bundle Diameter/mm</td>
<td>103.378</td>
</tr>
<tr>
<td>Length/mm</td>
<td>1000</td>
</tr>
<tr>
<td>First Ring Diameter/mm</td>
<td>29.769</td>
</tr>
<tr>
<td>Second Ring Diameter/mm</td>
<td>57.506</td>
</tr>
<tr>
<td>Third Ring Diameter/mm</td>
<td>86.614</td>
</tr>
</tbody>
</table>
Table 4.3 Mesh qualities for the fuel bundle

<table>
<thead>
<tr>
<th>Mesh Size</th>
<th>Size</th>
<th>Coarse</th>
<th>Medium</th>
<th>Fine</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nodes</td>
<td>150246</td>
<td>334290</td>
<td>512074</td>
<td></td>
</tr>
<tr>
<td>Cells</td>
<td>605022</td>
<td>2010134</td>
<td>3319162</td>
<td></td>
</tr>
<tr>
<td>Faces</td>
<td>1831244</td>
<td>3634805</td>
<td>728615</td>
<td></td>
</tr>
</tbody>
</table>

Table 4.4 Grid independence tests for the fuel bundle

<table>
<thead>
<tr>
<th>Mesh</th>
<th>Outlet Bulk Temperature/K</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coarse</td>
<td>669.22018</td>
<td></td>
</tr>
<tr>
<td>Medium</td>
<td>677.15363</td>
<td>1.17%</td>
</tr>
<tr>
<td>Fine</td>
<td>681.41892</td>
<td>0.626%</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Mesh</th>
<th>Outlet Mass Flow Rate/kg/s</th>
<th>Difference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coarse</td>
<td>0.648kg/s</td>
<td></td>
</tr>
<tr>
<td>Medium</td>
<td>0.638kg/s</td>
<td>1.57%</td>
</tr>
<tr>
<td>Fine</td>
<td>0.652kg/s</td>
<td>2.15%</td>
</tr>
</tbody>
</table>

Figure 4.3 Cross-section view of the computational domain of the fuel bundle
Figure 4.4 Cross-section view of the mesh of the fuel bundle

Figure 4.5 Comparison of outlet mass flow rate using different time steps
The cladding surface temperature distributions are shown in Figure 4.7. The maximum cladding surface temperature when the load is 100% is about 960K, which is below the limit of 1123.15K (850°C). Figure 4.8 to 4.10 show the maximum cladding surface temperatures along the circumference of the fuel rods at $z=0.1m$, $z=0.5m$, and $z=0.9m$ respectively. As can be seen from the figures, three highest cladding surface temperatures at $z=0.1m$, which is close to the inlet of the fuel bundle, occur at the fuel rod #3, #10, and #9. At $z=0.5m$, the three highest cladding surface temperatures are at the fuel rod #1, #2, and #3. The three highest cladding surface temperatures at $z=0.9m$, which is close to the outlet, are 803.774K (rod #2), 765.918K (rod #3), 763.339K (rod #1). Besides, Figure 4.11 shows the minimum cladding surface temperature at $z=0.9m$. Comparing Figures 4.10 and 4.11, it can be seen the maximum cladding temperature difference in one fuel rod in the fuel bundle at $z=0.9m$ is 128.30K at fuel rod #2. So, the non-uniformity in the temperature distribution around the rod is very high for this 37-
element fuel bundle Canadian SCWR.

Figure 4.7 Cladding surface temperature distributions (K)

Figure 4.8 z=0.1m Cladding surface temperature maximum value (K)
Figure 4.9 $z=0.5\text{m}$ Cladding surface temperature maximum value (K)

Figure 4.10 $z=0.9\text{m}$ Cladding surface temperature maximum value (K)
4.4 Linear Dynamic Models Construction

The governing equations for the fluid flow and heat transfer of the supercritical water in the fuel bundles cannot be used for the design of the feedback control system directly. To design a control system, the dynamic relationship between the inputs and outputs of the system is required. So, the linear dynamic control models can be constructed. Due to the lack of the experimental data, the dynamic relationship between the inputs and outputs of the SCWR fuel bundle will be obtained by the full-scale CFD simulations of the SCWR fuel bundle. And then these linear dynamic models can be used to design the controllers.
4.4.1 Construction of Transfer Functions

The fluid flow and heat transfer process in the reactor fuel bundle can be treated as a dynamic process with two inputs and two outputs. The input variables are the inlet mass flow rate of the supercritical water and heat flux from the fuel rods. The output variables are the outlet mass flow rate and the outlet bulk temperature of the supercritical water. The control system is used to keep the system operated at the design point. The dynamic process in the SCWR fuel bundle is demonstrated in Figure 4.12.

The procedure for constructing the linear dynamic models is: a 10% step change for each input variable is applied separately, which means only one of these two inputs is changed at once and another one is kept at the design value. The outputs responses are recorded from the CFD simulations, which are shown in Figures 4.13 and 4.14 when the inlet mass flow rate has a 10% step decrease, and Figures 4.15 and 4.16 when the heat flux has a 10% step decrease.

![Diagram](image)

**Figure 4.12 Dynamic process in the Canadian SCWR fuel bundle**
Figure 4.13 Inlet and outlet mass flow rate variations when the inlet mass flow rate has a 10% step decrease

Figure 4.14 Outlet bulk temperature of the fuel bundle variations when the inlet mass flow rate has a 10% step decrease
Figure 4.15 Inlet and outlet mass flow rate variations when the wall heat flux has a 10% step decrease

Figure 4.16 Outlet bulk temperature of the fuel bundle variations when the wall heat flux has a 10% step decrease
The system identification technique then is used to obtain the transfer functions between the inputs and outputs. The relevant transfer functions for the linear dynamics models are shown in Laplace forms as:

\[
\begin{align*}
G_{\text{C011}} &= \frac{3289e^6 + 6.183e^4s + 1.126e^3s^2 + 1.235e^2s^3 + 9.781e^2s^4 + 5.716e^2s^5 + 2.327e^2s^6 + 6.496e^2s^7 + 1.127e^2s^8 + 8.836e^2}{s^6 + 3270e^4s^4 + 6.308e^4s^5 + 1.114e^4s^6 + 1.257e^4s^7 + 9.67e^4s^8 + 5.804e^4s^9 + 2.302e^4s^10 + 6.613e^4s^11 + 1.1e^4s^12 + 8.836e^2} \\
G_{\text{C012}} &= \frac{1.423e^6s^2 + 3.829e^6s + 6.138e^6}{s^2 + 9216s^3 + 1.512e^4s + 5155} \\
G_{\text{C021}} &= \frac{0.001456s + 0.001528}{s^4 + 8.624e^2s^2 + 67.97e^4s^4 + 334.23e^4s^6 + 1216e^4s^8 + 3073e^4s^{10} + 4881e^4s^{12} + 4780s + 2150} \\
G_{\text{C022}} &= \frac{0.4332s + 0.1323}{s^7 + 9.696e^4s^3 + 60.95e^4s^5 + 245.7e^4s^7 + 602.2e^4s^9 + 894.3e^4s^{11} + 819.2e^4s^{13} + 177.5}
\end{align*}
\]

where:

- \( G_{11} \) is the transfer function for the inlet mass flow rate and the outlet mass flow rate;
- \( G_{12} \) is the transfer function for the inlet mass flow rate and the outlet bulk temperature;
- \( G_{21} \) is the transfer function for the wall heat flux and the outlet mass flow rate;
- \( G_{22} \) is the transfer function for the wall heat flux and the outlet bulk temperature.

### 4.4.2 Validation of the Transfer Functions

The transfer functions derived from the data sets of CFD simulations need to be validated to make sure they can truly represent the dynamic characteristics of the process in the SCWR fuel bundle. Figures 4.17 to 4.20 show the comparisons of the results from the linear dynamic models with those from the non-linear CFD simulations for the responses of the output variables when step changes are applied on input
variables.

Figure 4.17 Comparisons of the responses of the outlet mass flow rate between the linear dynamic model and non-linear CFD simulations when the inlet mass flow rate decreases 10%.

Figure 4.18 Comparisons of the responses of the outlet bulk temperature between the linear dynamic model and non-linear CFD simulations when the inlet mass flow rate decreases 10%.
Figure 4.19 Comparisons of the responses of the outlet mass flow rate between the linear dynamic model and non-linear CFD simulations when the wall heat flux decreases 10%.

Figure 4.20 Comparisons of the responses of the outlet bulk temperature between the linear dynamic model and non-linear CFD simulations when the wall heat flux decreases 10%.
As these figures show, although there are minor discrepancies for the results from linear dynamic models at first, the results from the linear dynamic models can agree well with the responses from the CFD simulations generally. It can be concluded that these linear dynamic models can indeed represent the non-linear dynamic behaviors of the Canadian SCWR fuel bundle when the disturbance on the input variables is low (10%).

4.5 Control System Design and Validation

4.5.1 Controller Design

The Canadian SCWR system can be subjected to different disturbances during the normal operation. Thus, an appropriate control system is needed to regulate the inputs to make sure the reactor can return to the design point quickly. If the coupling relations of all inputs and outputs are included, the control system design will be very complicated. To simplify the controller design process, the input that has the bigger influence on one output is assumed to dominate the output variable. So, the reactor system can be treated as a multiple single-input-single-output (SISO) system. Since the outlet mass flow rate mainly depends on the inlet mass flow rate, the outlet bulk temperature primarily depends on the input heat flux, two PID controllers are needed to control the Canadian SCWR fuel bundle outputs. The scheme of the closed-loop control system for the SCWR is shown in Figure 4.21.
The closed-loop feedback control system is designed to maintain the controlled variable staying around the design point as much as possible when disturbances occur. The control system is used and the adjustment will stop only after the deviation is zero. The transfer function used in the PID controller takes the following form (Nise, 2000):

\[
C(s) = K_P + \frac{K_I}{s} + K_Ds
\]  

(5.5)

Here, \(K_P\), \(K_I\), and \(K_D\) are the controller dominant parameters. And the parameters for this 37-element Canadian SCWR under full load working condition are shown in Table 4.5. The parameters are tuned by trial and error method. The method requires a closed loop system stepping through the system from proportional to integral to derivative. At the beginning, each parameter of the PID controller is set to zero. Then
the proportional term is considered by increasing its value until obtaining a steady error
is obtained. The theoretical proportional term should be half of the value. The integral
term is increased until steady errors are again obtained. The theoretical integral term
should be three times of the value. Then for the derivative control, this derivative value
is increased until the errors are at a constant period and amplitude for a final time.
Finally, the theoretical derivative term is one third of the value. The parameters for the
two PID controllers are presented in Table 4.6. It is shown that the maximum overshoot
is below 10% and the rise time is under 1s. This satisfies the basic limitations of SISO
systems.

Table 4.5 Parameters for the controllers

<table>
<thead>
<tr>
<th>Parameters Controllers</th>
<th>$K_P$</th>
<th>$K_I$</th>
<th>$K_D$</th>
</tr>
</thead>
<tbody>
<tr>
<td>PID₁</td>
<td>1.558</td>
<td>9232</td>
<td>0</td>
</tr>
<tr>
<td>PID₂</td>
<td>353.8</td>
<td>722.4</td>
<td>43.32</td>
</tr>
</tbody>
</table>

Table 4.6 Control system characteristics

<table>
<thead>
<tr>
<th></th>
<th>Rise Time/s</th>
<th>Settling Time/s</th>
<th>Overshoot</th>
</tr>
</thead>
<tbody>
<tr>
<td>PID₁</td>
<td>0.000259</td>
<td>0.000921</td>
<td>6.31%</td>
</tr>
<tr>
<td>PID₂</td>
<td>0.9</td>
<td>8.28</td>
<td>7.51%</td>
</tr>
</tbody>
</table>

4.5.2 Performance Evaluation of the Controllers

The control system based on the linear dynamic models are used to bring the reactor to
the desired operating point when it is subjected to disturbances. Thus, the performance
evaluation for the control system under the nonlinear environment is very important. In
this work, the performance evaluation procedure is as follows. First, a small disturbance
(10% of the design value) is applied on the full-scale nonlinear CFD simulation and the
linear dynamic models from the desired point till $t=10$s simultaneously. Then the
controller is activated to return the reactor to the design value. This can be achieved by
combining the control system into the CFD simulation process. The output data by the
CFD simulations at each time step are used as the control system outputs. And then the controllers adjust the input variables accordingly in the next time step. Actually, the CFD simulation and the controllers are operating in a closed-loop mode. The time step for CFD simulation and the sampling interval for the control system are both 0.05s. The results for the outlet mass flow rate and the outlet bulk temperature from the CFD simulations are shown in Figures 4.22, 4.23, and 4.24, 4.25 respectively. The system can return to its desired point in 20s. It can be seen the designed controller works well to keep the reactor working at the design point when there are small disturbances.

Figure 4.22 Closed-loop responses of the outlet mass flow rate under 10% perturbances of the inlet mass flow rate using CFD model
Figure 4.23 Closed-loop responses of the outlet bulk temperature under 10% perturbances of the inlet mass flow rate using CFD model

Figure 4.24 Closed-loop response of the outlet mass flow rate under 10% perturbances of the wall heat flux using CFD model
4.6 Conclusions

It is important to maintain the Canadian SCWR working safely. Thus, the dynamic relationships of inputs and outputs are needed for the control system design. In this work, the transfer functions used for the linear dynamic models of the Canadian SCWR are constructed based on the data from the full-scale CFD simulation using the system identification technique. These linear dynamic models are then fully validated by comparing with results from full-scale nonlinear CFD simulations. A closed-loop control system is designed to return the reactor to the design point when there are perturbances. The control system consists of two PID controllers which are synthesized based on the linear dynamic models. And the performance evaluation of the control
system for the Canadian SCWR is performed. The results show that the proposed control system is able to return the Canadian SCWR system to the design point when perturbances are introduced.
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Chapter 5  Conclusions and Future Work

5.1 Conclusions

A closed-loop control system for the Canadian SCWR is constructed based on CFD simulations. This research deeply investigates the supercritical water flow characteristics and heat phenomenon in 37-element Canadian supercritical water-cooled reactor fuel bundles. Firstly, the fluid flow and heat transfer characteristics of supercritical water in the single-rod channel and the multi-rod channel are compared. The results show that there are secondary flows in the single channel. The maximum cladding surface temperature for the single rod channel and the multi-rod channel is about 670K and 780K, respectively. Besides, the cladding surface temperature distributions are also not same for the single fuel rod and the multi rods. The gradient of the circumference cladding surface temperature for the multi-rod channel can be up to 78.537K. Although the maximum cladding surface temperatures of these two channels are below the designed limit value. The large circumferential temperature gradient should be considered in the Canadian SCWR fuel rods arrangement design procedures. Since the heat transfer characteristics for the single rod channel and the multi-rod channel are not similar. The numerical simulations for supercritical water in the Canadian SCWR should adopt the multi-rod channel types to obtain more accurate results.

Then, a closed-loop control system is constructed based on the dynamic relationships of inputs and outputs determined by the full-scale CFD simulations. Then the thermo-hydraulic behaviors of the supercritical water in the 37-element vertical Canadian
SCWR tube bundles under full load condition are numerically predicted. The cladding surface temperature distribution is still not uniform for each fuel rod although the maximum value is below the design value. The transfer functions of linear dynamic models of inputs and outputs of the Canadian SCWR is constructed by the system identification techniques. These linear dynamic models are then fully validated by comparing with the results from the nonlinear CFD simulations. The validation results show that the designed linear dynamic models can agree around 80-95% accuracy with the CFD simulation results. Two PID controllers are synthesized based on the linear dynamic models. And the performance evaluations of the control system for the Canadian SCWR is presented. The results show that the designed closed-loop control system can return the Canadian SCWR to the design point quickly when disturbances occur suddenly.

5.2 Future Work

Based on the research conducted in this thesis, some suggestions as follows are made for the future work:

1. Validation of the numerical simulations results of the supercritical water in the Canadian SCWR fuel bundles.

Currently most experimental researches for the supercritical water were done in circle channels or annular channels. The differences of the thermos-hydraulic behaviors of the supercritical water in the single-rod channel and the multi-rod channel are obvious. Since the Canadian SCWR system is still in the conceptual process, the experimental
data are really needed in the future to validate the numerical results of the supercritical water heat transfer process.

2. Transient simulations of the supercritical water in the fuel bundles under different operation conditions
The work in this thesis only simulate behaviors under the conditions of 100% nuclear reactor's load. Much more researches should be investigated on varying processes to make sure the design requirements satisfying, such as the startup and shutdown procedures.

3. A closed-loop control design for the whole Canadian SCWR reactor system
The control system in this thesis focuses on the nuclear reactor part. And the supercritical water in the reactor is assumed as a single-input single-output process. The whole nuclear reactor system consists of reactor and turbine parts. Much more inputs and outputs are needed to be included to complete a whole control system design based on linear dynamic models.
Curriculum Vitae

Name: Huirui Han

Post-Secondary Education and Degrees:
Bachelor of Engineering, Thermal Energy and Power Engineering
North China Electric Power University
Changping, Beijing, China
2011-2015

M. E. Sc, Mechanical & Materials Engineering
The University of Western Ontario
London, Ontario, Canada
2016-2018

Honors and Rewards:
Excellent Student Scholarship
North China Electric Power University
2011-2014

Related Work Experience:
Teaching Assistant
The University of Western Ontario
2016-2017