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Modeling the partial loss of coolant flow accident in the Super-critical water reactor

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HIGHLIGHTS

- A thermal-hydraulic computer code was developed to analyze the SCWR reactor core in transient mode.
- Porous media approach for a thermal-hydraulic analysis of the SCWR core has been implemented.
- In the new design, the amount of reactor thermal power at the end of the transition reaches 5.5% of the initial value.
- The results have been improved compared with those of Oka's design.

ABSTRACT

In this study, thermal-hydraulic analysis of partial loss of coolant flow accident in supercritical pressure light water reactor (SCWR) with a new geometric design has been investigated. In the new design, the coolant and moderator circuits are separated. This analysis was performed using the development of a transient-state thermal-hydraulic code in which the equations of mass, momentum, and energy are solved. The porous Media approach is used to solve these equations. By extracting the results of transition modeling, it is observed that in the new geometric design, by separating the coolant and moderator circuits, the maximum fuel clad temperature is lower than the maximum fuel clad temperature value of the previous designs. As in the new design at the end of the transition, the maximum fuel clad temperature has decreased by about 37% compared to the initial state. The result of the calculations in this study shows that the new design, in which the coolant and moderator circuits are separated, has created more safety in a chosen transition.

KEYWORDS

SCWR
Thermal-hydraulic
Porous media approach
Moderator circuit

HISTORY

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Nomenclature

A	Area (m^2)
D	Diameter (m)
$\vec{g}$	Acceleration of gravity ($m.s^{-2}$)
h	Enthalpy ($kJ.kg^{-1}$)
k	Thermal conductivity ($W.m^{-1}.°C^{-1}$)
P	Pressure (Pa)
$\bar{\bar{\tau}}$	Shear stress tensor ($Pa.m.s^{-1}$)
$\vec{R}$	Distributed resistance ($N.m^3$)
q''	Heat flux ($W.m^{-2}$)
q'''	Volumetric heat ($W.m^{-3}$)
Φ	Dissipation function ($W.m^{-3}$)
t	Time (s)
T	Temperature (°C)
$\vec{v}$	Velocity ($m.s^{-1}$)
V	Volume (m^3)
ρ	Density ($kg.m^3$)
z	Length (m)
Nu	Nusselt number
Pr	Prandtl number
Re	Reynolds number

Subscripts

B	Bulk
h	Hydraulic
i	Counter
f	Fluid or fuel
T	Total
rb	Dispersed
W	Wall

1 Introduction

Nuclear technology after several decades of rapid evolution has passed the stages of technological maturity and is now a tried and tested technology. The main application of a nuclear reactor is to generate electricity, and its other applications, such as hydrogen production, seawater desalination, etc., are potentially important soon. Light

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water reactors are currently one of the largest projects of nuclear power plants. Their structure consists of a pressure chamber, control rods, a holding structure, steam turbines, feedwater pumps, an emergency core cooling system, and so on.

The fourth-generation reactors, which are expected to enter the global market by 2030, are still in the design and testing phase. This generation of reactors has been started in international cooperation to promote safety, economic recovery, reducing the production of nuclear waste, and reducing the risk of producing nuclear weapons (IAEA, 2002).

One of the fourth generation reactors studied is supercritical pressure reactors (SCWR). According to the proposed design, the core of a light water reactor is placed in a supercritical fossil fuel power plant as a boiler. Its direct-cycle turbine is assembled, and the temperature of the reactor core outlet water temperature rises to 500 °C. In a water phasic diagram, areas above the critical point are called supercritical. The critical point for water occurs at a pressure of 22.1 MPa and a temperature of 374.2 °C. Under supercritical pressure conditions, the thermodynamic properties of water around the quasi-critical point changed dramatically. The quasi-critical temperature is the temperature that indicates the maximum amount of fluid-specific heat at the critical pressure. At this point, the fluid-specific heat rises sharply. The highest specific heat of water occurs at 25 MPa at 384 °C. At quasi-critical temperature, because the fluid-specific heat has a maximum value, therefore, the greatest changes in coolant density are observed in this range. Figure 1 shows the changes in water properties near the quasi-critical point at 25 MPa.

Supercritical pressure light water reactors (SCWR) are the new generation of nuclear reactors being designed. These types of reactors are at high coolant pressure and high temperature. Coolant pressure in these reactors is higher than critical pressure (22.1 MPa). Therefore, the coolant fluid (light water) is single-phase and no boiling is seen in it. An outline of the old design of SCWR reactors with a circuit containing coolant and moderator is shown in Fig. 2.

In the new design of SCWR reactors, according to Fig. 3, water enters the pressurized chamber from the moderator inlet as a moderator and passes through the moderator channels in an independent circuit through the fuel assemblies, and finally, the fluid exits from the moderator outlet. Water enters as a coolant from the coolant inlet at the bottom of the pressure chamber and finally streams to the Steam plenum to remove reactor core heat.

In the proposed new model (SCWR-ND), by changing the mass flow rate through the moderator channels, it will be possible to control the reactor and prevent sudden power changes in emergencies and transient accidents. The conceptual design of a light water reactor operating at supercritical pressure was first studied in 1990 at the University Of Tokyo, Japan, and preliminary analyzes using a single-channel heated model show the reactor outlet temperature of the reactor core Reaches 416 °C and the reactor efficiency reaches 41.2% (Oka et al., 1992; Okano et al., 1994; Dobashi et al., 1997).

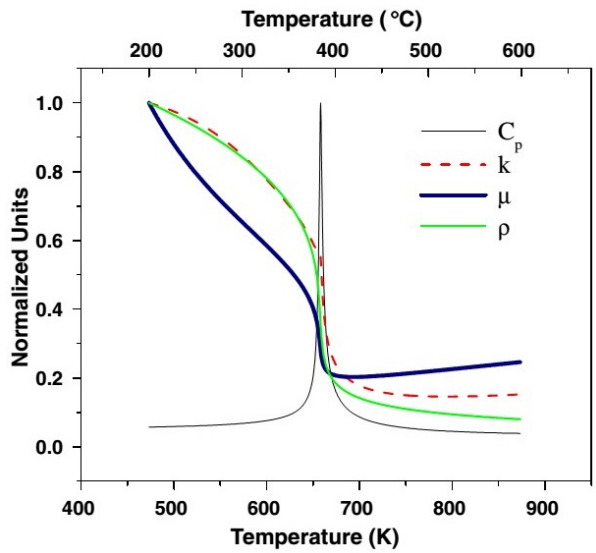

Figure 1: Changes in water properties near the quasi-critical point at MPa 25 (Licht et al., 2008).

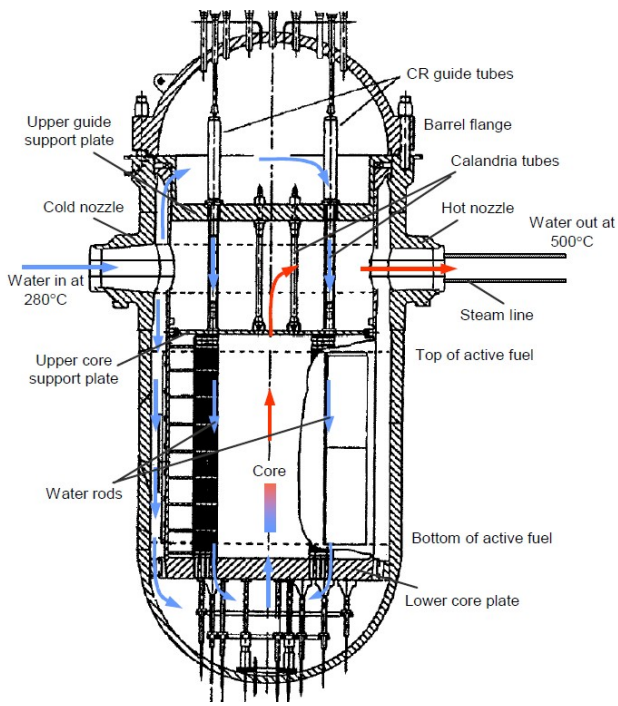

Figure 2: General design of SCWR reactor with integrated coolant and moderator circuit (Buongiorno and MacDonald, 2003).

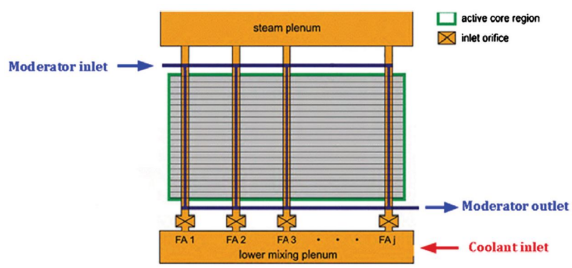

Figure 3: View of the coolant and moderator circuits in the new design for the HPLWR reactor (Jahanfarnia et al., 2013).

One of the design challenges of reactors with supercritical pressure is neutron moderation reduction by the moderator fluid in the upper part of the reactor core. This challenge was resolved by Okano et al. (Okano et al., 1994) by introducing moderator secondary channels in fuel assemblies. The first suggestion for neutron-thermal-hydraulic coupling was presented by Broeders et al. in 2003 (Broeders et al., 2003). The first thermal-hydraulic analysis of square and hexagonal lattices of SCWR reactor fuel rods using a sub-channel approach was proposed by Cheng et al. (Cheng et al., 2003) in 2003 with no neutron-thermal hydraulic coupling.

Numerous neutron and thermal-hydraulic calculations have been performed to investigate the transient state response of the SCWR reactor. In 2013, a thermal-hydraulic code was developed by Liu et al. (Liu et al., 2013a) to investigate the various transients in the SCWR reactor. The TACOS calculation code was developed at Xian Jiaotong University to analyze the transient state of SCWR reactors. This code is used to study the transient state of the SCWR-M reactor by Zhu et al. (Zhu et al., 2013), to compare different designs for supercritical water concept reactors. Thermal-hydraulic analysis of the flow obstruction in the supercritical reactor fuel assemblies was performed by Liu et al. (Liu et al., 2013b) in 2013, using a code with a sub-channel approach. In this study, the challenges of changing the physical properties of water around a quasi-critical point have been addressed using various methods and equations. The STTA code is a 3-D code for transient state analysis of the SCWR reactor developed by Lianjie et al. (Wang et al., 2015) in 2015. This code is the three-dimensional code of neutron spatial kinetics NGFMN_K and the thermal-hydraulic code ATHAS coupling.

The present study aims to investigate the thermal-hydraulic analysis of the SCWR reactor core with a new design in the transient mode of partial loss of coolant flow accident. Since the existing thermal-hydraulic codes cannot analyze the SCWR reactor core, so it is necessary to develop a thermal-hydraulic computer code that can analyze the SCWR reactor core in transient mode. Also, in the developed computer code, the thermal-hydraulic conditions and properties of water should be considered as coolant and moderator at supercritical pressure, as well as heat transfer between the coolant and moderator channels. Generally, the geometry used in the study by Bahrevar et al. (Bahrevar et al., 2018), used in this study. The thermal power of Bahrevar et al. (Bahrevar et al., 2018) study has been used for code input. In the study of Bahrevar et al. (Bahrevar et al., 2018), the steady-state operation of the new design SCWR reactor has been done by the sub-channel method. In this paper, the same issue has been accomplished for the transient state by the porous media approach and the investigation of the accident results.

2 Methods and Materials

To model the complex thermal-hydraulic problems of the reactor core, generally, two approaches are used to investigate the thermal-hydraulic behavior of the coolant flow, these two approaches are 1-porous media approach and 2-sub-channel approach. From an engineering point of view, the sub-channel approach is a simplified version of the porous media approach, assuming axial flow prevails in the system. Given that the sub-channel approach has been used in several studies, but there is always doubt about the correctness of the sub-channel approach in the thermal-hydraulic analysis of fluid, especially during severe cross-flow, such as the clogged fluid channel. For this reason, the porous media approach has been studied and developed to improve the prediction of fluid behavior. The final conservation equations of mass, momentum, and energy (a function of enthalpy) will be equal to (Todreas and Kazimi, 2001):

$$\frac{\partial \rho}{\partial t} + \nabla.(\rho \vec{v}) = 0 \tag{1}$$

$$\frac{\partial (\rho \vec{v})}{\partial t} + \nabla.(\rho \vec{v}\vec{v}) = -\nabla P + \nabla.\bar{\bar{\tau}} + \rho \vec{g} \tag{2}$$

$$\frac{\partial (\rho h)}{\partial t} + \nabla.(\rho h \vec{v}) = -\nabla \vec{q''} + q''' + \frac{DP}{Dt} + \Phi \tag{3}$$

To explain the porous media approach, an environment is considered, which this environment containing a single-phase fluid in which a solid is distributed. Two sets of relations are needed to obtain the required equations in the porous environment method. These two are a set of definitions of environmental porosity and a set of mathematical theorems. Figure 4 depicts the environment consisting of a single-phase fluid with a stationary solid.

The ratio of fluid volume V_f to total volume V_T is called volume porosity, in other words:

$$\gamma_V = \frac{V_f}{V_T} \tag{4}$$

Using the gas density function and volume averaging on the equations of mass, momentum, and energy, and using the definition of volumetric porosity, the triple equations are transformed into Eqs. (5), (6), and (7):

$$\gamma_V \frac{\partial\, {}^i\langle \rho \rangle}{\partial t} + \frac{1}{V_T} \int_{A_f} \rho \vec{v}.\vec{n}\, \mathrm{d}A = 0 \tag{5}$$

$$\begin{aligned} \gamma_V \frac{\partial\, {}^i\langle \rho \rangle}{\partial t} + \frac{1}{V_T} \int_{A_f} \rho \vec{v}(\vec{v}.\vec{n})\, \mathrm{d}A = \gamma_V\, {}^i\langle \rho \rangle \vec{g} \\ + \frac{1}{V_T} \int_{A_f} \big(-P.\vec{n} + \bar{\bar{\tau}}.\vec{n} \big)\, \mathrm{d}A + \gamma_V\, {}^i\langle \vec{R} \rangle \end{aligned} \tag{6}$$

$$\begin{aligned} \gamma_V \frac{\partial\, {}^i\langle \rho h \rangle}{\partial t} + \frac{1}{V_T} \int_{A_f} \rho h \vec{v}.\vec{n}\, \mathrm{d}A = \frac{1}{V_T} \int_{A_f} k_e \vec{n}.\nabla T\, \mathrm{d}A \\ + \gamma_V\, {}^i\langle \frac{DP}{Dt} \rangle + \gamma_V \Big({}^i\langle q''_{rb} \rangle + {}^i\langle q''' \rangle + {}^i\langle \Phi \rangle \Big) \end{aligned} \tag{7}$$

The resistance $\vec{R}$ in Eq. (6), which is a key concept in the porous media approach, expresses the force exerted on a fluid by the solid dispersed in it per unit volume of fluid and is defined by Eq. (8):

$$\int_{V_f} \vec{R}\, \mathrm{d}V = \int_{A_{fs}} (-P.\vec{n} + \bar{\bar{\tau}}.\vec{n}) \tag{8}$$

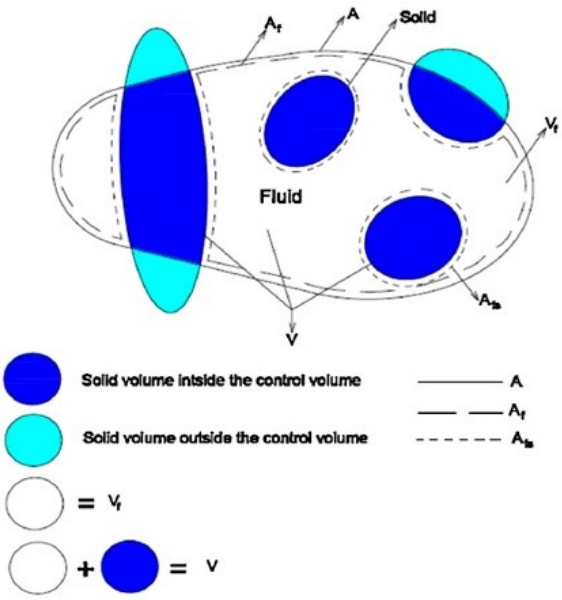

Figure 4: The media consists of a single-phase fluid with a static solid.

In Eq. (7), the term $\frac{DP}{Dt}$ can be written as Eq. (9):

$$\langle \frac{DP}{Dt} \rangle = \frac{\partial \langle P \rangle}{\partial t} + \nabla . \langle \rho \vec{v} \rangle - \langle P \nabla . \vec{v} \rangle \tag{9}$$

Now, a case of Cartesian coordinates is considered that the volume V_T, according to Fig. 5, is equal to $\Delta x \Delta y \Delta z$.

The velocity components in the direction of the x, y, and z axes are $\vec{v}_x$, $\vec{v}_y$, and $\vec{v}_z$, respectively. The integral value of Equation 5 on the volume level V_T will be equal to:

$$\gamma_V \frac{\partial \, {}^i\langle \rho \rangle}{\partial t} + \frac{\Delta_x \left(\gamma_{Ax} \, {}^{i(x)}\{\rho v_x\} \right)}{\Delta x} + \frac{\Delta_y \left(\gamma_{Ay} \, {}^{i(y)}\{\rho v_y\} \right)}{\Delta y} + \frac{\Delta_z \left(\gamma_{Az} \, {}^{i(z)}\{\rho v_z\} \right)}{\Delta z} = 0 \tag{10}$$

where Δ_x is:

$$\Delta_x = (\ )_{x+\frac{x}{2}} - (\ )_{x-\frac{x}{2}} \tag{11}$$

Similarly, these methods have been implemented for the momentum and energy equations. These equations will be solved for different nodes in a given longitudinal direction and time steps. By implementing a combination of triple conservation equations for the entire lattice nodalization, a three-diagonal matrix is formed, which is solved despite the initial and boundary conditions of the pressure equation and the pressure of different points in the z-direction, will be calculated based on the specified algorithm. The solution of the three-diagonal matrix is done in various ways. One of these methods is the Gauss-Jordan method. In this study, the Gauss-Jordan method was used to solve the three-diagonal matrix. The initial and boundary conditions of the reactor core for the coolant and moderator fluid are given in Table 1 and Table 2, respectively. To solve the equations, it is necessary to have the geometry of the reactor core and the fuel assemblies. The geometric specification of the fuel assemblies and the reactor core is given in Table 3.

After calculating the pressure, having the pressure of different points, the axial velocity in all nodes is calculated. Then the temperatures of different nodes are extracted using the energy equation. The remaining thermal-hydraulic variables are calculated using thermodynamic tables and having temperature-pressure. In the study of Bahrevar et al. (Bahrevar et al., 2018) for steady-state, the axial length of the reactor core is divided into 21 nodes. To implement the transient state in this paper, the reactor core is divided into 21 computational nodes. The number of radial nodes intended for the fuel rod is 5. This is shown in Fig. 6 for an SCWR reactor fuel assemblies.

Table 1: Initial and boundary conditions for coolant.

Initial conditions	Inlet	Outlet	Value
Density (kg.m^{-3})	*		743
Enthalpy (kJ.kg^{-1})	*		1331
Boundary conditions	Inlet	Outlet	Value
Pressure (bar)		*	250
Temperature (°C)	*		300
Axial velocity (m.s^{-1})	*		1.2

Table 2: Initial and boundary conditions for moderator.

Initial conditions	Inlet	Outlet	Value
Density (kg.m^{-3})	*		776.9
Enthalpy (kJ.kg^{-1})	*		1230
Boundary conditions	Inlet	Outlet	Value
Pressure (bar)		*	250
Temperature (°C)	*		280
Axial velocity (m.s^{-1})	*		0.2

Table 3: Properties of the SCWR core (Jahanfarnia et al., 2013; Cheng et al., 2003).

Fuel rods per assembly	40
Water boxes per assembly	1
Assemblies per cluster	9
Number of fuel assembly in 1/8 core	99
Fuel rod diameter	8 mm
Fuel pellet outer diameter	6.7 mm
Cladding thickness	0.5 mm
Pitch of fuel rods	9.2 mm
Control rods per cluster	5
Active core height	4200 mm
Total core height	5331 mm
Water box wall thickness	0.3 mm
Water box outer width	26.8 mm
Gap between fuel rods and box wall	1mm
1/2 Gap around the fuel assembly	5 mm
Moderator mass flow fraction	100%
Flow direction in water boxes	Downwards
Flow direction between assembly boxes	Upwards
Flow direction in the radial reflector	Downwards
Control rod absorber material	B_4C
Total thermal power in the core	2075 MW
Total coolant mass flow rate	11.8554 kg.s^{-1}
Moderator mass flow rate	0.07784 kg.s^{-1}
Mass flow rate in the assembly gap	0.15568 kg.s^{-1}

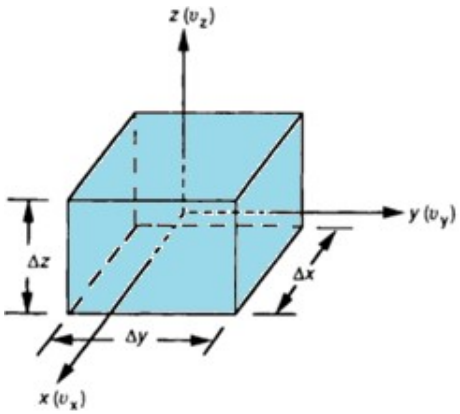

Figure 5: The volume element used in Cartesian coordinates to describe conservation equations.

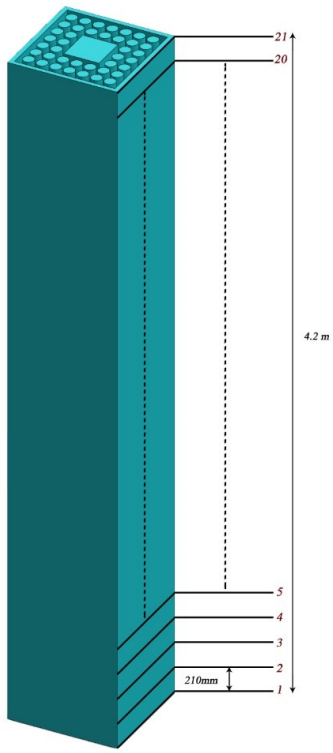

Figure 6: SCWR reactor fuel assembly nodalization (Bahrevar et al., 2018).

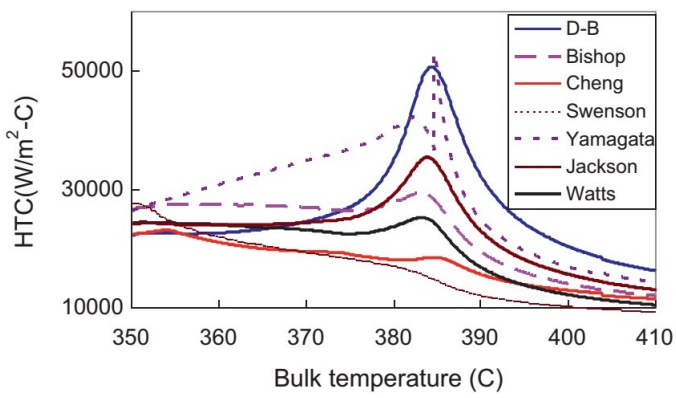

Figure 7: Changes in heat transfer coefficient in terms of fluid bulk temperature with different correlations at MPa 25 (Liu et al., 2013a).

In reactors operating at supercritical pressure, it is generally important to consider the appropriate relationship to calculate the heat transfer coefficient for two reasons. The strange behavior of water characteristics at supercritical pressure, especially near the critical point, doubles the importance of using a heat transfer model to increase computational accuracy. There are various equations for modeling heat transfer through the wall between two fluids. Figure 7 shows how the heat transfer coefficient changes with differential equations in one mode of operation of the SCWR reactor. In designing complex fuel assemblies such as SCWR reactor fuel assemblies, the maximum fuel clad temperature is the important design criterion (Liu et al., 2013a). Analysis by Liu et al. (Liu et al., 2013b) is performed for a laboratory sample of SCWR reactor conditions, by using the Bishop equation, the maximum fuel clad temperature is extracted from the other equations. This issue is due to the existence of the term (ρ_w/ρ_B) in the Bishop equation (Liu et al., 2013b). Therefore, in this study, to calculate the transfer heat transfer coefficient, Bishop Correlation has been used. The Bishop correlation is in the form of Eq. (12):

$$\mathrm{Nu}_B = 0.0069\ \mathrm{Re}_B^{0.90}\,.\,\mathrm{Pr}_B^{0.66}.\left(\frac{\rho_w}{\rho_B}\right).\left(\frac{z+2.4D_h}{z}\right) \quad (12)$$

The second reason for the importance of choosing the best heat transfer coefficient relationship is the high-temperature difference between the coolant and moderator channels in SCWR reactors. This issue is very significant for the thermal-hydraulic analysis of the SCWR reactor core.

Figure 8 depicts a diagram of the thermal-hydraulic calculations flowchart.

3 Modeling results and Discussion

Before describing transient mode modeling and presenting its results, it is necessary to validate the implemented model in steady-state. The model mentioned in this study has been implemented for the reactor geometry of Oka et al. (Oka et al., 2010) study and the modeling results have been compared with the reference in Table 4. According to this Table, it can be seen that for the comparative significant thermal-hydraulic parameters, the calculated error value for the steady-state is acceptable. The implemented model is ready to solve the transient state.

One of the transitions discussed for light water reactors with supercritical pressure is the partial loss of coolant flow accident. In this transition, one of the cooling system pumps first stops at zero moments. Due to the pump inertia, the coolant flow decreases linearly to 50% of its initial value in 5 seconds. After the accident starts in the first second, the reactor shutdown command is issued. After reactor shutdown, the heat in the reactor core is in the form of residual heat. Figure 9 shows the changes in the coolant mass flow rate in the study of Oka et al. (Oka et al., 2010), which is given as input to the transient state solution program.

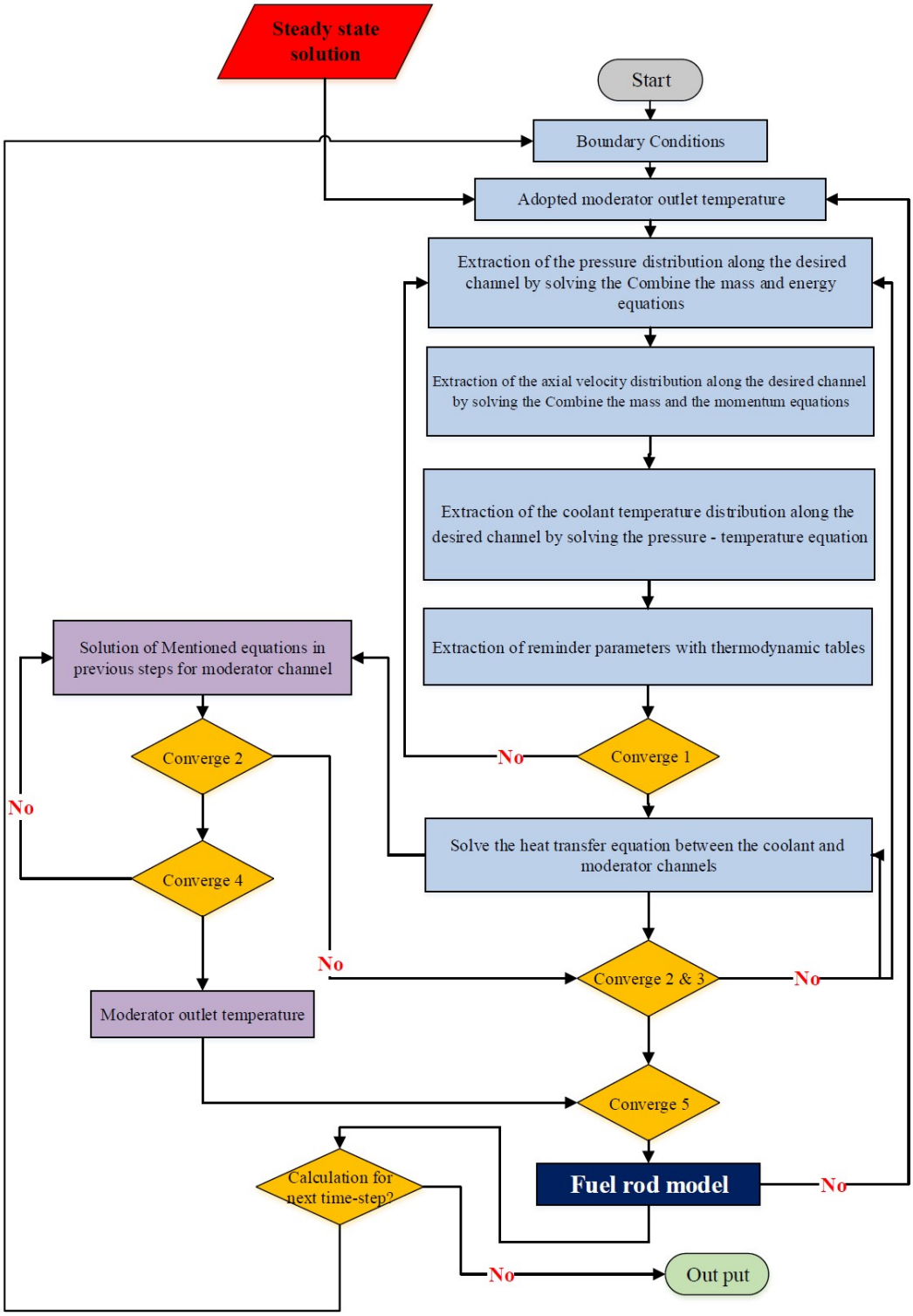

Figure 8: Thermal-hydraulic calculations flowchart.

Table 4: Validation of significant thermal-hydraulic parameters in steady-state calculations.

Parameter	Oka (Oka et al., 2010)	Steady state calculation	Error (%)
Maximum Clad Temperature (°C)	650	632	3.17
Reactor Power (MWt)	2744	2744	0
Outlet coolant temperature (°C)	500	498	0.40
Core pressure (MPa)	25	25.1	0.39

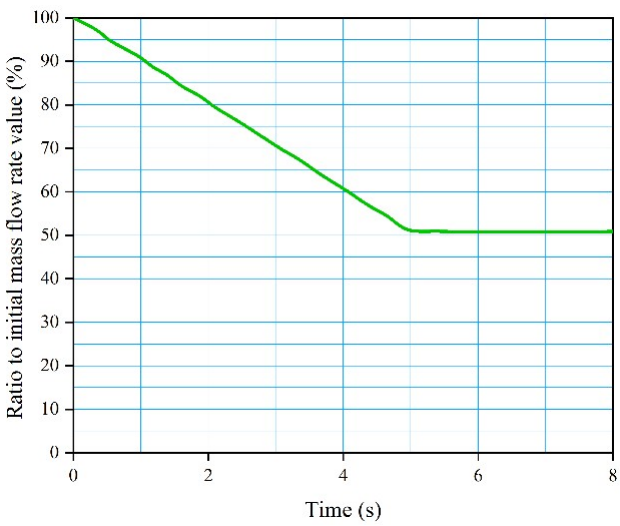

Figure 9: Changes in coolant mass flow rate during the time.

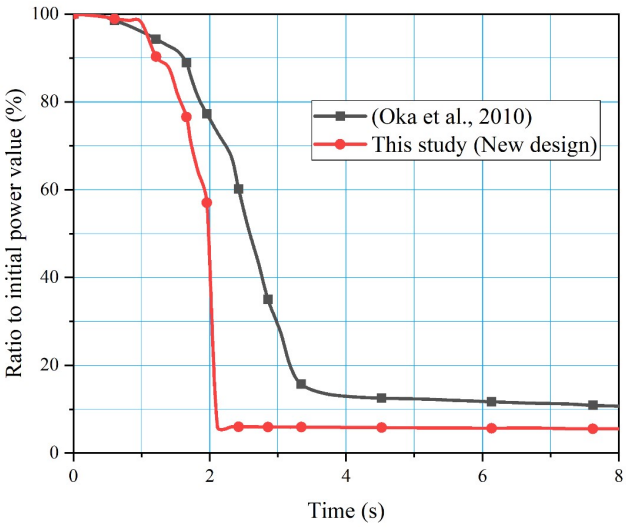

Figure 10: Changes in reactor power output relative to initial value in terms of time.

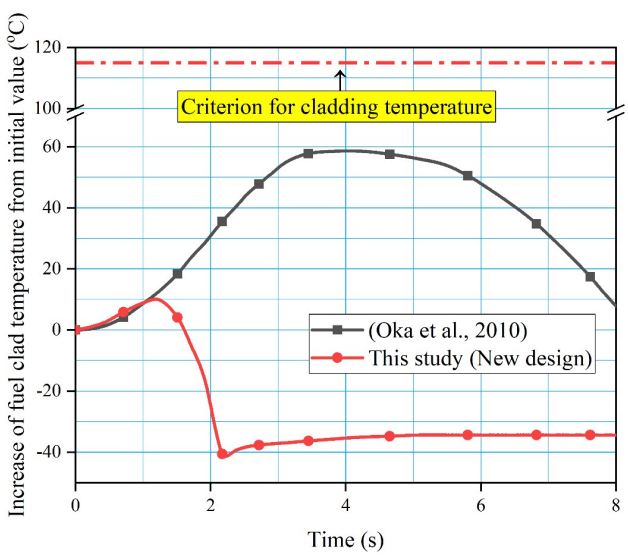

Figure 11: Changes in maximum fuel cladding temperature relative to the initial value in terms of time.

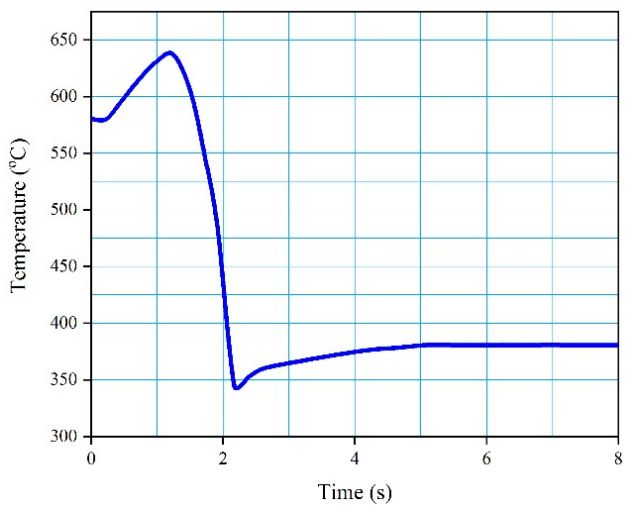

Figure 12: Changes in maximum fuel cladding temperature during the time.

Figure 10 depicts the changes in reactor thermal power in the new design and Oka et al.'s (Oka et al., 2010) design compared to the initial value during the time. According to the figure, it is observed that before the time 1 s when the reactor shutdown command is issued, the behavior of thermal power changes is the same in both designs, and the thermal power of the reactor decreases due to the negative feedback of the coolant density coefficient. After applying the reactor shutdown in the new design, the amount of heat due to heat removal by the moderator channel is higher than the Oka et al. (Oka et al., 2010) design. In the new design, the amount of reactor thermal power at the end of the transition reaches 5.5% of the initial value, while in the compared design, this value has reached 10.78%.

Figure 11 depicts the temperature changes of the hottest fuel rod in the new design and Oka et al. (Oka et al., 2010) design compared to the initial value in terms of time. According to this figure, the fuel cladding temperature changes before the reactor shutdown command, up to 1 s, is the same in both designs, and fuel clad temperature increases due to the reduction of the coolant flow rate. But after the reactor shuts down, it is observed that the fluid flow rate in the water columns is reduced in the design of Oka et al. (Oka et al., 2010) dramatically. This causes the maximum fuel clad temperature to rise in 3.6 s to 60 °C compared to the initial state. Then, with decreasing thermal power, the maximum fuel cladding temperature decreases with time. However, according to Fig. 11, after shutting down the reactor in the new design, due to the heat removal from the reactor core by the moderator channels, fuel cladding temperature drops sharply and at the end of the transition is reduced by about 37% compared to the initial state.

Figure 12 shows the maximum fuel cladding temperature changes during the time. According to the figure, the initial maximum fuel cladding temperature is 557 °C and the maximum fuel cladding temperature at the end of the transition is about 380 °C. Figure 13 depicts the changes in average fuel temperature in terms of time and

the axial length of the reactor core. According to this figure, the maximum average fuel temperature occurred at the moment of shutdown of the reactor at 1016 °C.

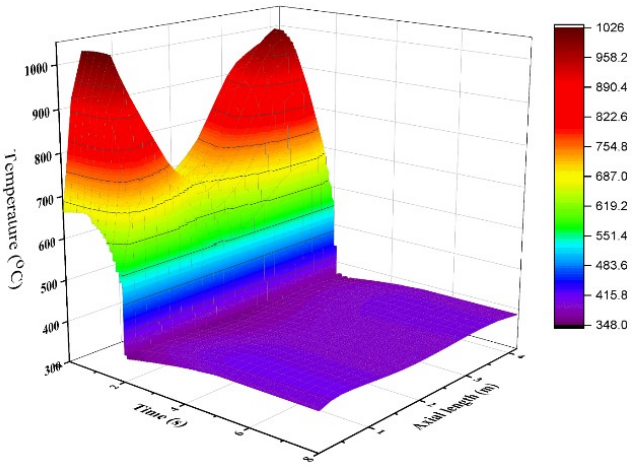

Figure 13: Changes in the average fuel temperature relative to the time and axial length of the reactor core.

4 Conclusion

In today's world, energy production is a significant issue for many countries and their leaders. Meanwhile, the development of nuclear power plants to generate electricity using nuclear power is one of the current issues in the world. Nowadays, new and various strategies to increase thermal efficiency in energy production systems have been proposed and studied, including the use of supercritical fluids in nuclear power plants. With the introduction of supercritical fluids since 1991 to the design of nuclear reactors, a new window has opened in conceptual studies and simulations to achieve safer and cheaper nuclear power plants. Therefore, in this study, the transient state of the reactor with supercritical pressure has been investigated. The SCWR reactor in this study has a new geometric design. In this new design, the coolant and the moderator fluid circuit are separated. In this research, to develop a transition thermal-hydraulic code, the porous media approach has been used. To solve the conservation equations of mass, energy, and momentum for fluid with supercritical pressure, the quasi-implicit method has been used. This method is such that the values of unknown quantities in the new time in the target nodes and their neighbors are calculated simultaneously.

In the loss of partial coolant flow accident, one of the cooling system pumps will first fail. The coolant flow decreases linearly to 50% of its initial value in 5 seconds. In the first second after the accident, the reactor is shut down and this shutdown takes about one second. In the new design, the amount of reactor core thermal power at the end of the transition reaches 5.5% of its initial value, while in the compared design, this value has reached 10.78%. After applying the shutdown command of the reactor in the new design, the amount of heat reduction due to heat removal by the moderator channel is more than the previous designs. In the new design, due to the heat dissipation from the reactor core by the moderator fluid in the water columns, the maximum fuel cladding temperature drops sharply and at the end of the transition is reduced by about 37% compared to the initial state. Since the safety issue is very important in the nuclear power plants discussion, the result of the calculations in this study shows that the new design, in which the coolant and moderator circuits are separated, has created more safety in a chosen transition.

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