STUDIES ON EFFECTS OF WIRE WRAPPED ROD BUNDLES ON THERMAL HYDRAULICS PERFORMANCE

A Thesis Submitted
in Partial Fulfillment of the Requirements
for the Degree of

DOCTOR OF PHILOSOPHY

by

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CANDIDATE'S DECLARATION

I, **Gaurav Kumar** hereby certify that the work which is being presented in the thesis entitled **"Studies on Effects of Wire Wrapped Rod Bundles on Thermal Hydraulics Performance"** in partial fulfilment of the requirements for the award of the degree of **Doctor of Philosophy**, submitted in the Department of Mechanical Engineering, Delhi Technological University is an authentic record of my own work carried out during the period from 01-08-2017 to 30-07-2024 under the supervision of Prof. Raj Kumar Singh

The matter presented in the thesis has not been submitted by me for the award of any other degree of this or any other University/Institution.

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CERTIFICATE BY THE SUPERVISOR

Certified that **Gaurav Kumar (**2K17/PhD/ME/57) has carried out his research work presented in this thesis entitled **"Studies on Effects of Wire Wrapped Rod Bundles on Thermal Hydraulics Performance"** for the award of degree of **Doctor of Philosophy** from Department of Mechanical Engineering, Delhi Technological University, Delhi, under my supervision. The thesis embodies results of original work, and studies are carried out by the student himself and the contents of the thesis do not form the basis for the award of any other degree to the candidate or to anybody else from this or any other University/Institution.

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ABSTRACT

This thesis investigates the thermal hydraulics and optimization of wire wrapped spacers in 2×2 heated rod bundles operating under supercritical circumstances. The main objective was to improve the performance of Supercritical Water Reactors (SCWRs).

There is crucial significance of nuclear power in attaining carbon neutrality and reducing the impact of global warming. Previous studies examine nuclear energy as a viable and dependable source of power, emphasizing the potential of SCWRs as a feasible substitute for traditional reactor designs. The studies emphasize the importance of doing advanced thermal hydraulic analysis and optimization in order to improve reactor performance. It also provides the rationale for the research. Many previous research conducted on SCWRs, including both experimental and numerical investigations. The study explores the heat transport characteristics of supercritical water, evaluates the effectiveness of different spacer designs, and highlights areas where our understanding is lacking. The literature assessment compiles important discoveries from prior research, emphasizing the need for thorough examination of wire spacer arrangements and their influence on thermal hydraulics.

The computer modelling techniques employed in the investigation, includes geometry preparation, mesh production, and the governing equations for the simulations. A user defined function for each thermo-physical property of supercritical water at the operating condition was made and interpreted in the Fluent software. The boundary and operating conditions, the solver and solution methods, and the verification of the CFD model using grid independence tests and validation has been described. A novel optimization framework has been presented, which integrates Taguchi design, Radial Basis Function (RBF) surrogate modelling, and the Non-Dominated Sorting Genetic Algorithm II (NSGA-II).

An analysis of the thermal hydraulic characteristics of 2×2 rod bundles with wire wrapped spacers operating at supercritical conditions was conducted. The analysis investigates fluctuations in axial velocity, velocity vectors, pressure distribution, and temperature distribution. The study contrasts traditional helical wire wrap configurations which is in clockwise direction with novel pattern geometries which wraps wire in clockwise and anti-clockwise direction, emphasizing the advantages of improved coolant mixing and decreased high-temperature areas in the new patterns by 8.5°C. The study examines the differences between straight and helical wire spacers, highlighting the enhanced swirl and turbulence properties of helical wires, despite the presence of a greater pressure drop.

The design optimization of wire spacers with varying wire diameter and varying pitch of wrapping by utilizing the RBF surrogate model and NSGA-II optimization

technique was done. The optimization technique was used to achieve a balance between the heat transfer factor and the pressure drop factor. The optimization framework uses the Taguchi approach for selecting sample points, applies RBF surrogate modelling to create a response surface, and employs NSGA-II for global optimization. The Pareto front analysis offers extremely helpful insights into the ideal ranges for pressure drop factor (ranging from 5.051 to 8.082) and heat transfer factor (ranging from 1.05 to 1.087).

The potential areas for future research of this study may be, extended design parameters, conducting transient flow and temperature studies, implementing real-time optimization, validating through experiments, and integrating with reactor core design. There are the societal implications of the study, emphasizing its contributions to energy security, environmental sustainability, economic growth, technical safety, and contribution to global energy goals.

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GAURAV KUMAR

"This thesis is dedicated to my loving parents and my great forebears, whose unflinching support, love, and sacrifices have been the basis of my road forward.

Your unwavering support and belief in my ability have given me the confidence to follow my aspirations, to my parents, who have been my main source of inspiration and strength. Your relentless work, selfless giving, and pure love have given me the means to cross this milestone.

I am especially appreciative of the wisdom, tenacity, and heritage of my forebears who have moulded the principles I treasure. Their tenacity, clarity, and will to build a better future for next generations have motivated me to keep on and aim high.

This work reflects my great admiration and sincere thanks and pays homage to their continuing spirit and the teachings they imparted in me."

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International Journals

1. Gaurav Kumar, Raj Kumar Singh, Supercritical water flow in heated wire wrapped rod bundle channels: A review, Progress in Nuclear Energy, Volume 158, 2023, 104620, ISSN 0149-1970, https://doi.org/10.1016/j.pnucene.2023.104620. **(SCIE, Published)**
2. Gaurav Kumar, Raj Kumar Singh, Numerical simulation of thermal hydraulics of supercritical pressure water with 2×2 rod assembly wrapped differently with a wire, Progress in Nuclear Energy, Volume 168, 2024, 105029, ISSN 0149-1970, https://doi.org/10.1016/j.pnucene.2023.105029. **(SCIE, Published)**
3. Gaurav Kumar, Raj Kumar Singh, CFD Simulation of 2x2 Rod Bundle at Supercritical Condition for Dual-Objective Design Optimization by RBF Surrogate Model and Genetic Algorithm. **(Communicated)**
4. Gaurav Kumar, Raj Kumar Singh, Role of nuclear power in climate change actions. **(Communicated)**

International Conferences

1. Gaurav Kumar, Raj Kumar Singh; CFD analysis of 2x2 rod bundles at supercritical flow condition. ICTFSD-2022, organized by Department of Mechanical Engineering, Birla Institute of Technology, Mesra, Ranchi, held on 22 - 23 March, 2022.
2. Gaurav Kumar, Raj Kumar Singh; Numerical Analysis of Straight Wire and Helical Wire wrap 2x2 rod Bundle with Supercritical Water. ICMME-2023, jointly organized by Department of Mechanical Engineering, GLA University, Mathura (India) and UFRGS, Porto Alegre (Brazil), held on 13th – 14th October 2023.

List of Symbols

Acronyms

SCWR	Supercritical Water Cooled Reactor
LWR	Light Water Reactor
SCW	Supercritical Water
RPV	Reactor Pressure Vessel
LMFBR	Liquid Metal Fast Breeder Reactor
HTE	Heat Transfer Enhancement
HTD	Heat Transfer Degradation
CHF	Critical Heat Flux
HTC	Heat Transfer Coefficient
ATHAS	Advanced Thermal-Hydraulics Analysis Sub-channel
HPLWR	High Performance Light Water Reactor

Greek Letters

ρ	Density (kg/m^2)
ℇ	Blockage Ratio

Symbols

C_p	Specific Heat (KJ/Kg-K)
Re	Reynold Number
Pr	Prandtl Number
Nu	Nusselt Number
D_h	Hydraulic Diameter (m)
H	Wire Wrap Lead Length (m)
D	Fuel Rod Diameter (m)
D_W	Diameter of Wire spacer
G	Cross-Sectional Average Mass Flux (kg/m^2s)
q	Heat Flux (KW/m^2)
P	Pressure (MPa)
0_T	Heat Transfer Factor
0_P	Pressure Drop Factor

Subscript

b	Bulk
w	Wall
m	Film Temperature ($(t_b + t_w)/2$)
f	Fully-Developed

Chapter 1

Introduction

1.1 Nuclear Power's Contribution to Reaching Carbon Neutrality

Over the course of several decades, nuclear energy has emerged as a significant low-carbon emission energy source, particularly in the field of electricity generation. There is also a significant amount of potential for nuclear power facilities to be utilised in non-electric applications, such as desalination, hydrogen production, and district heating. The flexibility of the energy system can be improved by the use of nuclear power plants, which are a source of low-carbon baseload electricity. These plants are also compatible with renewable energy sources. Advanced nuclear reactors, such as the fourth generation nuclear reactors and small modular reactors (SMRs), have the potential to enhance the economics, sustainability, safety, and flexibility of nuclear energy. As a result, these reactors are excellent prospects for carbon neutrality. Nuclear energy, on the other hand, is confronted with a number of obstacles, including the ageing of plants, a decline in economic competence, concerns over sustainability and safety, and unfavourable public support in certain nations. It is vital to make efforts to find solutions to both the technical and non-technical issues in order to fulfil the significant role of achieving the goal of carbon neutrality for the environment [1].

1.2 Nuclear Power: Present Views on a Potential Solution to Reduce Global Warming

Only high-investment scenarios in nuclear power plants have the potential to contribute more effectively to non-CO_2 emissions. This is because, from a global perspective, averted CO_2 emissions will remain at levels that are comparable to the ones that are currently being produced. The countries that are able to make the most significant contributions to the spread of nuclear energy and, as a result, do not release carbon dioxide into the atmosphere are the main world economies as well as rising economies. On the other hand, it is essential to take into consideration a number of aspects of the problems at hand:

a) The installation time: In many countries, the installation time is long, so planning nuclear plants well in advance is needed to get them up and running on time. For example, the plants in the USA and India need to be planned well in advance. It is vital to pay close attention to decisions that include freezing or even banning the participation of nuclear energy in the energy mix of the country, as was the case in Belgium. This is the case because the economic policy of the country requires careful consideration. Due to the fact that it takes a considerable amount of time for this energy to be put into operation, the local economy may be significantly impacted in the event that this step is reversed for any reason in the future.
b) Popular opinion: People's views on nuclear energy, whether they are for or against it, are often used to put pressure on governments, which then decide whether to include this source of energy in their energy plans and, by extension,

their efforts to control the effects of greenhouse gas emissions or other greenhouse gas emissions.

c) The usable life of nuclear facilities: A great number of nations own a substantial quantity of nuclear reactors, the useful life of which was expected to be thirty years and have the potential to extend to forty years. This indicates that if the nation continues to make use of this energy source, it is imperative that it make preparations to ensure that new nuclear power plants are available to supply the reactors that are being decommissioned when they reach their maximum useful life. The new nuclear reactors, which are being referred to as Generation III and Generation III Plus, are already being planned to operate for up to sixty years, which results in a base generation that has low greenhouse gas emissions and a long life. This is another intriguing part of the examination of this item.

d) Changes in the management of electricity production in the country: France is investing in renewable energy sources and lowering the amount of nuclear energy it uses. This means that the country's government is spread out, and the network's loads change more often, which is similar to what has been happening in Germany since the beginning of the year. Because of this, countries that have raised the amount of nuclear power they use will have to deal with the problems that come with centralized power generation.

e) Economic and environmental expenses connected with nuclear generating are lower than those associated with other fuel sources. Although the construction of nuclear power plants is expensive, once they are operating, their costs are far lower than those of other fuel sources. In order to reduce expenses, particularly with regard to the importation of fuels like natural gas and coal, a number of nations have already begun to evaluate the significance of the incorporation of nuclear energy into their respective matrixes. Nuclear energy generation also results in major environmental benefits, both in terms of the avoided emission of carbon dioxide and the local impact. This is due to the fact that, in contrast to renewable energy generation plants, nuclear power plants require just a small area to operate.

f) Because it is a fundamental method of producing energy, the addition or removal of this source from a nation's overall electricity mix can have a direct influence on the amount of carbon dioxide emissions produced.

g) Numerous nations recognise the nuclear power plant as a significant source of fundamental energy generation and have expressed their desire to implement it. However, a significant number of these nations do not own the domain of nuclear technology. As a consequence of this, these nations are dependent on other nations to carry out and/or manage these projects. As a consequence of this, many people are unable to provide accurate predictions of the true potential for expansion in nuclear power plants.

Nuclear power, which is regarded as a low-carbon technology, is an essential component of the climate change mitigation plan for the purpose of achieving long-term CO_2 reduction targets. As a result, it is feasible to draw the conclusion that nuclear power has a significant potential to assist in the reduction of greenhouse gas emissions; however, the actual values are quite unknown due to the enormous number of variables that are involved in the difficulties of implementation. The use of nuclear energy technology should be incorporated into the development of new and improved techniques for addressing global greenhouse gas emissions [2].

1.3 Exploring Nuclear Energy as a Pathway Towards Future Energy Independence from Carbon

Providing a considerable and increasing amount of electricity, nuclear energy is vital in tackling climate change and lowering greenhouse gas emissions. In addition, nuclear energy provides a source of energy. The ability of this source of energy to rapidly transition away from fossil fuels is something that society needs to be aware of in order to fully appreciate its potential. It is also possible to construct a hybrid energy system by combining small modular reactors (SMRs) with renewable energy sources. This would result in an increase in the efficiency of renewable resources. Nuclear energy is a kind of energy that is safe, sustainable, and carbon-free, and it has immense potential. However, the fear that is associated with nuclear energy is frequently illogical and is founded on political propaganda and "greenwashing." Considering the current state of technology and discoveries, Nuclear energy is the only low-carbon energy source that could be used instead of fossil fuels. This is in addition to renewable energy sources.

More research needs to be done to look into the role of nuclear energy in a local green energy mix, to make plans for nuclear reactors that use less energy, and to learn more about how to properly dispose of and recycle nuclear waste. Moreover, analysing the potential of nuclear energy for small economies would be a significant step in determining the significance of nuclear energy in terms of maintaining energy independence and stability for small countries. Other suggestions for future work include: doing more research to look into its role in the green energy mix on a local level; coming up with plans for nuclear reactors that use less energy; figuring out what role it plays for small economies; learning more about how to get rid of nuclear waste and recycle it; and creating public education campaigns that stress the benefits and safety of nuclear energy to fight political propaganda and "greenwashing." All of these recommendations are intended to further advance the utilisation of nuclear energy as a low-carbon source of energy. To help reach the goals of the Paris Agreement, it is important to keep working on the negative points and adding to the positive conclusions found in the review paper. This will also ensure that advanced economies have a steady supply of electricity for their operations and growth, and it will also help developing countries boost their economies and raise their living standards.

Within the context of the continuing dilemma involving climate change and resources, nuclear energy holds a significant amount of relevance. The use of renewable energy sources will continue to increase in order to meet the objectives of the Paris Agreement. However, it is essential to emphasise that nuclear energy already offers a consistent and dependable source of electricity, which is essential for the operation and expansion of advanced economies, as well as for the ability of developing countries to increase their economic output and improve their living standards [3].

1.4 Examining Nuclear Power's Function in Energy Transitions

Nuclear energy and renewable energy both have the potential to contribute to economic growth while simultaneously lowering CO_2 emissions. The development of these two types of energy sources will bring about a significant reduction in the amount of greenhouse gas emissions that are produced. Renewable energy should be developed, new technologies should be developed, application areas should be expanded, and the market for renewable energy should be accelerated. Governments should encourage the development of renewable energy. As a result of public concerns regarding the radioactive waste and radioactive leaks that have been produced by nuclear energy, certain nations are gradually decreasing their utilisation of nuclear energy. Nuclear power, which is one of the key options for decreasing carbon emissions, has the potential to not only bring economic advantages and reduce carbon emissions, but it can also reduce reliance on energy sources and increase the reliability of energy supply. As a result, the government must to make certain that the nuclear power plants that are already in existence are operating in a secure manner while simultaneously supporting the development of new technologies that will enhance the safety of nuclear energy. The conclusion is that both renewable and nuclear energy are necessary for the continued growth of the economy as well as the preservation of the environment. As a result, it is essential to enhance the proportion of these energy sources that are included in the overall energy mix. Taking into consideration the framework of the Paris Agreement, nations ought to formulate logical strategies that are founded on their current energy mix and economic standing [4].

1.5 Nuclear Power and the IAEA's Position

The Agency has increased its high case prediction for global nuclear capacity for electricity generation to 873 gigawatts (electrical) (GW(e)) in 2050. This is part of its updated view for nuclear capacity everywhere in the world. In order for this to become a reality, it would be necessary to deploy long-term operation (LTO) on a massive scale across the existing fleet, as well as to construct almost 600 GW(e) of new power plants over the next three decades.

Earth's nuclear power output reached 393.8 gigawatts of electricity (GW(e)) at the end of 2022. This was provided by 438 nuclear power reactors running in 32 different countries. Over 7.4 gigawatts electrical (GW(e)) of new capacity was linked to the grid during the course of the year. This additional capacity came from six pressurised water reactors. Additionally, 3.3 (GW(e)) of capacity was retired as a result of the permanent shutdown of five nuclear power reactors. More than a quarter of the world's low-carbon electricity production came from nuclear power, which was responsible for almost 10% of all the energy made in the world. That is because nuclear power was responsible for the generation of 2486.8 terawatt-hours of electricity that did not emit any greenhouse gases. At the end of the year, 59.3 gigawatts electrical (GW(e)) of capacity was being built, which included 58 reactors, eight of which brought the total capacity to 9.1 (GW(e)). Construction on these reactors began in 2022.

We want to learn more about how these systems work and what effect they have on meeting current and future energy needs. To do this, we have started a new coordinated research project (CRP) to study and improve nuclear–renewable hybrid energy systems. The Agency started a CRP on Advancing Thermal-Hydraulic Models and Predictive Tools for Design and Operation of SCWR Prototypes to build a coherent body of knowledge about fluids at supercritical pressures and/or temperatures. This is needed to make prototypes of designs for supercritical water cooled reactors (SCWRs). The CRP also wanted to fill in gaps in technology areas that are important for design choices. The agency made changes to the Thermo-Physical Materials Properties Database (THERPRO), which has details about the different types of materials used in light and heavy water reactors and their improved designs. [5].

1.6 GIF Future Plans and Ongoing Projects

As a cooperative multinational effort to produce the research required to validate the feasibility and performance of fourth-generation nuclear systems and to make them industrially deployable by the year 2030, the Generation IV International Forum (GIF) was established in the year 2001. Research and development on these systems is coordinated by the Global Initiative for the Future (GIF) with thirteen countries (Argentina, Australia, Brazil, Canada, China, France, Japan, Korea, Russia, South Africa, Switzerland, the United Kingdom, and the United States) and Euratom, which represents twenty-seven members of the European Union.

The Generation IV International Forum (GIF) community is currently facing a number of problems and opportunities as a result of the fact that several of the six Generation IV (Gen IV) systems are currently entering the demonstration phase. In general, the research and development (R&D) infrastructure that was utilised to develop advanced reactor technology continues to be relevant for the demonstration and implementation of Gen IV. This includes both the availability of knowledge and facilities. However, the implementation and operating expertise of the company should be used to inform and prioritise the main research and development subjects. It is therefore essential to maintain a close working relationship with the advanced reactor industry, which is engaged in the process of obtaining licences, building advanced reactors, and operating them, in order to guarantee that Gen IV collaborations will continue to be relevant.

Furthermore, demonstration systems frequently rely on design choices that are low-risk and characterised by a high level of technical maturity, even when new technological solutions provide significant increases in performance standards. Therefore, in order to facilitate future developments, it is necessary to maintain a robust research and development infrastructure that allows for the rapid maturation of promising design elements. For the purpose of enabling widespread deployment (after the demonstration phase) of high-performance, robust Gen IV advanced reactors, this provision for future enhancements is absolutely necessary.

During the year 2022, the activities of the GIF were centred on enhancing engagement with the business sector. Particular prospects for collaboration have been identified and their implementation has begun. Moreover, opportunities have been pursued in order to highlight the joint research accomplishments of the GIF and the impact that these accomplishments have had on the aims of sustainable development for advanced reactors.

Research and development efforts for six different reactor technologies have been selected by the Global Initiative for Nuclear Energy (GIF). These reactor technologies are as follows: the gas-cooled fast reactor (GFR), the lead-cooled fast reactor (LFR), the molten salt reactor (MSR), the sodium-cooled fast reactor (SFR), the supercritical water-cooled reactor (SCWR), and the very high-temperature reactor [6].

1.7 Working of a Nuclear Reactor

Light Water Reactors (LWRs) are the predominant nuclear reactors employed globally for the purpose of generating energy. Nuclear power plants function based on the principle of nuclear fission, which involves the splitting of the nucleus of a heavy atom like uranium-235 into two smaller nuclei, resulting in the release of a substantial quantity of energy. An LWR operates through a series of distinct stages:

1. **Fuel Rods and Control Rods:**

Fuel rods in Light Water Reactors (LWRs) consist of uranium dioxide (UO_2) pellets enclosed within metal tubes. The rods are consolidated into bundles and inserted into the reactor core.

Control rods consist of elements such as boron or cadmium that have the ability to absorb neutrons. They are introduced into the reactor core in order to manage the fission reaction by absorbing neutrons.

2. **Nuclear Fission:**

Within the nucleus of the reactor, neutrons collide with uranium-235 atoms, inducing their division and subsequent emission of thermal energy.

The fission process also emits additional neutrons, which subsequently proceed to divide other uranium atoms, so sustaining a continuous chain reaction.

3. **Heat Generation:**

Water functions as both the coolant and moderator, absorbing the heat produced by fission.

Coolant: Water functions by absorbing the heat generated in the reactor core and subsequently transporting it away.

Moderator: Water acts as a moderator by reducing the speed of neutrons, hence enhancing their ability to maintain the fission chain reaction.

4. Steam Production:

In a Pressurized Water Reactor (PWR), the water is maintained at elevated pressure to inhibit boiling. The hot water transfers its thermal energy to a secondary water circuit in a steam generator, resulting in the production of steam.

In a Boiling Water Reactor (BWR), the reactor core directly induces the boiling of water, resulting in the generation of steam which powers the turbine.

5. Electricity Generation:

The steam propels a turbine linked to a generator, transforming thermal energy into mechanical energy and then into electrical energy.

6. Condensation and Recirculation:

The steam is then condensed back into water in a condenser and recirculated back into the reactor or steam generator to continue the process.

1.8 Exploring the Potential of Supercritical Water Reactors as a Novel Alternative

With the advantages of a higher thermal efficiency of the supercritical water-cooled reactor (SCWR) and the simplification of its system, as shown in Figure

1. 1, this is the most attractive option among the six Generation IV reactors that the GIF has available [7–9].

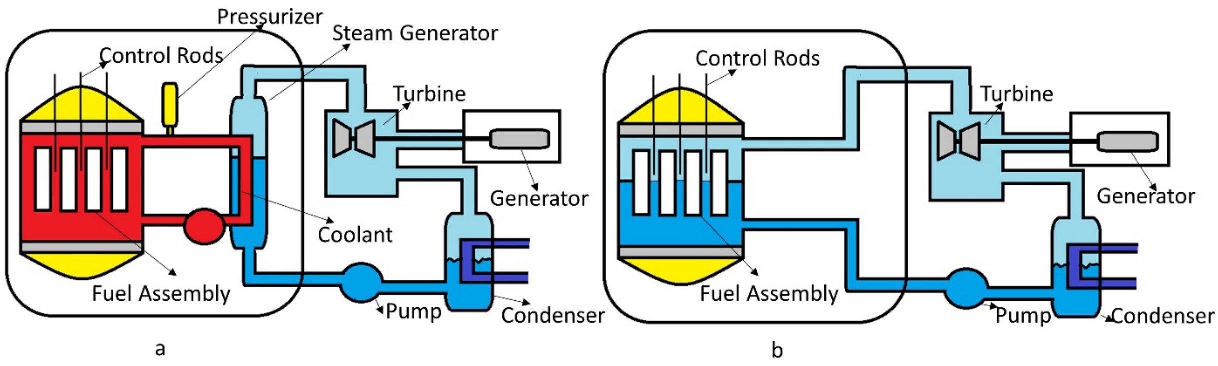

Figure 1. 1 Schematic Diagram of Reactors (a) PHWR (b) SCWR

SCWR designs aim to directly reduce the surface temperature of the cladding in order to ensure the safety of the fuel rod assembly. The utilisation of grid spacers is an important way that may be utilised to reduce the temperature of the cladding [10]. Two of the most common forms of spacers that are utilised in SCWRs are wire spacers and spacer grids. [11]. This next-generation technology is a modern light water reactor (LWR), and it is known as the SCWR [12]. Technologies that are already in use at low-water reactors and fossil fuel plants were incorporated into its design in order to guarantee that it would meet the commercial requirements for increased safety, thermal efficiency, and cost-effectiveness. It is anticipated that the SCWR would function under supercritical pressure settings, which will make heat transport in the reactor core substantially more challenging or significantly different than it is in traditional LWRs with the same characteristics. Over the course of the past ten years, the focus of research has been on the fundamental aspects of heat transfer in tubes and normal SCWR sub-channels that include supercritical fluids [13].

When it comes to the SCWR idea, the most important characteristic is that the nominal pressure of the coolant is typically 25 MPa. After reaching the critical water pressure value of 22.064 MPa, the pressure density reaches its maximum. The temperature of the water, which is 500 degrees Celsius, is higher than the critical temperature, which is 373.95 °C, at the outflow of the pressure vessel connected to the reactor. In the process of approaching the reactor pressure vessel (RPV), the coolant, which is at a sub-critical temperature but a supercritical pressure fluid, goes through the pseudo-critical transition, which is determined by the pressure (384.95 °C at 25 MPa) at the pseudo-critical temperature. After this transition, the coolant exits the reactor pressure vessel as a supercritical fluid, which is water in the case of the supercritical water reactor (SCWR) [14]. The SCWR is a water-cooled reactor of the once-through type that operates at a pressure that is higher than the critical water pressure (22.1 MPa). It supplies the turbine system with supercritical pressure steam that is at a high temperature throughout the process [15]. The design of the plant is anticipated to attain a higher thermal efficiency and a more straightforward system in comparison to the nuclear power plants that are currently in operation [16]. It is possible for us to increase our thermal efficiency from 33 to 35% to approximately 45% if we make use of supercritical water. Consequently, a great number of theoretical and experimental investigations pertaining to supercritical water have been conducted and published [17].

SCWR designs are targeted at producing an operation that is safe, dependable, and expensive in terms of power consumption. Nevertheless, because the reactor coolant does not experience the transfer of liquid vapour beyond the critical pressure, the system of a power plant that uses SCWR will become smaller by removing the recirculation pump, steam generator, and separation of the steam-water system. This will further reduce the amount of money that is spent on capital expenditures [18].

Not having a complete understanding of the state of the supercritical water (SCW) heat transfer phenomenon and the production of materials for fuel and core geometries that are sufficiently resistant to corrosion in order to withstand the nominal supercritical water conditions are two of the most significant technical challenges that must be overcome when designing a supercritical water reactor (SCWR). In order to provide accurate forecasting of the pressure loss, flow field, temperature distribution, and coefficient of heat transfer in the fuel assembly, the thermal-hydraulic system of the reactor core must be in place [19]. Additionally, fuel rods, spacers, flow channels, and other components typically come together to create fuel assemblies within the core of the nuclear reactor, which is the location where atomic fission heat generation and coolant flow conveyance take place. Providing support to the fuel rods and committing to the mixing of sub-channels are the two primary functions that the grid spacer serves within the core of the reactor. Several studies have demonstrated that the spacer grid is capable of facilitating significant alterations in flow topologies and heat transfer properties [20,21].

The wire-wrap spacers, which are depicted in Figure 1. 2, not only maintain the geometric structure of the rod bundle, but they also have an effect on the transmission of heat and make mixing easier inside the flow field [22–24].

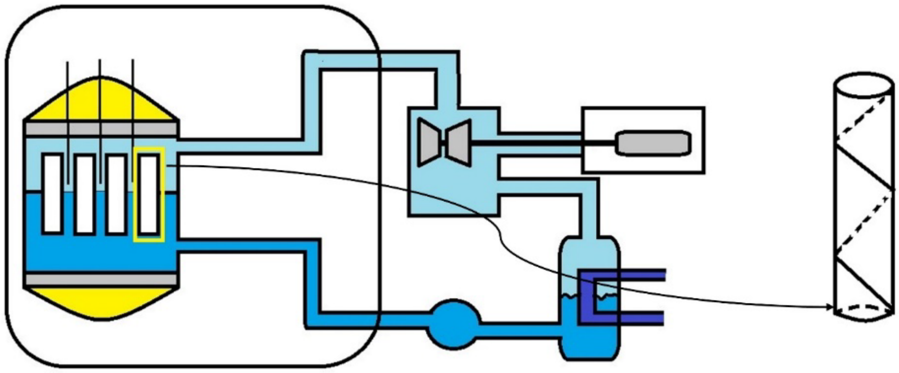

Figure 1. 2 Schematic of Fuel Rod with a Wire Spacer

The increased mixing is essential because it reduces the temperature of hot spots, which are hot areas that are damaging to the cladding of the fuel. As a result, it increases the margin of safety. It is essential to have a solid understanding of the spacer's impact on fluid particles and the mechanism by which heat is transferred in order to build and optimise the fuel assembly [25]. It is possible to increase the uniformity of the coolant flow distribution by decreasing the diameter of the wire that is used for the peripheral rods. This, in turn, results in a temperature distribution that is more consistent throughout the entire system. A wire with a smaller diameter has a friction factor that is somewhat lower than that of a standard wire-wrapped bundle, and its heat transfer coefficient is much higher than that of the traditional bundle [26]. The increase in the transverse velocity is directly proportional to the number of turns that the wires must undergo. Because each turn of the wire increases the resistance of the wire, the values of pressure drop throughout the heated length are larger the more twists of wires there are. This is because the resistance of the wire increases with each twist. The decrease in the average wall temperature of rods that occurs as a consequence of an increase in the number of wire turns also has the effect of homogenising the temperature distributions of the sub-channels that surround rods that are identical to one another [14]. Taking into consideration the heat transfer coefficient, which is depicted in Figure 1. 3, it is feasible to assert that the use of wrapped wire spacers results in the greatest increase in heat transfer in the case of corner type sub-channels (SC-1) when the pitch is increased. When these sub-channels were in their most basic geometric form, they experienced the highest levels of overheating on the initial stage. It is possible to produce a moderate rise in the wall type of sub-channels (SC-2); however, in the central type of sub-channel (SC-5), which had the best cooling in bare geometry, the wires do not in any way boost the heat transfer. This is because the wires are not able to enhance the heat transfer [14].

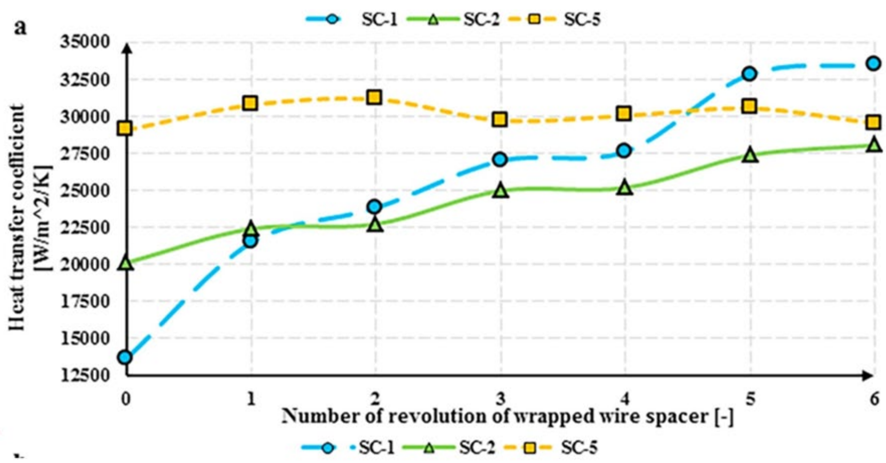

Figure 1. 3 Average heat transfer coefficients versus the number of turns in the wrapped wire spacers [14]

The geometries of the wire wraps aid to establish a robust secondary flow arrangement due to the helically structured wires that are wrapped on the surface of the fuel rod. This is in contrast to spacer grids, which produce local turbulence at the site of the spacer and in the immediate downstream region [25,27–29]. Wire wrap spacers, which are an alternative to grid spacers, were utilised widely in high conversion LMFBR (Liquid Metal Fast Breeder Reactor) and tight-lattice cores. This was mostly due to the advantages that wire wrap spacers offered in terms of pressure drop and enhanced mixing capabilities respectively [30].

As a consequence of this, the amount of power required to pump the coolant is decreased, and the temperature distribution of the coolant is made more uniform throughout the bundle. This results in a reduction in the temperature at which the cladding reaches its greatest point. The capacity of wire wrap spacers to improve heat flow within the SCWR assembly as a result of their ability to reduce heat transfer losses is yet another possible advantage of using wire wrap spacers [31,32].

It is presumed that SCWRs are operated at a critical pressure of 25 MPa, with temperatures of 280 °C and 500 °C or higher at the intake and exit, respectively. This assumption is made from the perspective of the professional designer. Given that supercritical water is a fine coolant, the heat flux of the fuel rod surface is greater than 1500 kW/m^2, while the average mass flux is approximately 1600 kg/m^{2}s. This is done in order to maintain the peak temperature of the cladding at a level that is lower than 650 °C [33]. When compared to conventional reactors, SCWRs have a lower flow rate of the coolant because the supercritical water in the pseudo-critical region has a far higher heat capacity than the regular reactors. The burning phenomenon does take place in SCWRs, despite the fact that there is no phase transition that takes place that takes place [34]. Rapid fluctuations can be observed in the properties of supercritical water that are located close to the pseudo-critical area [35]. The investigation of the features of heat transfer became increasingly important as a result of the fast variations in the properties concerned. Because of the differences in its properties, a great number of researchers have investigated the one-of-a-kind heat transfer that is characteristic of supercritical water in order to gain a deeper knowledge of it [36–42]. In recent times, a number of universities in China, including Shanghai Jiao Tong University and Xi'an Jiao Tong University, have conducted research on the heat transfer characteristics of a 2 × 2 rod bundle that contains supercritical water. The objective of this study is to acquire a more comprehensive understanding of fuel rods that utilise supercritical water as a coolant [43–45].

The enhancement of heat transfer is a crucial phenomenon that ought to be taken into consideration throughout the design process of the reactor core. By ensuring that there is a proper transmission of heat from the fuel assembly to the coolant, it is possible to prevent the fuel rod from melting out and so reduce the likelihood of an accident occurring. Spacers of varying forms are utilised in order to significantly improve the rate at which heat is transferred from the fuel assembly. The aftermath of spacers has been examined by a large number of researchers in order to determine the heat transfer phenomenon that occurs from fuel rods to water [22,29,46,47].

In spite of this, a great number of studies have been conducted with grid spacers, and a small amount of research has been conducted with wire-wrapped spacers. In contrast to the grid spacers, the wrapped wire site does not experience the localised turbulence that is characteristic of the grid spacers. Instead, secondary flow patterns are created across the helical structures [9]. The heat transfer coefficient of a 2 × 2 wrapped wire bundle was 8.4% greater than that of a bare rod assembly. This was due to the swirl flow pattern of helically wrapped wire, which was responsible for the higher heat transfer [34]. When the helical wire is installed around the 2×2 rod bundle, there is a contraction in the circumferential wall temperatures of the rod. This contraction is particularly noticeable at bulk fluid enthalpies that are subcritical and supercritical [44].

A great number of scholars have utilised these experimental data for the aim of statistical validation using numerical simulation. Additionally, Computational Fluid Dynamics (CFD) has the potential to investigate complex thermal systems on high-performance computers, which can result in a reduction in the overall cost of the project. This can be accomplished without the need to conduct experiments in the real world. It is possible for us to get a general concept of how well the system is functioning based on the data that came out of the CFD [25,48].

1.9 Motivation of the Research Work

In order to fulfil the growing demand for energy and tackle the urgent issues posed by climate change, the global energy landscape is progressively moving towards low-carbon and sustainable solutions. Large-scale, low-emission power generation is still made possible by nuclear energy, which is why it is still essential to this transition. Supercritical Water Reactors (SCWRs), one of the several cutting-edge nuclear reactor technologies in development, are notable for their potential to greatly increase thermal efficiency and economic performance in comparison to conventional reactors.

By operating below the critical point, SCWRs prevent the phase transition of water from liquid to vapour by keeping it in a supercritical state. Higher thermal efficiency, less coolant flow requirements, and simpler reactor systems are the outcomes of this. These advantages do, however, come with a cost in terms of technical difficulties, especially in the field of thermal hydraulics, where it is necessary to fully comprehend and control the behaviour of supercritical fluids at high pressure and temperature in order to maintain reactor performance and safety.

The driving force for this PhD study is the urgent need to deepen our understanding of the unique thermal-hydraulic phenomena associated with SCWRs, as these phenomena are essential to the successful development, operation, and commercialization of this cutting-edge reactor technology. Three main areas of interest are the fluid dynamics in the reactor core, the stability of the flow under different operating circumstances, and the heat transfer properties of supercritical water. These elements are essential to making sure SCWRs can function effectively and safely while using the full potential of the supercritical water features.

Although supercritical water reactors (SCWRs) hold great potential, there is a significant deficiency in the current body of literature and experimental evidence concerning the exact behaviour of supercritical water in the intricate settings of nuclear reactors. The thermal-hydraulic performance under supercritical conditions is frequently not well predicted by conventional correlations and models designed for subcritical conditions. This gap requires a thorough analysis using sophisticated numerical simulations as well as experimental studies in order to create and test new models that can accurately anticipate the behaviour of supercritical water in SCWR systems.

Furthermore, problems with material performance, corrosion resistance, and the general design of reactor components arise from the special thermal characteristics of supercritical water. Comprehending these obstacles is essential for creating resilient materials and safety measures that can tolerate the severe circumstances of SCWR functioning.

The present study aims to address these issues by delving further into the examination of thermal-hydraulic processes in SCWRs. This project seeks to develop comprehensive models that help improve the design and safety of SCWRs by combining advanced computational modelling with practical studies. The results of this study will support the larger objective of attaining a sustainable, safe, and efficient energy future in addition to advancing SCWR technology.

In conclusion, the urgent need to get past the technical obstacles in SCWR thermal hydraulics is what drives this PhD study. By doing so, we can fully utilise this next-generation nuclear technology and support international efforts to mitigate climate change and develop sustainable energy sources.

1.10 Organization of Thesis

This thesis is painstakingly structured into six thorough chapters, each of which explores different parts of the research on enhancing the design of wire spacers for 2×2 rod bundles in supercritical settings. The framework of the study guarantees a coherent progression of information, leading the reader from the foundational principles and literature review to the intricate techniques, outcomes, and wider implications of the research.

- Chapter 1: Introduction

The opening chapter establishes the context for the thesis by delineating the primary motives, aims, and importance of the research. The article offers a comprehensive summary of the background in which the study is set, highlighting the crucial importance of nuclear power in attaining carbon neutrality and the possibility of supercritical water reactors (SCWRs) as a feasible technological advancement.

- Chapter 2: Literature Survey

The literature survey offers a thorough examination of current research and studies pertaining to supercritical water reactors and wire spacers. It identifies deficiencies in the existing knowledge and establishes the basis for the research technique.

- Chapter 3: Mathematical Model and Research Methodology

This chapter provides an overview of the mathematical models and research approaches used in the study. The text offers comprehensive explanations of the computational methods and optimization frameworks employed to accomplish the study goals.

- Chapter 4: Thermal Hydraulics of 2×2 Rod Bundles with A Wire Spacer at Supercritical Condition

This chapter specifically examines the thermal-hydraulics of 2×2 rod bundles with wire spacers when subjected to supercritical conditions. It provides a thorough study and comparison investigations.

- Chapter 5: Design Optimization of Wire Spacer for 2×2 Rod Bundles

This chapter discusses the process of optimizing the design of wire spacers for 2×2 rod bundles, with the main objective of enhancing thermal-hydraulic performance in supercritical circumstances.

- Chapter 6: Conclusions, Future Scope and Social Impact

The concluding section provides a concise overview of the research results, explores potential future directions, and emphasizes the societal implications of the study.

Chapter 2

Literature Survey

2.1 Introduction

This dynamic evolution is being pushed by the demand for safer, more efficient, and environmentally sustainable power generation, which is driving the field of nuclear energy through its current state of flux. Supercritical water reactors, also known as SCWRs, have emerged as a viable technology among the different advanced reactor designs. SCWRs have the potential to improve the economic and operational efficiency of nuclear power plants, which is a significant advantage. Supercritical water reactors (SCWRs) are able to operate at pressures and temperatures that are supercritical, which eliminates the phase change of water and enables a thermodynamic cycle that is more efficient than that of conventional nuclear reactors. Because of this property, SCWRs are positioning themselves as an appealing alternative for the generation of nuclear electricity in the future.

The purpose of this literature review is to provide a comprehensive overview of the present state of research in the topic of SCWRs, with a particular emphasis on both experimental and numerical analysis. Regarding the complicated thermal-hydraulic behaviours, material performance, and safety considerations that are involved with SCWRs, the survey will investigate the progress that has been made in understanding these behaviour patterns.

Because they give empirical data on the behaviour of materials and fluids under supercritical conditions, experimental studies in SCWR research are essential for verifying theoretical models and numerical simulations. This is because they provide real-world examples of how these phenomena occur. The scope of these studies encompasses a wide variety of subjects, such as the heat transfer properties of supercritical water, the corrosion and structural integrity of materials that are subjected to supercritical environments, and the performance of reactor components under conditions of high temperature and high pressure.

The use of numerical analysis, on the other hand, is of critical importance when it comes to modelling the behaviour of SCWR systems and estimating their performance under a variety of different operating circumstances. When it comes to the research of thermal-hydraulic processes, reactor core design, and safety analysis of SCWRs, computational approaches such as computational fluid dynamics (CFD) and multi-physics simulations are utilised to a significant degree. It is difficult to duplicate the complex interactions that occur within the reactor, such as those involving fluid flow, heat transfer, and structural behaviour, in experimental settings; however, these technologies make it possible to conduct in-depth investigations of these interactions.

Developing a stable foundation for the advancement of SCWR design and optimisation is made possible by the convergence of experimental and numerical methodologies. During this literature review, the synergy that exists between these techniques, focusing on the most important findings, innovations, and problems that are associated with SCWR research has been investigated. In addition to identifying areas that require additional study and development, the purpose of this undertaking is to provide insights into the possibilities and limitations of SCWR technology as it exists at the present time.

2.2 Experimental Investigations

An increase in heat transfer can take place in the pseudo-critical region when there is a high flux of mass and a relatively low flux of heat. On the other hand, deterioration was observed for the annular geometries, which are square in shape and have a spacer that is wrapped in helical wire, when there is a low flux of mass and a relatively high flux of heat [49]. It can be seen that these results are in agreement with the preceding observations that were summarised and presented in Table 2. 1 for the circular geometry [50,51]. The authors employ the terms degradation and enhancement to characterise advances in heat transfer in reference to empirical correlations, as mentioned in the literature. This is due to the fact that there is no absolute and exclusive sense to the influence of degradation and enhancement.

When the supercritical pressure is low, the flow has a more prominent transfer of heat enhancement. On the other hand, when the supercritical pressure is high, the beginning of the deterioration of the transfer of heat is delayed, and the flow displays less obvious deterioration. This is because the pseudo-critical area has a high specific heat. At supercritical pressure and under general heat transfer conditions, the helical wire-wrapped spacer does not significantly improve the heat transfer. However, it does commit to the enhanced transfer of heat in the pseudo-critical region and a downstream improvement in the onset of degradation when there is relatively high heat flux and low mass flux. This is because the spacer is wrapped in helical wire [49].

The helical spacer has a heightened impact on the local heat transfer, particularly in the vicinity of the pseudocritical temperature because of its spatial arrangement. Generally speaking, the flow conditions have a strong influence on the enhancement in heat transfer that occurs with spacers and the distance that is concerned with them. A decrease in the transport of heat takes place as a consequence of the significant changes in the thermo-physical properties of water that are located in close proximity to the pseudo-critical temperature. On the other hand, because of the spinning and distracting action of the spiral spacer, there is no decrease in the transfer of heat at the location where the spacer is put. In the context of forecasting supercritical water transfer of heat in annular channels, the RNG k-ε turbulence model is generally relevant. This is indicated by the excellent compliance of numerical data with the experimental dataset, with the exception of a relatively higher variation that appears in the vicinity of the pseudo-critical temperature. There is no doubt that the turbulent kinetic energy of the flow with a spacer is higher than that of the flow without a spacer. This is the primary explanation for the greater influence that the spacer has on heat transfer. An additional point to consider is that the kinetic energy of turbulence is growing as the mass flux increases. When there is a spacer in close proximity to the wall, the velocity of the fluid moving downstream is substantially higher than when there is no spacer present. This is yet another essential component that contributes to the enhancement of heat transfer. One of the factors that contributes to the lower temperature of the boundary layer is the combination of the higher velocity effect and the higher strength of the turbulence. The buoyancy effect is undermined by a minor gradient of radial density, and the heat conduction increases at the near-wall area with higher thermal conductivity. Both of these factors will contribute to an improvement in the transmission of heat [18].

The wall temperature around the heated rod was examined for its circumferential variation, which was taken into consideration. It was noted that the temperature of the wall was at its highest in the corner area, while the temperature of the wall was at its lowest in the centre sub-channel region according to the observations. If the temperature of the fluid is raised, the gradient of the wall temperature will decrease until the pseudo-critical point, but it will rise beyond the pseudo-critical point. On the other hand, if the flux of heat or the flux of mass is increased, the gradient of the wall temperature will grow. On the other hand, the effect of a heat flux on the heat transfer coefficient is significant in the region of pseudo-critical enthalpy. The transport of heat is significantly influenced by the mass flux that occurs. An increase in the flow of mass has resulted in a rise in the amount of heat that is being transferred. It can be noticed that the pressure that exists between the parameters of the system has a relatively minor impact on the transmission of heat, and this phenomenon is primarily observed in the pseudo-critical region [43].

The features of heat transfer in supercritical water are greatly influenced by a flux of heat, particularly in the region of pseudo-critical enthalpy. There is a significant increase in wall temperatures when there is an increase in heat flux, and the coefficients of heat transfer that correspond to this increase obviously drop. There was no evidence of a rising wall temperature in any of the two annular gap channels when the SCWR was operating under its envisaged working condition. There are extraordinary effects that mass flux has on the temperature of the wall and the heat transfer coefficient across the full area of enthalpy. A significant improvement in heat transfer can be attributed to the expansion of mass flux. Positive effects on heat transport can be observed at the local stage when spiral spacers are utilised [52].

The investigation focused on the non-uniform distributions of temperature in the wall surrounding the circle of the heated tube. The wall temperature between the heated rod and the ceramic flow tube reached its highest point at the narrow gap region, while the temperatures between the heated rods at the middle sub-channel were at their lowest. In general, this was the situation that was encountered. The circumferential gradient of the wall temperature is greater in locations with low enthalpy and high enthalpy, although the proximity of the pseudo-critical enthalpy zone is much reduced in these areas. Because of the additional heating of the wire geometry that results from the heating of the joule, the wall temperatures at the wire-wrapped sites are significantly higher than those at the other sites. With the decrease in the flow of mass and heat, the height of the coefficient of heat transfer shifts to lower enthalpies of bulk-fluid than the pseudo-critical value. This occurs because the pseudo-critical value is increased. The usual models of bulk-fluid enthalpy, heat flux, mass flux, and pressure are obeyed by the average temperature of the wall and the mean coefficient of heat transfer when the wire-wrapped spacer is placed over the diameter of the heated conduit. In accordance with the findings presented in Figure 2. 1, the temperature of the circumferential wall dropped as a consequence of the installation of wire-wrapped geometric spacers. Additionally, the temperature gradients of the circumferential wall were minimised, in contrast to those of the rod bundle that did not contain any wires for support. It is likewise the case that the average coefficients of heat transfer for the helical wire-wrapped geometries are larger than those for the bare-rod bundles [44].

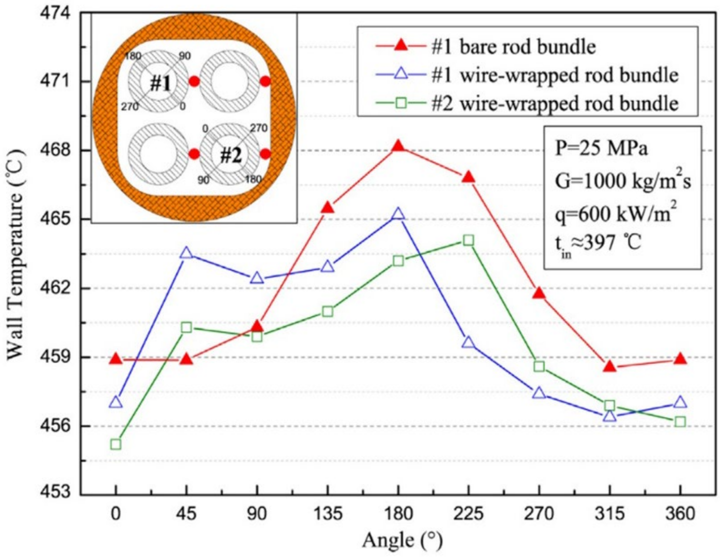

Figure 2. 1 Circumferential wall temperature variations around the heated rod with and without wire wrapped spacer with fluid as water [44]

When the system is functioning at the general heat transfer level, the effects of parameters on heat transfer in the downward flow, such as strain, mass flux, and heat flux, are identical to those of the upward flow. Because of the low heat flux and the high mass flux, the heat transfer coefficient (HTC) is able to be increased. As a result of the combination of high heat flux and low mass flux, the upward flow in the heat transfer will become less effective. The pressure has a marginal impact on HTC. The downward flow is where the heat transfer is eliminated, while the upward flow is where the degradation of heat transfer occurs. However, under typical operating conditions, the differential in heat transfer between the upward flow and the downward flow is not significant enough to call attention to. As a result of the swirl flow that occurs as a result of wrapped wire, the coefficient of heat transfers in rod bundles 2 ×

2 that contain wire wrapped is 8.4% higher than the coefficient of heat transfers in rod bundles that do not contain wire [34].

It is not possible to use the known correlations for forecasting the augmentation of heat transfer caused by the spacer effect in supercritical water since a significant underestimation was seen at the exit of the spacer grid. On the basis of the databank of heat transport under supercritical conditions, a new correlation that has a structure that is comparable to that of sub-critical correlations was developed. Newly discovered correlations with the spacer effect demonstrate a level of accuracy that is satisfactory and have the ability to forecast the enhancement of heat transfer that is caused by the spacer grid effect [53].

Table 2. 1 Experimental studies on heat transfer of SCW in rod-bundles

Year	Authors	Flow Geometries	Spacers	Major Observations
2021	Zhao et al. [53]	2 × 2 Rod Bundles	Grid	• Existing correlations for predicting spacer effect heat transfer enhancement cannot be applied in supercritical water due to substantial underestimation at spacer grid departure. The supercritical heat transport databank suggested a new association with a sub-critical structure.
2020	S. Chen et al. [47]	3 Rod Bundles	Grid	• Significant augmentation of heat transport is found in the region after the spacer grids.
2020	J. Zang et al. [8]	2 × 2 Rod Bundles	None	• The friction coefficient of supercritical water has a distinct "V" shaped profile

				in relation to the Reynolds number, which is different from the flow of subcritical fluids. • The augmentation of heat flux and mass flux results in a reduction of the friction coefficient in the examined experimental data.
2018	Z.X. Hu et al. [34]	2 × 2 Rod Bundles	Wire	• Heat transfer deterioration appears in the upward flow and disappears in the downward flow. • Wrapped wire cannot avoid the heat transfer deterioration but flowing downward. • However, under the normal working condition, the difference of heat transfer in the upward and downward flow is ignorable.
2017	H. Li et al. [54]	2 × 2 Rod Bundles	Wire	• Four distinct transient heat transfer processes occur during depressurization: boiling crisis in the 2 × 2 bundle, boiling crisis in the top part of the bundle, more intense boiling crisis with wall temperature fluctuation, and absence of boiling crisis in the bundle. • The occurrence of a boiling crisis is probable in SCWR (Supercritical Water-cooled Reactor) during the process of depressurization.

				• This is because the operating condition of SCWR is encompassed within the test condition. Therefore, it is necessary to maintain a high enough mass flux and rapidly reduce the heat flux to a significantly lower level in order to prevent a boiling crisis during the transition from supercritical to sub-critical condition.
2017	Z. Shen et al. [55]	Circular Tube	None	• The specific heat ratio is calculated to account for the impact of property fluctuations, buoyancy characteristics, and acceleration parameters. • The dependency of HTC on these dimensionless parameters is generally weak, with most of them showing very limited independence. • However, the buoyancy parameter given by Jackson exhibits a distinct and strong level of independence.
2016	H. Wang et al. [44]	2 × 2 Rod Bundles	Wire	• The installation of wire-wrapped spacers has resulted in decreased circumferential wall temperatures, particularly at low (subcritical) and high (supercritical) bulk-fluid enthalpies. • The temperature gradients along the circumferential

				wall have also been decreased in comparison to those observed in the rod bundle without any covering. • The wire-wrapped bundle exhibits greater mean heat-transfer coefficients compared to the bare-rod bundle.
2015	H.Y. Gu et al. [45]	2 × 2 Rod Bundles	None	• The impact of system characteristics such as heat flux, mass flux, and pressure on the heat transfer of supercritical water in the bundle is comparable to what is observed in tube or annuli.
2014	W. Gang et al. [52]	2 × 2 Rod Bundles	Wire	• The spiral spacer has a beneficial impact on augmenting local heat transfer. • While the overall heat transfer of a 6mm gap channel is superior to that of a 4mm gap channel, the heat transfer around the spacer position is more intense for the 4mm gap. • The origin of this phenomenon may be attributed to a significant decrease in the flow area at the spacer site in the 4 mm gap channel, resulting in increased velocity and intensified turbulence. • The reduction of flow area in the 6 mm gap channel is

				minimal when using the same spiral spacers. • As a result, the heat transfer enhancement in the 6 mm gap is not as significant as in the 4 mm gap.
2014	H. Wang et al. [43]	2 × 2 Rod Bundles	Wire	• The heat transfer process is significantly influenced by system factors. • As the heat flux increases, the readings of wall temperature steadily increase. • Nevertheless, the impact of heat flow on the heat transfer coefficient is significant within the pseudo-critical enthalpy area. The rate at which mass flows has a significant influence on the passage of heat. • The augmentation in heat transfer is a result of the escalation in mass flux. • Of all the system parameters, pressure has the least significant impact on heat transfer, and its influence is primarily observed in the pseudo-critical zone.
2013	Z. Yang et al. [41]	Circular Tube	Grid	• The spacer enhanced the heat transfer at downstream locations. • The affected zone of the spacer effect depended on mass flux.

				• The experiment signifies the reduction in spacer enhancement effect on heat transfer with increasing distance away from the spacer location.
2012	H. Wang et al. [18]	1 Rod Circular Channel	Wire	• Heat transfer deterioration is caused by the drastic change in thermo-physical properties of water near the pseudo-critical temperature. • However, because of the swirling and disturbing effect of the spiral spacer, heat transfer deterioration is eliminated at the location where the spacer is arranged.
2009	H. Li et al. [49]	1 Rod Circular Channel	Wire	• The helical wire-wrapped spacer does not enhance heat transfer significantly under normal heat transfer conditions at supercritical pressures, but it contributes to improvement of the heat transfers in the pseudocritical region and to a downstream shift of the onset of the deterioration when the mass flux is low and the heat flux is relatively high.
2008	V.G. Razumovskiy et al. [56]	7 Rod Bundles	None	• It was found that cooling of the vertical rods by supercritical water could take place both at normal and deteriorated heat-transfer regimes with a

				stable temperature profile along the heated length. • No thermo-acoustic oscillations of hydraulic resistance of the channel were detected. • The impact of the operating parameters on the HTC was analysed.
1972	K. Yamagata et al. [36]	Circular Tube	None	• At sufficiently low heat fluxes, the heat transfer coefficients for horizontal flow are uniform around the tube periphery and are equal to those for vertical flow. • At some higher heat fluxes, the heat transfer coefficient at the bottom of the horizontal tube is higher and the coefficient at the top is lower than those for vertical flow.
1965	H.S. Swenson et al. [57]	Circular Tube	None	• In the pseudocritical region, heat-transfer coefficient is strongly affected by heat flux. • At low heat fluxes, it has a sharp maximum near the pseudocritical temperature. • At high heat fluxes, it is much lower and does not have a sharp peak.

2.3 Numerical Investigation

The capability of the computational fluid dynamics (CFD) approach to mimic the intricate three-dimensional flow and heat transport aspects of nuclear reactors holds enormous potential for growing its role in the future of reactor design and thermal hydraulics analysis. This is due to the fact that the CFD method can replicate these complicated aspects [39,58]. Simulations of nuclear reactor fuel assemblies can be performed with a high degree of accuracy by employing the computational fluid dynamics (CFD) method, which is an effective instrument for the design and optimisation of the structure [46].

One of the proposed models, HPLWR, as well as the rod distance and rod bundle sub-channels, were the subjects of a numerical research that was carried out. Through the utilisation of the increased near-wall treatment, the RNG k-ε model was able to achieve the most accurate prediction. This model was then utilised to explore the flow and heat transfer characteristics in traditional sub-channels. During the investigation of wire-wrapped, tests with spacers demonstrated that the outcome was a strong mixing, and the pressure loss in the assembly was found to be minimal. With the wire-wrapped spacer, the reliability of the wrapping surface improves when the temperature drops below the limits of the material it is made of [59].

An investigation was conducted to determine the sensitivity of the predicted temperature of the wall to turbulence models utilising k-ω and RSM turbulence models. This particular investigation was conducted for both the non-wired and wired wrap configurations. In comparison to the k-ω model, the RSM turbulence model was able to accurately forecast homogenised velocities. This was primarily attributed to its capacity to take into consideration the impacts of anisotropic flow. A degradation in heat transfer is absorbed by the k-ω model with low y + wall treatment, whereas the RSM does not absorb this degradation. As a result of the existence of a wired structure, secondary flow and inter-channel mixing were increased, which led to the wall temperatures being lower when compared to the temperatures of the rod bundles that were not wired [60].

The ATHAS (in-house) code served as the foundation for the development of a sub-channel code for wire-wrapped SCWR rod bundle software. When compared to the scattered grid assembly, the wire-wrapped assembly has the ability to obtain a more ordered profile of the temperature of the coolant, which ultimately results in a lower peak cladding temperature. It is possible for the operating power of the pump to be reduced since the pressure drop in a wire-wrapped rod bundle assembly is lower than the pressure drop in a grid-attached assembly. The assembly flow is significantly impacted by the pitch of the wire cover, which is a persuasive factor. When the H/D is smaller, the crossflow will be improved, the temperature profile of the coolant will be more uniform, and the pressure drop will be bigger [31].

The simulations in STAR-CCM + were compared to the measurements of wall temperature for the 2-rod bundle geometry, and a preliminary evaluation of the simulations was carried out. k-ω and v2-f are two forms of turbulence models that have been reported to have a sensitivity to the predictions made by CFD. When considering all three experimental test scenarios, it can be observed that the k-ω turbulence model failed to accurately predict the results. Under the conditions of subcritical testing, the v2-f model was found to be under-predicted, however it was found to be over-predicted under nearby pseudo-supercritical and supercritical conditions within close proximity. Measurements of the wall temperature obtained with the non-wired geometry demonstrated asymmetrical fluctuation both before and beyond the narrow gap area where the highest temperature was observed. This was demonstrated by the fact that the high temperature was observed. Because of the presence of the wire-wrapped geometry, the findings obtained with the wire-wrapped package indicated asymmetric changes before and beyond the narrow gap area. These changes are attributable to the fact that the wire-wrapped geometry predicts that the temperature will increase when the wired condition is present. When compared to the measurements taken for the non-wired rod bundle, the peak wall temperature data for the wire-wrapped set are substantially lower overall [61].

When considering the quality of the experimental data, it was found that the SST k-ω turbulence model provided the most accurate forecasts for both the pseudocritical and supercritical datasets. On the other hand, the v2-f model provided the most accurate predictions for the sub-critical testing conditions. The increased rate of kinetic turbulence in wire-wrapped geometries led to lower temperatures of the wall as compared to non-wired bundles. This was the case because wire-wrapped geometries were wrapped in wire [62].

When doing early research to investigate the thermal-hydraulic impacts of wrapped wire spacers, the BSL-RSM CFD model is a handy tool to use. The wires that are wrapped around structures cause the fluid particles that are flowing from supercritical water to be forced to follow the curvature of the wires because they prevent the fluid particles from following a straight path. A portion of the flow momentum, which was solely axial to the wires upstream, gets transferred into components of the transverse flow momentum as a result of this particular circumstance. In proportion to the amount of movement of the wires, the transverse velocity increases. A significant impact that was brought about by the wires was the construction of a mechanism that is referred to as the external twisting mechanism, which was located alongside the rod bundle wall and rotated in the opposite direction of the clock. Increasing the number of twists in the wires will result in a higher decrease in pressure values as well as an increase in the range of temperatures [14].

When performing a numerical simulation of computational fluid dynamics (CFD), it is always necessary to strike a balance between the amount of time spent computing and the amount of information acquired from the simulations. The RANS model can be used in general to determine the macroscopic distributions of three-dimensional thermal hydraulic parameters. This is in contrast to the LES, DES, and DNS models, which are able to provide flow field distribution features that are more extensive. On the other hand, they are limited by the computational capabilities that are currently accessible, and they have not yet been directly employed in the engineering process [39].

For the purpose of simulating the flow and heat transfer process of supercritical water in the four-wire structure, the RSM turbulence model with the increased wall treatment is utilised. It is possible to compare the fuel rod bundle's thermal and hydraulic properties with and without the twisted wire. Because the wire is twisted, a strong secondary flow is created. This makes the heat transfer effect between the fuel rod wall and the fluid even stronger. Additionally, the "side flow effect" caused by mass redistribution makes it hard for the large amount of supercritical water that has built up in the outer sub-channels to effectively contribute to the heat transfer process. Specifically, the wrapped wire is what moves the supercritical water through the exterior sub-channels and into the corner sub-channels. When compared to the bare bundle, the peak heat transfer coefficient of the sub-channels inside the bundle went up from 29.95 kW/m^2 K to 41.75 kW/m^2 K. In the same way, the exterior sub-channels' peak heat transfer coefficient went up from 26.70 kW/m^2 K to 35.59 kW/m^2 K. Both of these increases were observed in comparison to the bare bundle. Thermal and hydraulic parameters are not uniform around the circumference for a basic reason, and the local hydraulic diameter is one of the factors that affects how mass is redistributed. The mass redistribution is the fundamental reason. By changing the direction of the fluid flow near the fuel rods, the four-wire structure can get rid of the effects of the local hydraulic diameter on the temperature of the wall around the outside. This, in turn, effectively reduces the circumferential non-uniformity of the fuel rods that are located at the outermost end of the structure. A decrease of 26.48 degrees Celsius was seen in the highest wall temperature, whereas a decrease of 33.90 degrees Celsius was observed in the maximum wall temperature gradient of the section [9].

With regard to the coefficient of heat transfer, it is possible to observe that the utilisation of wrapped wire constructions results in the greatest rise in the transmission of heat in the sub-channels of the corner, which were initially the regions of the bare rod bundles that experienced the highest level of heat transfer. The increase in the number of turns of wire geometry brings about a drop in the average temperature of the rod wall and brings about a homogenization of the temperature distributions of the sub-channel around a rod that is comparable. The numerical simulation of the thermal-hydraulics of SCW in three different sub-channels is presented in Table 2. 2. This simulation was carried out by a number of different researchers.

Table 2. 2 Numerical simulation on thermal hydraulics of SCW in different sub-channels

Year	Authors	CFD Code	Spacer	Flow Geometry	Turbulence Model
2022	O. Bovati et al. [63]	Fluent	Wire	61 Rod Bundles	LES
2019	A. Kiss et al. [14]	CFX	Wire	2x2 Rod Bundles	BSL-RSM
2019	J. Liu et al. [64]	In house code	None	Circular Tube	DNS
2019	C. Eze et al. [65]	Fluent	None	Circular Tube	SST K-ω
2017	K. Podila et al. [25]	STARCCM+	Wire	2x2 Rod Bundles	SST K-ω
2016	L.K.H. Leung et al. [61]	ASSERT, STARCCM+	Wire	2x2 Rod Bundles	K-ω, v2-f
2016	K. Podila et al. [62]	STARCCM+	Wire	2x2 Rod Bundles	SST K-ω, v2-f
2015	J. Xiong et al. [66]	CFX	None	2x2 Rod Bundles	SSG, ω-RSM, BSL-RSM
2015	D. Steven Chang et al. [67]	Fluent, STARCCM+	None	7 Rod Bundles	k-ε, k-ω
2015	A. Kiss et al. [48]	CFX	Wire	2x2 Rod Bundles	RANS
2014	K. Podila et al. [21]	STARCCM+	Grid	7 Rod Bundles	k-ε
2014	K. Podila et al. [60]	STARCCM+	Wire	64 Rod Bundles	k-ω
2014	J. Shan et al. [31]	ATHAS	Wire	40 Rod Bundles	-
2014	K. Podila et al. [68]	STARCCM+	Wire	64 Rod Bundles	k-ω

2013	M. Jaromin et al. [69]	CFX	None	Circular Tubes	SST k-ω
2013	A. Debbarma et al. [59]	Fluent	Wire	1 Rod, Square Channel	RNG k-ε
2011	Y. Zhang et al. [70]	Fluent	None	37 Rod Bundles	RSM
2011	J. Gou et al. [71]	STAR-CD	None	7 Rod Bundles	High Re k-ε
2011	Z. Shang et al. [72]	STAR-CD	None	7 Rod Bundles	High Re k-ε
2010	M.T. Kao et al. [42]	Fluent	None	Circular Tube	RNG k-ε, RSM
2009	Z. Shang et al. [73]	STAR-CD	None	Circular, Square & Hexagonal	High Re k-ε
2008	H.Y. Gu et al. [74]	CFX	None	Square & Triangular	SSG
2007	J. Yang et al. [75]	STAR-CD	None	Square & Triangular	High Re k-ε
2007	X. Cheng et al. [76]	CFX	None	Circular, Square & Triangular	Various Models

2.4 Heat Transfer Behaviours of Supercritical Water in A Heated Channel

Over the course of the 1950s, a number of researchers have been investigating the heat transport characteristics of water when it is subjected to supercritical pressures. During the pseudo-critical temperature stage, the thermo-physical properties of the supercritical water that is flowing through the heated channels undergo changes that are significant. It results to an rate in heat transport that is unprecedentedly supercritical. Heat transfer enhancement (HTE) and heat transfer degradation (HTD) are the two types of heat transfer that are produced as a consequence of the significant changes in characteristics [18,36,55,77,78].

Studies have shown that there is an improvement in heat transfer, and many experts believe that this gain might be attributed to the significant differences in attributes that have been discovered [47,79]. A lot of researchers have discovered that there is an increase in the amount of heat that is transmitted, and they relate this discovery to the significant changes in properties that they have noticed. On the other hand, there hasn't been a lot of research done to find out how the various property alterations will increase the performance of the heat transfer. There is a possibility that the water wall tube will be damaged as a consequence of the soaring wall temperature that is caused by the decreased heat transfer. As a potential way for avoiding this occurrence from taking place, the deterioration of heat transfer has received a lot of interest as a prospective method. The purpose of this method is to maintain wall temperature within acceptable boundaries [55]. Following the progression of the experiments, it became apparent that buoyancy and thermally induced flow acceleration were the primary factors responsible for this degradation. This was detailed in greater depth [80]. At locations where the temperature at the pseudocritical point, which can be defined as the temperature with the specific heat as a local maximum at constant pressure, is between the bulk temperature and the temperature of the wall, the heat transfer may exhibit unusual behaviours such as enhancement or deterioration. This is because the pseudocritical point is the boundary between the bulk temperature and the wall temperature [81].

Because of the buoyancy effect, the HTD occurs when there is a low mass flux. This conclusion has been described as the difference between the two flow patterns in the direction of buoyant force. In their experiments, HTD occurs in the upward flow, but it disappears in the downward flow. This result has been observed [80,82,83]. It is possible for the form of HTD to take place when there is a high flux of mass and a high flux of heat, and this can take place in both upward and downhill flow. It is possible that the primary cause of this deterioration is the acceleration of the flow that is generated by thermal forces [37].

In prior research, the influence of many parameters on supercritical heat transfer was thoroughly explored. These parameters included heat flux, pressure, and mass flow, among others. In addition, the general assumption is that the rate of heat transfer is enhanced when the pressure is reduced, the mass flux is increased, and the heat flux is decreased, as demonstrated in Figure 2. 2, Figure 2. 3 and Figure 2. 4. On the other hand, as was said before, the variations in the characteristics, the influence of buoyancy, and acceleration are significantly related with heat transfer which occurs at supercritical pressure [55,69,84].

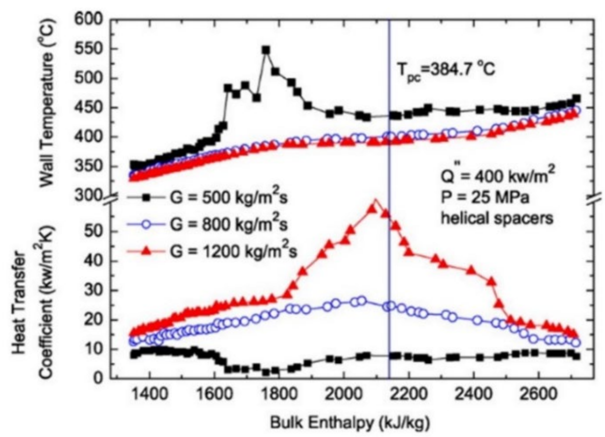

Figure 2. 2 Effect of supercritical water mass flux parameters on heat transfer co-efficient [49]

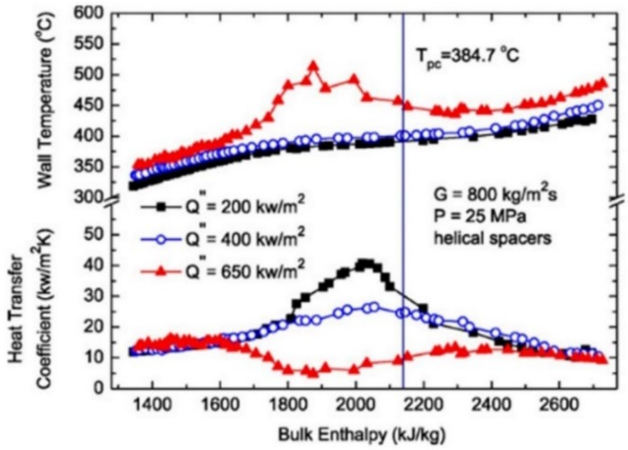

Figure 2. 3 Effect of supercritical water heat flux parameters on heat transfer co-efficient [49]

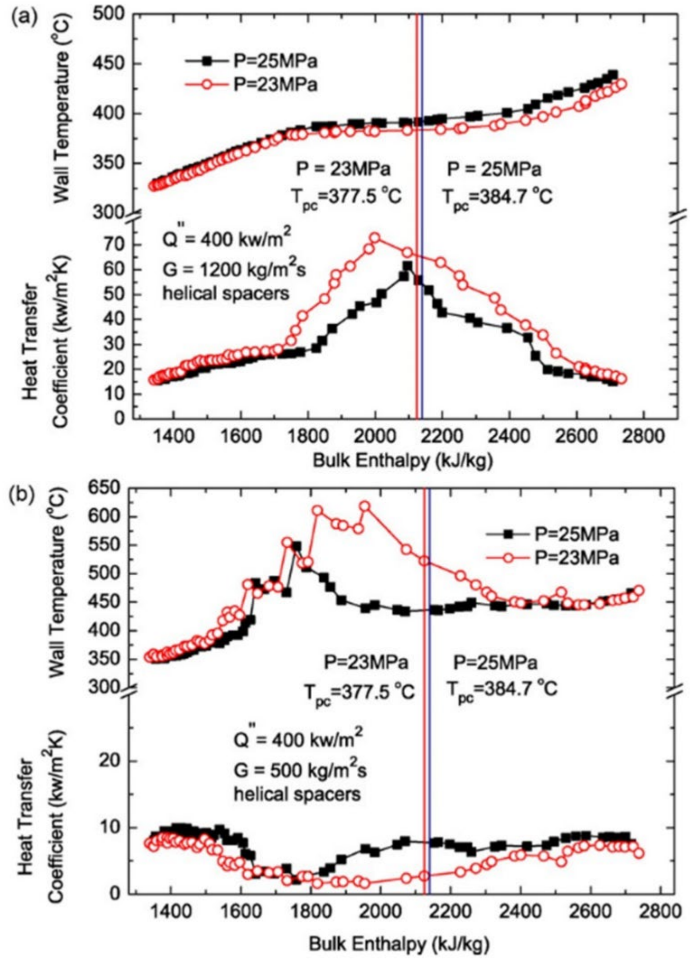

Figure 2. 4 Effect of supercritical water different pressure parameters on heat transfer coefficient a) High mass flux b) Low mass flux [49]

It is generally accepted that the effects of buoyancy are caused by radial density gradients, whereas the effects of acceleration are caused by axial density gradients [49]. At supercritical pressure, the transfer of heat is increased close to the pseudocritical temperature; however, when heat flux is present, the coefficient of heat transfer decreases. It was found that a deteriorated heat transfer phenomenon occurs when the flux of heat is strong enough, and this phenomenon is accompanied by a fast rise in the temperature of the wall. This phenomenon has the potential to either cause the cladding of the fuel assembly to break or to declare a significant temperature fluctuation throughout the perimeter of the fuel cladding, which can result in bowing [50,79,85,86].

One of the factors that contributes to the distinct distinction in the heat transfer characteristics of supercritical and subcritical water is the extreme variability of the properties of supercritical water [87]. It is common knowledge that water at supercritical pressure is a single-phase fluid. In comparison to water, it possesses distinctive qualities regarding the transmission of heat in the subcritical pressure region during the process [36,38,88,89].

Because CHF (critical heat flux) cannot continue to exist under supercritical pressure, the designs of SCWRs typically include explicit restrictions on the temperature of the surface of the cladding. This is done to ensure the safety of the fuel rod assembly [21].

2.5 Summary of Literature Survey

The literature that was discussed above discusses heat transport in supercritical water, which includes heat flow, mass flux, pressure, and spacers that are wrapped in helical wires. According to the findings of the study, heat transfer is investigated in a variety of environments, including helical wire-wrapped spacers. The findings indicate that the spacer does a better job of improving pseudocritical heat transfer, which eliminates degradation in the vicinity of its position. The pseudocritical enthalpy region is particularly affected by pressure, mass flow, and heat flux, all of which have an impact on heat transfer. Depending on the gap size and position, the spiral spacer can improve the degree to which local heat transfer occurs.

In comparison to bare rod bundles, wire-wrapped spacers are superior in terms of their ability to lower circumferential wall temperatures and gradients. This research sheds light on the intricate dynamics of heat transfer in supercritical water, including the effect that variables play and the utilisation of wire-wrapped spacers to enhance heat transfer.

2.6 Research Gaps

According to the literature study, additional research is needed to understand the mechanisms and optimisation parameters that improve supercritical water heat transfer. This study should examine wire-wrapped spacers' effects. These spacers improve heat transfer, according to current studies. The optimal design elements, such as gap size and position, to maximize heat transfer rate are unknown.

Additionally, nothing is known about how design variables like wire wrapping pattern affect wire-wrapped spacers' heat transfer rate. By resolving these shortcomings, supercritical water heat transfer systems could be improved.

2.7 Research Objectives

Based on the above literature survey, some research objectives have been prepared for the current research work, that is included in the thesis. These objectives can be listed as follows:

1) To investigate the temperature distributions in a heat generating rod bundles with helical wire.
2) To investigate the thermal hydraulics of rod bundles with computational simulation of bundles with different helical pitches.
3) To find out the pressure drop by comparing the results with that of straight wire wrap rod bundles.
4) To investigate the effect of wire diameter on thermal hydraulics performance.

Chapter 3

Mathematical Model and Research Methodology

3.1 Introduction

The Supercritical Water Reactor (SCWR) is a ground-breaking advancement in the field of advanced nuclear reactor architecture, offering improved efficiency and lower operational expenses. An essential factor in achieving these advantages is the precise modelling and simulation of the intricate thermal-hydraulic phenomena occurring within the reactor. This chapter explores the creation and use of mathematical models for Computational Fluid Dynamics (CFD) and optimisation approaches. These models are crucial for predicting and improving the performance of Supercritical Water-cooled Reactors (SCWRs).

The mathematical modelling of Supercritical Water Reactors (SCWRs) is inherently intricate due to the distinctive characteristics of supercritical water, which undergoes significant alterations in thermo-physical properties in close proximity to the critical point. The alterations can have a substantial impact on the transmission of heat and the movement of fluids, rendering conventional models insufficient for accurately representing the intricacies of supercritical conditions. Therefore, there is a strong requirement for resilient and accurate mathematical models that can precisely depict the behaviour of supercritical fluids.

Computational Fluid Dynamics (CFD) is essential in this context since it offers a robust method for simulating the movement of fluids and the transfer of heat in Supercritical Water-Cooled Reactors (SCWRs). Computational Fluid Dynamics (CFD) allows for the thorough examination and interpretation of flow properties in the reactor core and related systems by solving the governing equations of fluid mechanics and heat transport. This chapter will describe the creation of computational fluid dynamics (CFD) models designed specifically for supercritical water-cooled reactors (SCWRs). It will cover the establishment of boundary conditions, meshing procedures, and the choice of suitable turbulence models to accurately simulate the intricate flow patterns.

Optimisation approaches enhance CFD simulations by allowing for the precise adjustment of reactor design and operational parameters to get the best possible performance. Optimisation entails the manipulation of design variables in order to minimise or maximise objective functions, such as thermal efficiency or safety margins, while simultaneously adhering to a set of limitations. Optimisation in the context of SCWRs can be used to determine the optimal configurations and operating circumstances that provide a balance between performance, safety, and economic factors. This chapter will examine different optimisation approaches, such as gradient-based and evolutionary algorithms, and analyse how they might be applied to address the design issues of SCWR.

The research methodology described in this chapter combines these computational methodologies to create a unified framework for SCWR analysis. The process encompasses a methodical approach to constructing models, verifying them using experimental data, and continuously improving them through optimisation. The methodology prioritises interdisciplinary collaboration, integrating knowledge from thermal-hydraulics, materials science, and reactor physics to tackle the complex difficulties of SCWR development.

This chapter introduces sophisticated SCWR analysis methods such mathematical modelling, CFD simulation, and optimization. This document aims to help scholars understand these methodologies so they can advance SCWR technology. This will advance sustainable and efficient nuclear energy generation.

3.2 Experimental Loop

The previous experiments served as the basis for all computational studies. The studies were conducted at the steam-water test loop at Xi'an Jiaotong University, which operates at high temperatures and pressures. Figure 3. 1 displays a schematic diagram of the test loop. The water tank contained distilled and de-ionized supply water, which was forced through a filter using a high-pressure plunger pump running at pressures of up to 40 MPa. The predetermined quantity of water was modified by manipulating the primary valve, while the excess water was redirected back to the water tank via a bypass line. The high-temperature discharged water from the test portion was utilized to pre-heat the water by directing it into a heat exchanger. The temperature of the system was raised to the predetermined level using the preheaters before to entering the test portion. The heat extracted from the bundle was transferred to the heat exchanger, where it preheated the inlet water. Subsequently, it was sent to a condenser in order to achieve further cooling that fulfils the pump's specifications. The chilled water returned to the water tank via an additional valve and a rotameter. The primary valve and the bypass valve were adjusted to control the pressure and mass flow rate. A Rosemount 3051 capacitance-type pressure transmitter was used to measure the pressure at the test section's entry. A set of mass flow meters was used to measure the mass flow rate. A Rosemount 3051 capacitance-type differential pressure transducer was used to measure the change in pressure across the test section. Fluid temperatures were recorded by employing a collection of K-type thermocouples, each of which was enclosed within a 3-mm outer diameter sheath. The collection and recording of all data were conducted utilizing an IMP 3595 data collecting device [44].

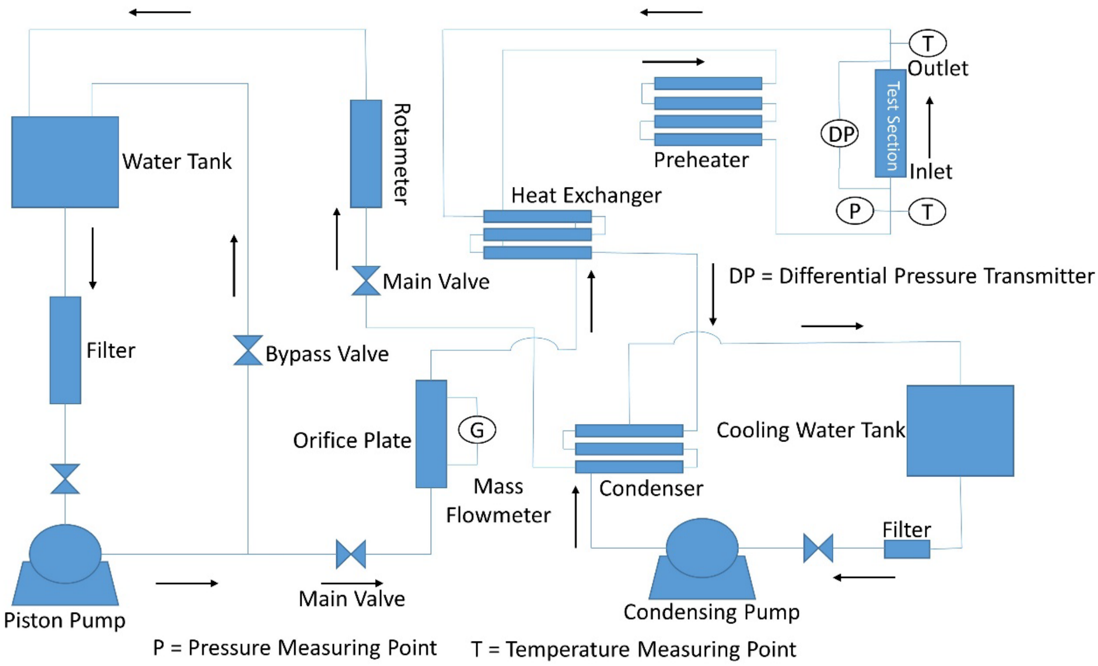

Figure 3. 1 Schematic diagram of the supercritical pressure water test loop

3.3 Computational Modelling

Computational Fluid Dynamics (CFD) is a pivotal tool in computational modelling that involves the use of numerical methods and algorithms to simulate the behaviour of fluids, including their interactions with solid boundaries and internal heat transfer processes. In the context of complex systems like Supercritical Water Reactors (SCWRs), CFD enables detailed analysis of fluid flow, temperature distribution, and pressure variations within the reactor core and associated components. By solving the fundamental equations of fluid dynamics—such as the Navier-Stokes equations—CFD provides insights into the intricate thermal-hydraulic phenomena that occur under supercritical conditions, which are critical for optimizing reactor performance and ensuring safety. The precision and versatility of CFD make it an indispensable tool in the design and optimization of advanced nuclear reactors, facilitating the development of models that can predict fluid behaviour under a wide range of operational scenarios.

3.3.1 Test Section Geometry Preparation

Using the specifications that are presented in Table 3. 1 and the software known as Ansys 2022 R1, the computer-aided design (CAD) model of the computational domain was created. In light of the fact that conjugate heat transfer was taken during the inquiry, the solid and fluid areas of the geometry were taken into consideration and did not require any pre-serving. The cad model should receive special attention when it comes to the wire-wrapped geometry. This is because the wire contact might result in mesh singularity, which in turn results in high skewness in the mesh generation, which ultimately results in a converged solution that is less accurate. Researchers from a variety of institutions have researched the wire-wrapped structure in order to determine the answer to the difficulty described above. This was accomplished by embossing the wire structure onto the surface of the rod in order to give it a cross-sectional shape that is nearly cylindrical, as depicted in Figure 3. 2 [62,90]. It was the wrapping arrangement of the wire that was altered to a mixed blend of clockwise and anti-clockwise directions. This was the new pattern that have been taken into consideration.

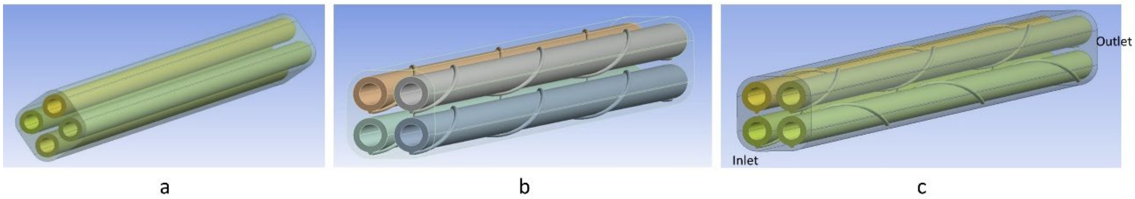

Figure 3. 2 Geometry of (a) bare rod (b) clockwise wired rod (c) new pattern

Table 3. 1 Dimensions of a 2x2 wire-wrapped rod bundle

Geometry	Numerical Value
The outer diameter of the rod (mm)	8.0
Wall thickness of the rod (mm)	1.5
Pitch to diameter ratio	1.18
Hydraulic equivalent diameter (mm)	3.88
Affected heated length (mm)	600
Center distance between two rods (mm)	9.44
Flow area (mm^2)	181.91
Diameter of wrapped wire (mm)	1.2
The pitch of wrapped wire (mm)	200
Rod to wall-corner gap (mm)	1.4

3.3.2 Mesh Generation

In the computational domains of solids and fluids, the generation of the mesh is accomplished by the use of a variety of meshing techniques. Due to the fact that the conjugate heat transfers was taken into account between the rod and the fuel, the non-conformal mesh that was formed was matched between the geometry of the rod and the fuel, as seen in Figure 3. 3. For the purpose of effectively addressing the boundary layer issue, ten layers of inflation were applied to the outer wall of the rod. A y+ value of less than one was taken into consideration in order to record the significant fluctuation of the fluid properties in the near-wall region, which could potentially lead to a decline in the transfer of heat. At the same time when the height of the first layer of thickness was judged to be 0.00296 mm, the base mesh size was considered to be 1.152 mm respectively. The skewness, on average, is 0.2, which indicates that the element distribution is balanced and that there is little mesh distortion. When the average orthogonal quality is 0.8, it shows that the mesh has a good geometric regularity, which improves the accuracy of the simulation. Furthermore, the data highlight the importance of mesh quality in order to produce accurate numerical estimates.

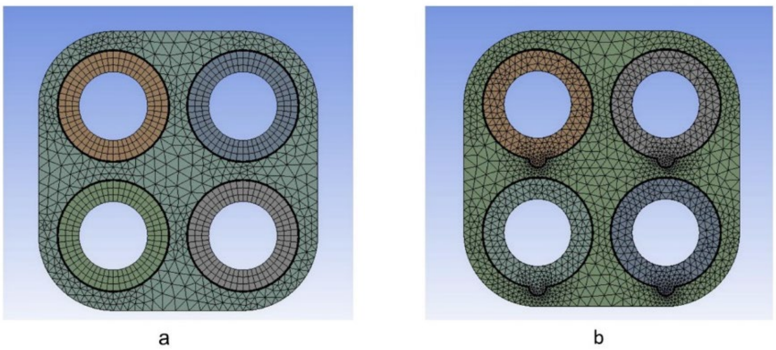

Figure 3. 3 Mesh representation of (a) bare rod (b) wired rod

3.3.3 Governing Equation

In addition to continuity, momentum, and energy equations, the turbulence model is solved with the help of the Ansys 2020 R1 programme. This software utilised the discretization of the equations by employing the finite volume approach. In the case of heat transfer and fluid flow, respectively, the following sets of equations from Equation 3.1 – Equation 3.6 serve as the governing ones:

Mass Conservation Equation:

$$\frac{\partial \bar{\rho}}{\partial t} + \frac{\partial (\bar{\rho}\bar{u}_i)}{\partial x_i} = 0 \qquad (3.\ 1)$$

Momentum Conservation Equation:

$$\frac{\partial (\bar{\rho}\bar{u}_i)}{\partial t} + \frac{\partial (\bar{\rho}\bar{u}_i\bar{u}_j)}{\partial x_j} = -\frac{\partial \bar{p}}{\partial x_i} + \frac{\partial}{\partial x_j}\left(\bar{\tau}_{ij} - \underbrace{\bar{\rho}\bar{u}_i\bar{u}_j}_{A}\right) + \underbrace{\bar{\rho}g_i}_{Buoyancy\ Source} \qquad (3.\ 2)$$

The viscous stress tensor, $\bar{\tau}_{ij}$ can be expressed as follows:

$$\bar{\tau}_{ij} = \eta\left[\left(\frac{\partial \bar{u}_i}{\partial x_i} + \frac{\partial \bar{u}_j}{\partial x_i}\right) - \frac{2}{3}\delta_{ij}\frac{\partial \bar{u}_k}{\partial x_k}\right] \qquad (3.\ 3)$$

Where the terms A and η in Eqs. (2) and (3) refer to Reynolds stress tensor and molecular viscosity, respectively. Since we used the SST k-ω turbulence model in our analysis, the stress terms were calculated using the Boussinesq eddy viscosity hypothesis.

Energy Equation:

$$\frac{\partial (\bar{\rho}\bar{T})}{\partial t} + \frac{\partial (\bar{\rho}\bar{u}_j\bar{T})}{\partial x_j} = \frac{\partial}{\partial x_j}\left(\underbrace{q_{HF}}_{B} - \underbrace{q_{HF}^{t}}_{C}\right) \qquad (3.\ 4)$$

Where the terms B and C in the above equation are associated with fluxes of heat and turbulent, they can also be defined by the Boussinesq hypothesis as in the below equations:

$$q_{HF} = \frac{k}{c_p}\frac{\partial T}{\partial x_i} \qquad (3.\ 5)$$

$$q_{HF}^{t} = -\bar{\rho}\bar{u}_j'\bar{T}' = \frac{\mu_t}{Pr_t}\frac{\partial T}{\partial x_j} \qquad (3.\ 6)$$

Where, Pr_t refer to the turbulent Prandtl number, which is used to define the turbulence mixing efficiency in the turbulent flow and has a commonly accepted value of 0.85 in the Ansys Fluent code.

3.3.4 Boundary Condition

The boundary conditions for the pressure outlet and the mass flow inlet were applied to the beginning and the end of the computational domain. As a result of the application of consistent heat flux boundary conditions, the inner face of the fuel rod was manufactured. In order to effectively capture the heat transfer characteristics in the supercritical flow condition, the gravitational force was taken into consideration in each and every simulation system that was used in the current experiment. In order to better evaluate the numerical models, the researchers have determined that the values of the turbulent Prandtl number (Pr_t) for the simulation of bare and wired rods are 1.8 and 1.2 [62]. These values have been determined by the researchers in the past. Table 3. 2 offers a visual representation of the values of the various input parameters for the bare rod.

Table 3. 2 Input parameters for simulation

Parameter	Numerical Value
Pressure (MPa)	25
Mass Flux (kg/m^{2}s)	1000
Heat Flux (kW/m^2)	400
Inlet Temperature for wired rod (°C)	430.72
Inlet Temperature for the bare rod (°C)	419.3

3.3.5 Operating Condition

As can be seen in Figure 3. 4, the most defining characteristic of SCW's thermal-hydraulics is the excessive non-linear variation of thermo-physical properties with respect to the critical and pseudo-critical temperatures. The information that is presented in Figure 3. 4 was obtained from the database that is accessible online in the National Institute of Standards and Technology (NIST) [91]. Often referred to as a pseudo-critical transition, the non-linear variability of the pseudo-critical temperature is mild to the higher pressures. This phenomenon is characterised by the fact that it occurs. The fluid particles undergo a change in behaviour during the pseudo-critical transition from a liquid state to a gas-like state. This transition is carried out in such a way that the fluid particles undergo this change [14,92].

In spite of this, the water will not exhibit any signs of phase transition when subjected to supercritical pressure. Furthermore, there is a pseudo-critical point that occurs when the properties of water, such as specific heat and density, experience significant changes under subcritical pressure, in a manner that is analogous to the steam boil cycle [93,94]. In Figure 3. 4, the thermo-physical properties of water are depicted. These attributes include a sharp variability in the properties of supercritical water, which leads to a considerable difference between the heat transfer characteristics of supercritical and subcritical water [50].

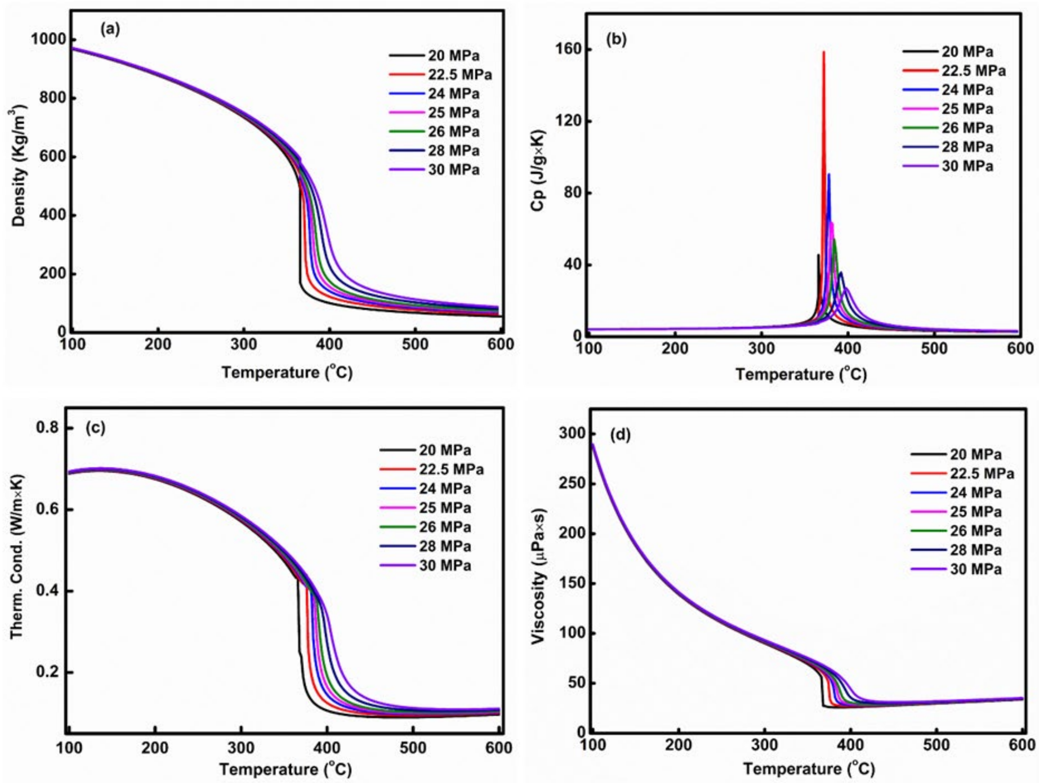

Figure 3. 4 Thermo-physical properties of supercritical water

It is through the transfer of heat between the surface and the particles of the flowing fluid that the creation of the thermal boundary layer takes place. The pressure change that occurs during this process is insignificant. In the context of the analysis of the convective transmission of heat within the tube, it is therefore convenient to take into consideration the factor of fluid characteristics that change with the change in temperature while the pressure remains constant. When the saturation temperature is attained, the thermodynamic and transport properties of the fluid particles undergo discontinuous changes at sub-critical pressure, which is referred to as phase transition. Under conditions of supercritical pressure, this activity would be replaced by a continuous variation of characteristics that would be concentrated slightly above the critical value within a relatively limited temperature band [87].

In the vicinity of the pseudocritical line, the density experiences a significant reduction. A large peak may be found in the coefficient of thermal expansion, which exhibits behaviour that is extremely near to that of the specific heat. It is observed that the thermal conductivity decreases as the temperature increases. The local maximum, on the other hand, is located relatively close to the pseudo-critical point. Once the temperature reaches the pseudo-critical point, the thermal conductivity begins to drop dramatically. A similar pattern of behaviour is also exhibited by the dynamic viscosity. Because of the significant rise in specific heat capacity, the Prandtl number reaches its highest value at the pseudo-critical point. This is the case because of the huge peak [50].

The rational function with a first-order numerator and a second-order denominator was used to fit the properties that were obtained from the NIST database. The rational function was specified in a piece-wise manner and was used to fit the properties as a function of temperature. Following that, a user-defined function as shown in Appendix I, was used in Ansys Fluent to call upon the properties that had been fitted at 25MPa of water pressure.

The fitted equations for the various thermo-physical properties of supercritical water can be defined from Equation 3.7 – Equation 3.16, as:

Density (ρ)

$$\frac{1035.78707-2.57809T}{1-0.00176T-1.3742\times10^{-6}T^2} \quad \{0°C - 386.67°C \tag{3. 7}$$

$$\frac{417.96962-1.20897T}{1+0.00729T-2.69036\times10^{-6}T^2} \quad \{386.67°C - 1000°C \tag{3. 8}$$

Thermal Conductivity (K)

$$\frac{0.5989-6.88295\times10^{-4}T^2}{1-0.00284T+6.01634\times10^{-6}T^2} \quad \{0°C - 383.34°C \tag{3. 9}$$

$$\frac{0.02941-8.82536\times10^{-5}T}{1-0.003761+2.94392\times10^{-6}T^2} \quad \{383.34°C - 515°C \tag{3. 10}$$

$$0.39694 - 0.00159T + 2.945 \times 10^{-6}T^2 - 2.23933 \times 10^{-9}T^3 + 6.44297 \times 10^{-13}T^{-4} \quad \{515°C - 1000°C \tag{3. 11}$$

Viscosity (μ)

$$\frac{0.00174+1.10612\times10^{-5}T}{1+0.03858T+4.93808\times10^{-4}T^2} \quad \{0°C - 383.34°C \tag{3. 12}$$

$$\frac{1.31196\times10^{-5}-3.47819\times10^{-8}T}{1-0.00384T+3.18022\times10^{-6}} \quad \{383.34°C - 420.01°C \tag{3. 13}$$

$$\frac{1.61037\times10^{-5}+1.11751\times10^{-8}T}{1-7.49286\times10^{-4}T+3.0576710^{-7}T^2} \quad \{420.01°C - 1000°C \tag{3. 14}$$

Specific Heat (Cp)

$$\frac{4.28532-0.00979T}{1-0.0019T-1.77617\times10^{-6}T^2} \quad \{0°C - 385.01°C \tag{3. 15}$$

$$\frac{1.13664-0.00425T}{1-0.00289T+7.11438\times10^{-7}T^2} \quad \{385.01°C - 1000°C \tag{3. 16}$$

3.3.6 Solver and Solution Methods

An implicit pressure-equations solver, such as the Semi-Implicit Method for Pressure-Linked Equations (SIMPLE), was utilised in order to calculate the governing equations for the computational domain in three dimensions with coupled pressure and velocity. This was accomplished by the utilisation of a segregated solver.

Second-order differential techniques were utilised in order to efficiently solve all of the problems. The flow, pressure, and energy under-relaxation factors (URF) were each assigned values of 0.7, 0.2, and 0.99 from the beginning of the experiment. When the fluid temperature has reached a constancy in the fluctuation at the monitored segment, which is Y = 0.5 m from the inlet, and the residuals have come to six orders of magnitude, the solution is said to have converged. This is the point at which the solution is considered to have reached a conclusion.

3.3.7 Verification of CFD Model

Various turbulence models were evaluated by comparing them to the 2 × 2 bundle trials to determine whether or not they were suitable for accurately estimating the temperatures of the fuel rods under supercritical, pseudo-critical, and subcritical test circumstances. Currently, there is no turbulence model that is suitable for supercritical flows that is suitable for its intended use. The findings of Podila et al. [62] revealed that the SST k-ω turbulence model provided an accurate approximation of the experimental results when applied to pseudo and supercritical test data. However, when used to subcritical test data, the v2-f model proved to be the most effective. In the supercritical test condition, the SST k-ω turbulence model was implemented. This model was derived from the earlier confirmed turbulence model that was found in the literature [61,62].

3.3.8 Grid Independence Test

Using the Richardson Extrapolation method, a grid convergence study was carried out in order to investigate the degree to which the CFD solution is sensitive to the size of the grid and to quantify the amount of discretization error introduced [95].

The research project involved the construction of three distinct grids, which were referred to as Coarse (3), Medium (2), and Fine (1) grids. The cell counts of these grids were 8.1×10^5, 15×10^5, and 25×10^5, respectively. since of the need for clarity and ease of tracking, the representative cell length h_c was utilised for the study instead of the cell count N_c. This was done since hc is equal to zero, but N_c is equal to ∞ for an indefinitely fine grid. The length of the representative cell is h_c, and it is provided by Equation 3.17, as:

$$h_c = \frac{1}{N_c} \sum_{Cells} A_p^{1/2} \qquad (3.\ 17)$$

where A_p represents the area of the cell and N_c represents the number of cells in the grid. Following this, the refinement ratio, denoted by the letter r, was computed. This ratio represents the ratio of h_c of medium grids to fine grids. It is recommended by that the representative cell lengths (h_c) should differ by at least 30% between each grid size [95]. The subsequent phase involved determining the order of convergence p of the grid by utilising the values of the solution variable ϕ for the Coarse (3), Medium (2), and Fine (1) grids. Within the scope of this particular investigation, the ratio of temperature was chosen to serve as the solution variable. A quantifiable representation of the change in discretization error that occurs as a result of a change in grid size is contained within the parameter p. The order of convergence, denoted by the letter p, can be expressed in Equation 3.18, as

$$p = \frac{ln\left(\frac{\phi_3 - \phi_2}{\phi_2 - \phi_1}\right)}{ln(r)} \qquad (3.\ 18)$$

Estimation of the Grid Convergence Index (GCI) was performed after the parameter p was discovered. Within the context of enhancing the relative error metric, Roache [96] put up this proposition. To calculate the relative error, a factor of $1/r^p - 1$ is applied to the data. The order of convergence of the solution is not contained within the definition of relative error; however, this scaling takes into account the order of convergence. It was determined by Roache [96] that a factor of safety of 1.25 was necessary to finish its definition. As a result, it is provided by Equation 3.19 and Equation 3.20, as:

$$GCI = \frac{1.25e}{r^p - 1} \quad (3.\ 19)$$

Where,

$$e = \frac{\phi_2 - \phi_1}{\phi_1} \quad (3.\ 20)$$

Table 3. 3 Grid independence characteristics

Mesh	Base Mesh Size (mm)	Total Mesh Count (millions)	Mesh Refinement Ratio (r)	Value	Maximum Fluid Temperature (°C)	Maximum Rod Temperature (°C)	Meshing Time (minutes)	Run Time (Days/Hours)
1	0.8	25	-	-	479.124	480.153	180	13.3/320
2	0.96	15	r_{21}	1.2	479.284	480.315	150	7.08/170
3	1.152	8.1	r_{32}	1.2	479.362	480.362	120	4/96

There is a parametric analysis of grid cells that is carried out by altering the number of grid cells that are present in the computational domain for the bare rod. This is done in order to obtain more accurate computational findings. Richardson's

Extrapolation (RE) method and Grid Convergence Index (GCI) procedures were utilised in order to assess the variation of the key parameter, which is the maximum fluid temperature and the maximum rod wall temperature at the sectional plane. This evaluation was carried out in conjunction with the change in the mesh size. It was decided to implement the GCI using three different mesh sizes. Alternating the base mesh size was the method that was utilised to achieve the increment in mesh count, as demonstrated in Table 3. 3 and Figure 3. 5. It is possible for the most refined mesh to make an accurate prediction of the actual solution, which is a reliable estimate. Nevertheless, for the current numerical study, it is possible to take into consideration a coarser mesh (mesh#3) because the relative error between GCI_{32} and GCI_{21} is rather low.

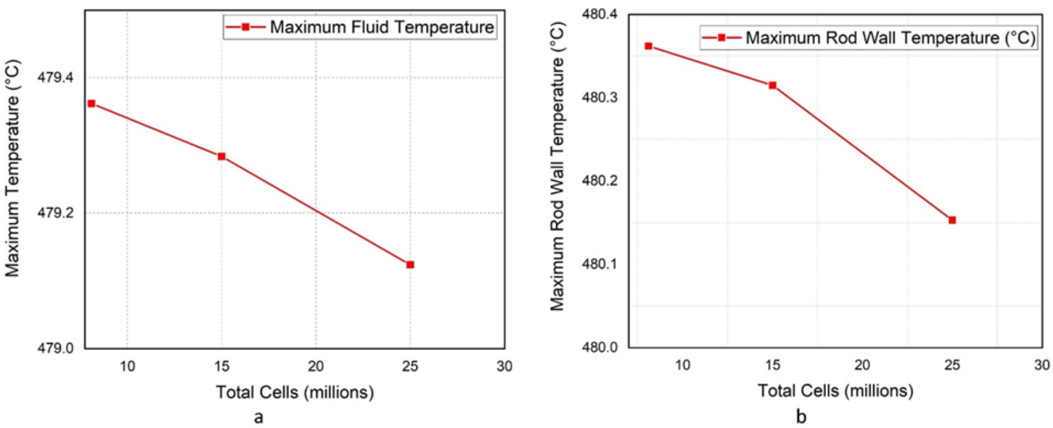

Figure 3. 5 Grid Independence Test at Different Base Mesh Sizes (a) Maximum Fluid Temperature (b) Maximum Rod wall Temperature

3.3.9 Validation of the CFD Simulation

In order to ensure the accuracy of the simulated results, the experimental data gathered from the tests conducted at Xi'an Jiao Tong University with a 2 × 2 rod bundle experimental setup was utilised to validate all of the simulated results [43,44].

The circumferential rod temperature for bare and wired rod has been verified, as can be seen from Figure 3. 6 at the section plane of Y = 0.5 m from the inlet.

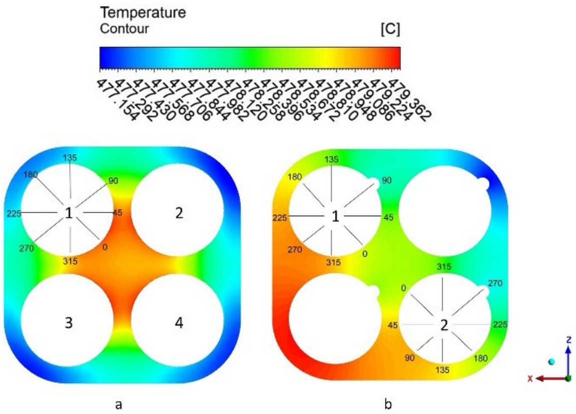

Figure 3. 6 Temperature distribution of (a) bare rod and (b) wired rod

A representation of the temperature distribution in the sectional plane can be found in Figure 3. 6. As can be observed in Figure 3. 7, the circumferential rod temperature for both bare and wired rod has been checked. This was done at the section plane of Y = 0.5 metres from the entrance.

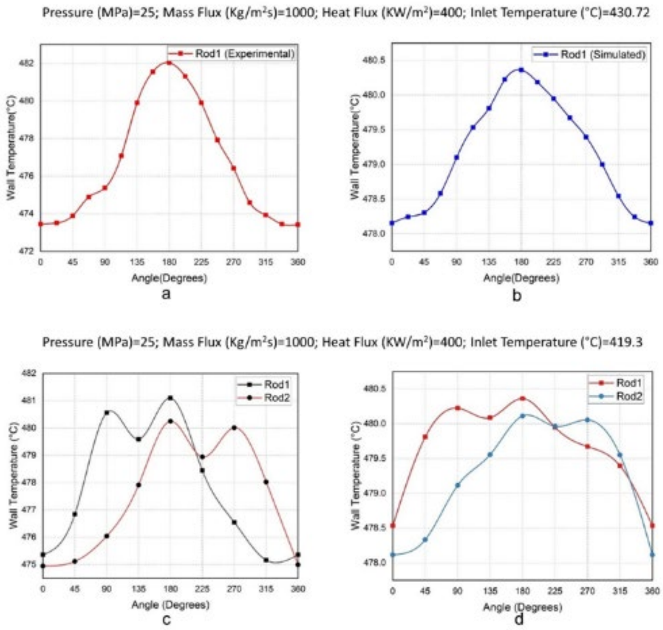

Figure 3. 7 Circumferential Wall Temperature of the (a) Bare Rod Experimental (b) Bare Rod Simulated (c) Wired Rod Experimental (d) Wired Rod Simulated

3.4 Optimization Framework

Surrogate models and genetic algorithms are frequently incorporated into an optimisation framework that makes use of computational fluid dynamics (CFD) data. This is done in order to quickly navigate the complicated design space of engineering challenges. As computationally inexpensive approximations of the detailed CFD simulations, surrogate models, such as Kriging or Radial Basis Function networks, capture the fundamental relationships between input parameters and performance measures. Surrogate models are used in the field of computational fluid dynamics (CFD). Because these models are trained on a collection of data points that were created by CFD, they are able to provide accurate predictions of the behaviour of the system without the need for a significant amount of computer resources. For the purpose of exploring the design space, genetic algorithms (GAs), which are derived from the concept of natural selection, are utilised. These GAs work by iteratively developing a population of candidate solutions in the direction of optimal configurations.

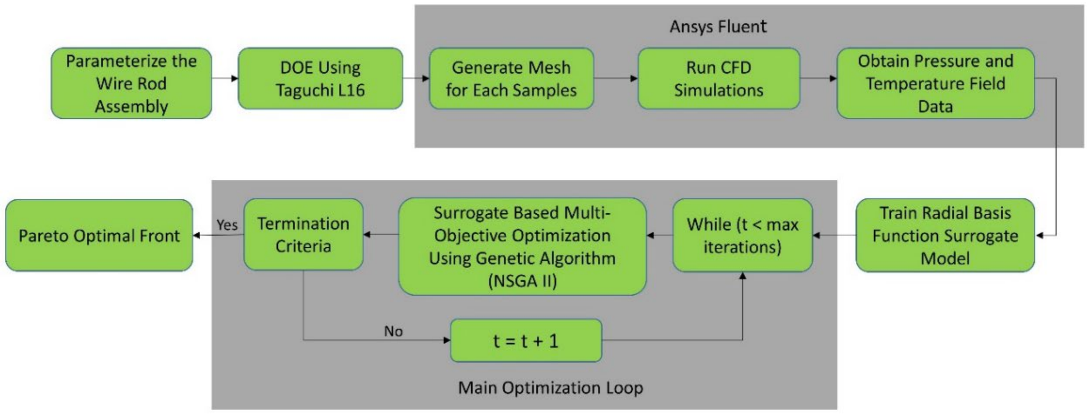

Figure 3. 8 Flowchart of the Optimization Framework

The framework finds a compromise between accuracy and computational efficiency by utilising surrogate models to evaluate fitness functions within the GA. This allows for the effective optimisation of complicated fluid dynamics issues, such as those found in Supercritical Water Reactors (SCWRs), which are a type of reactor. Not only does this method improve the speed at which the optimisation process is carried out, but it also makes it easier to find design ideas that are both original and high-performing. There is a flowchart that illustrates the full structure, and it can be found in Figure 3. 8.

3.4.1 Data Preparation

The generation of the necessary training data is an essential stage that must be completed before one can create a surrogate model that is successful. In order to accurately represent the full design space, the data points need to be distributed evenly. When it comes to collecting data points in the design space, the Taguchi Design of Experiment is a method that is widely available.

3.4.1.1 Taguchi Robust Design

To build a product or process design that is robust or insensitive to the uncontrollable or noisy characteristics is a core premise of the Taguchi technique. This can be accomplished by using the Taguchi method. Although it is possible to exercise control over the levels of noise factors during the design or development phase of the product, it is not possible to exercise control over them during the manufacturing or usage phase of the product. The goal of a robust design is to identify the settings of the control elements or parameters in such a way that the fluctuation in the response variable (or the intended quality feature) is minimised. This is because the impact of various noise components, such as humidity, temperature, and so on, cannot be totally removed. In order to produce a robust design, it is necessary to reduce the impact of the noisy factors and to pick the controllable factor levels in such a way that the desired quality characteristic is as close as possible to the target value. The idea of resilient design takes advantage of the nonlinearity that exists in the interaction between the control parameters and the noise elements that have an effect on the response quality characteristic. The purpose is to locate a combination of control parameter values that displays the least amount of fluctuation in the response variable in relation to a target level that is desired [97].

3.4.1.2 Selection of Sampling Points by Taguchi Method

This section provides a comprehensive explanation of the numerous procedures that are involved in selecting the sampling points by Taguchi method for the numerical simulations that are currently being performed [98].

- The goal of the analysis must be defined.
- Defining the quality attributes (performance metrics).
- Determining the influencing factors and their levels.
- Deciding on the most suitable orthogonal array (OA).

Finding the perfect balance between enhancing heat transfer and lowering pressure drop will determine the ideal design of the fuel assembly, which will result in the highest possible level of performance. Within this context, the objective function that pertains to heat transfer performance is defined as the maximum temperature divided by the area-averaged surface temperature of the fuel rod.

Here we can see how well the surface temperature is distributed using this function in equation 3.21.

$$O_T = \left(\frac{T_{max}}{T_{avg}}\right)_{heated\ surface} \qquad (3.\ 21)$$

With respect to pressure drop, the definition of the second objective function, which is equivalent to friction factor, is given in Equations 3.22 – Equation 3.24.

$$O_P = \left(\frac{\Delta P}{\Delta P_0}\right) \qquad (3.\ 22)$$

Where,

$$\Delta P_0 = f_0 \frac{\rho \bar{U}^2 H}{2D_e} \qquad (3.\ 23)$$

And,

$$f_0 = 2(2.236 lnRe - 4.639)^{-2} \qquad (3.\ 24)$$

Two dimensionless design variables as shown in Figure 3. 9, D_W/D and H/D, were utilized in this instance. H/D is the ratio of wire-wrap pitch to fuel rod diameter, while D_W/D is the ratio of wire-spacer diameter to fuel rod diameter. Both of these ratios are considered to be proportional to one another. Table 3. 4 contains a presentation of the non-dimensional design variables together with the ranges of those variables. The ranges are determined by utilizing previous experimental investigations that pertain to thermal hydraulics as well as some preliminary computations.

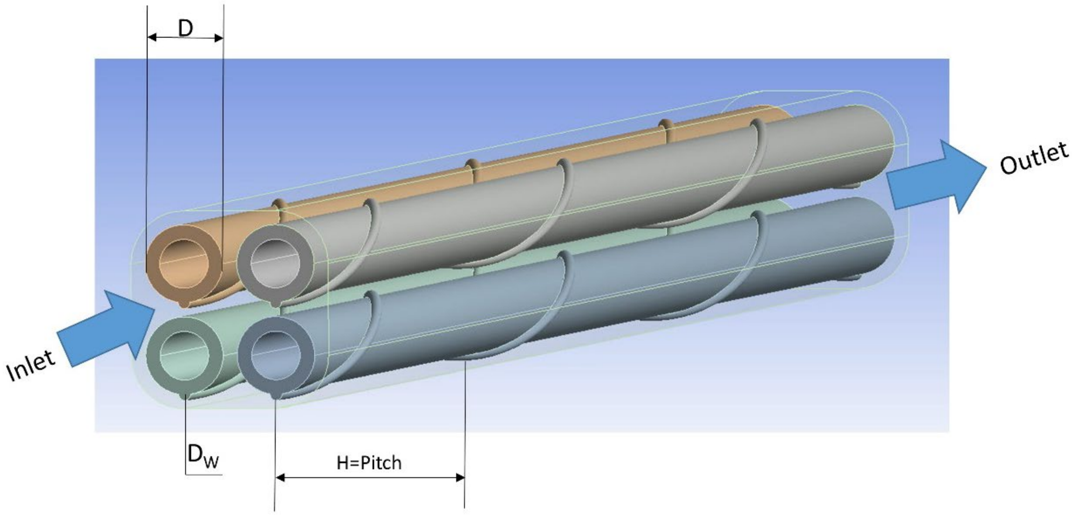

Figure 3. 9 Computational Domain Showing Pitch of Wire Wrapped in Rod

Table 3. 4 Design Variables with Ranges

Design Variable	Lower Bound	Upper Bound
H/D	12.5	75
D_W/D	0.075	0.15

The techniques developed by Taguchi are utilised in the production of the designs for statistical calculations. These methodologies are well-known for their reliability in resolving engineering issues. In Taguchi's method, distinct levels are assigned to each factor in an experiment based on the results of the experiment. Here two factors and four levels were taken. Both the process parameters and the operations themselves are efficiently optimised by these easy procedures. For the purpose of determining the effects of varying amounts of a number of heat transfer and pressure loss parameters, an orthogonal L_{16} array provides the necessary structure as shown in Table 3. 5.

Table 3. 5 Orthogonal Matrix of Experimental Design and Output Variables

Run	H/D	Dw/D	O_T	O_P
1	12.5	0.1	1.055	8.075
2	12.5	0.125	1.052	8.078
3	12.5	0.15	1.05	8.082
4	12.5	0.075	1.058	8.071
5	25	0.1	1.065	6.084
6	25	0.125	1.062	6.086
7	25	0.15	1.061	6.089
8	25	0.075	1.066	6.081
9	37.5	0.1	1.074	5.098
10	37.5	0.125	1.071	6.011
11	37.5	0.15	1.069	6.025
12	37.5	0.075	1.075	5.097
13	75	0.1	1.084	5.052
14	75	0.125	1.081	5.056
15	75	0.15	1.079	5.059
16	75	0.075	1.087	5.051

3.4.2 RBF Surrogate Modelling

It is necessary to have expensive computational resources in order to solve design optimization challenges that involve a large number of computer simulations. Using surrogate models could be a solution to the problem of high computing costs associated with simulations. These models are meant to quickly explore the design space and guess the output variables by putting together a response surface from a small number of trials. Consequently, this makes it possible for researchers to simulate complex systems with greater precision while using fewer processing resources. In order to train these models, a subset of the simulation data is used, and the remaining data is used to validate them. Previous research has demonstrated that the RBF surrogate model is capable of effectively describing substantial nonlinear interactions between variables. The fact that it is straightforward, straightforward to deploy, versatile, and efficient makes it the ideal surrogate model for this inquiry [99–104].

The value of each design sample acquired by the Taguchi Design of Experiment L_{16} technique is recorded as x_i, and the value of its related objective function is recorded as F_i (where i can range from 1 to n). Here, n is the total number of samples that were produced using the L_{16} method. This is the Basis Function that is chosen. A lot of people use the Gaussian basis function for radial basis function (RBF) proxy models because it can accurately show smooth changes in the data. This is because of its ability to capture smooth variations. A Gaussian distribution that is cantered at each data point is used to generate this basis function. By using this distribution, a smooth surface can be made that can be used to model how the underlying system works. The computational fluid dynamics (CFD) data that was shown is pretty smooth and doesn't have many big changes. Because of this, the Gaussian basis function makes sense for the problem that is being talked about right now. The Gaussian basis function is computationally efficient, which is a crucial factor to take into consideration when working with huge data sets. Through the utilization of the Gaussian basis function within the RBF model, it is possible to effectively represent the behaviour of the system in a manner that is both simplified and computationally efficient [105].

Consequently, the RBF can be stated in Equation 3.25, as:

$$\hat{y} = \sum_{i=1}^{n} \lambda_i \, \phi(r(x, x_i)) \qquad (3.\ 25)$$

In this equation, the variable $\hat{y}$ represents the predicted value for design sample x, λ_i represents the weighted coefficient of the i^{th} basis function, r (x, x_i) represents the distance between sample x_i and the testing point x, and ϕ is the representation of the basis function.

If the predicted value of the objective function at the known sample points is the same as the real answer value at those points, then the condition is met, it is possible to interpolate the parameters λ_i, as given in Equation 3.26:

$$\hat{y} = y(x_i), i = 1,2,3, \dots \dots, n \qquad (3.\ 26)$$

The equation for the matrix that is obtained by combining Equations (3.25) and Equation (3.26) is given in Equation 3.27, as:

$$\phi\lambda = y \qquad (3.\ 27)$$

On the basis of Equation (3.27), it is possible to ascertain the value of λ, and the RBF surrogate model may be employed to can be used to guess what the values will be at each sample point in the design space as a whole.

3.4.3 Non-Dominated Sorting Genetic Algorithm II (NSGA-II)

Optimization is achieved through the use of the genetic algorithm, which is based on the process of biological evolution. During the optimization process, the information about the search space will be automatically amassed in order to produce the best possible solutions. Fast non-dominated sorting and crowded sorting are two new concepts that are introduced by the NSGA-II algorithm, which is an improvement on the previous NSGA algorithm. Each generation of the rapid non-dominated sorting method yields a collection of non-dominated solutions, also known as Pareto-optimal solutions from the previous generation. Following this, the computations for the subsequent iteration are carried out with the help of these solutions. A comparison of the surrounding individual density of a particular site in the population can be accomplished through the use of crowding sorting. It is possible to determine the largest cuboid that encompasses the individual by employing the technique of crowded sorting. When two individuals have differing non-dominated sorting, the one with the lower number is placed ahead of the other individual in the sorting process. The individual who has the smaller crowded distance is the one who is placed in front of the other individual if the two individuals have the same non-dominated sorting. Figure 3. 10 is a representation of the fundamental idea behind the NSGA-II algorithm [106,107].

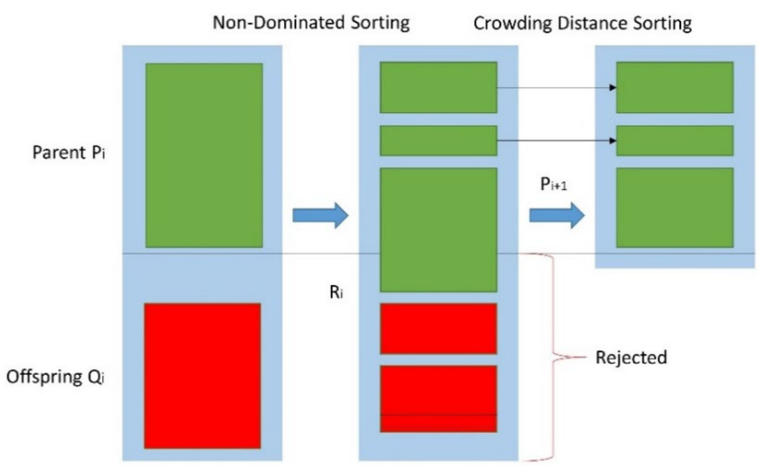

Figure 3. 10 The NSGA-II algorithm illustrated in an abstract form

Additional information regarding the NSGA-II algorithm can be found in other places. The flow chart diagram for the NSGA-II optimization process is shown in Figure 3. 11. In order to analyse the fitness functions (i.e., heat transfer factor and pressure drop factor), the data-driven surrogate model is utilized in conjunction with RBF.

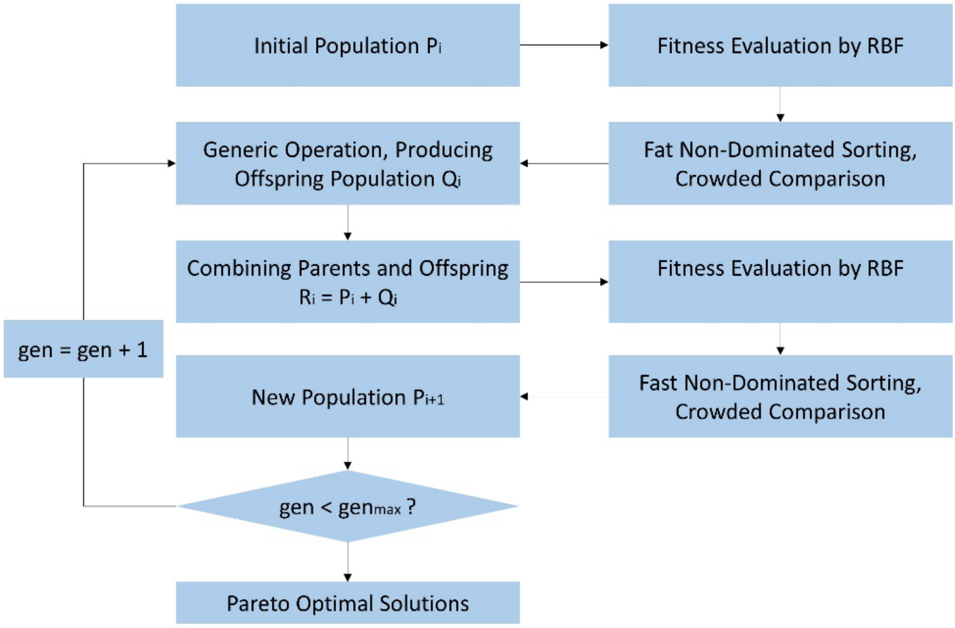

Figure 3. 11 Optimization flow diagram for NSGA-II in this study

Here are the primary procedures for optimizing utilizing the NSGA-II algorithm:

- Step 1: Take advantage of the real number in order to code the design variables and produce the initial population P_i, which consists of N population. The design variables will be used to construct the population matrix (F″).
- Step 2: Through the utilization of the data-driven surrogate model that is founded on RBF, evaluate the fitness functions (i.e., heat transfer factor and pressure drop factor) of each individual. The matrix F will be obtained.

- Step 3: Perform the quick non-dominated sorting method on the population in order to sort it. An additional sorting is performed using the crowded comparison operator on the individuals who have the same number after the rapid sorting process has been completed.
- Step 4: One can produce the offspring population by carrying out the genetic operations of matrix F, which include the election operation, the crossover operation, and the mutation operation.
- Step 5: The parent population and the offspring population are combined to produce a new population R_i that has a size of two times the size of the parent population.
- Step 6: When evaluating the fitness functions, the surrogate model should be utilized. The population should be sorted using the quick non-dominated sorting and crowded sorting techniques.
- Step 7: Acquire the subsequent population, P_{i+1} by keeping the N number of the population that is located in the centre of the population.

The Pareto optimum solutions can be produced by carrying out the processes described above until the iteration number approaches the maximum number of iterations that has been defined. The code for the same has been described in Appendix II. Several potential solutions will be considered by the designers in order to determine which one is the best option for a certain situation.

3.5 Summary

This chapter presents the detailed framework utilized for the mathematical modelling and research methods applied in the investigation of thermal hydraulics in Supercritical Water Reactors (SCWRs). The chapter is partitioned into two primary sections: The Computational Modelling and Optimization Framework provides a systematic framework for simulating and optimizing the thermal hydraulic performance of Supercritical Water-Cooled Reactors (SCWRs).

The computational model commences by meticulously preparing the geometric arrangement of the 2×2 rod bundle. This entails generating a comprehensive digital depiction of the bundle and wire spacers to precisely capture the intricate physical characteristics of the system. Mesh generation is essential for partitioning the geometry into finite components, hence allowing numerical analysis. An optimal mesh guarantees the precise resolution of the essential characteristics of fluid flow and heat transport, striking a balance between computational efficiency and simulation accuracy. The simulation depends on solving the Navier-Stokes equations for fluid flow, in addition to energy conservation equations, to accurately represent the intricate interactions of supercritical water within the rod bundle. The reactor environment is accurately modelled by applying appropriate boundary conditions, such as intake velocity and temperature, as well as operating conditions that reflect supercritical parameters. The work utilizes modern computational fluid dynamics (CFD) solvers, which apply finite volume approaches to discretize the governing equations. This allows for accurate forecasting of fluid flow and heat patterns under different circumstances. Verification of the computational fluid dynamics (CFD) model guarantees that the numerical implementation faithfully reflects the theoretical model. A grid independence test verifies that the simulation results are unaffected by changes in the mesh size, thus confirming the resilience and reliability of the findings. Validation entails the comparison of simulation findings with experimental data or established benchmarks to verify the accuracy of the model in predicting real-world phenomena, hence proving its credibility and applicability.

Data preparation involves gathering and preparing pertinent variables and parameters needed for the optimization process, assuring a comprehensive dataset for efficient analysis. The Taguchi method is utilized to achieve resilient design by determining the most favourable operating circumstances that result in lowest variability. This approach enhances the reliability and performance of the SCWR system. The Taguchi method was employed to strategically choose sampling locations, guaranteeing rapid and effective exploration of the parameter space without the necessity for lengthy simulations. Radial Basis Function (RBF) surrogate modelling is employed to generate a concise depiction of the Computational Fluid Dynamics (CFD) outcomes, enabling quick assessment of various design scenarios and streamlining the optimization process. The NSGA-II algorithm is utilized for multi-objective optimization, effectively managing conflicting objectives such as decreasing pressure drop factor and optimizing heat transfer factor. This evolutionary method facilitates the identification of optimal solutions that effectively balance multiple performance criteria.

This chapter presents a comprehensive approach for modelling and optimizing the thermal hydraulics of Supercritical Water-cooled Reactors (SCWRs).

By combining CFD models with modern optimization techniques, the process of exploring design changes for boosting reactor performance and safety becomes comprehensive and efficient, providing significant insights. The integration of rigorous computational approaches and robust optimization frameworks is crucial for the advancement of nuclear reactor technology.

Chapter 4

Thermal Hydraulics of 2×2 Rod Bundles with A Wire Spacer at Supercritical Condition

4.1 Introduction

As the search for more advanced designs for nuclear reactors continues, the Supercritical Water Reactor (SCWR) stands out as a candidate because of its potential to improve thermal efficiency and minimize the complexity of operational procedures. The fuel assembly, which is characteristic of SCWRs and normally comprises of rod bundles, is an essential component. In order to ensure the overall performance and safety of the reactor, it is essential to consider the thermal hydraulics of these rod bundles, particularly when they are running under supercritical circumstances. The primary focus of this chapter is on the thermal hydraulic behaviour of 2×2 rod bundles with wire spacers when subjected to supercritical conditions. It offers a comprehensive analysis of the heat transfer and fluid flow properties that are essential for the efficient operation of the reactor.

Although the 2×2 rod bundle structure is less complicated than full-scale assemblies, it provides substantial insights into the thermal hydraulic phenomena that are encountered in SCWRs. The employment of wire spacers in these bundles serves various goals, including the maintenance of the distance between fuel rods, the enhancement of turbulence, and the promotion of more uniform coolant flow distribution. For the purpose of improving the design and ensuring the safe and effective removal of heat from the fuel rods, it is vital to have a solid understanding of the impact that wire spacers have on the thermal and hydraulic performance of the bundles.

Within the realm of thermal hydraulics, operating in supercritical circumstances presents a one-of-a-kind set of problems and opportunities. Supercritical water, in contrast to ordinary subcritical coolants, does not go through the process of phase change. This makes the process of heat transfer easier to undertake, but it also necessitates careful control and a grasp of the thermo-physical properties of the water. The density, specific heat, and thermal conductivity of water undergo considerable changes when it is subjected to supercritical pressures and temperatures. These changes, in turn, have an impact on the heat transfer coefficients and flow dynamics that occur within the rod bundle.

The purpose of this chapter is to investigate the intricate relationship that exists between these parameters in a rod bundle arrangement that is 2×2. A mix of experimental data, computational fluid dynamics (CFD) simulations, and analytical modelling will be utilized in order to study the impact that wire spacers have on the rates of heat transfer, pressure drops, and flow distribution when supercritical conditions are present. In addition to this, the thermal hydraulic stability of the system has been investigated and fundamental characteristics has been determined that have an impact on its performance. These parameters include the rod-to-rod spacing, the wire spacer design, and the operating circumstances.

The purpose of this chapter is to contribute to a more profound knowledge of the fundamental mechanisms that govern heat transfer and fluid flow in SCWRs. This is accomplished by providing a comprehensive examination of the thermal hydraulics characteristics of 2×2 rod bundles that contain wire spacers wrapped clockwise and anti-clockwise direction. It is anticipated that the data provided here will provide information that will be used to inform the design and optimization of bigger fuel assemblies, which will ultimately help the development of SCWR technologies that are more efficient and reliable.

4.2 Variation of Axial Velocity

The purpose of this study was to conduct a comprehensive investigation into the distribution of axial velocity within a 2×2 rod bundle that was supplied with wire spacers. The investigation focused on two separate spacer arrangements, namely clockwise and mixed designs of clock and anti-clock wire wrappings. Using sophisticated Computational Fluid Dynamics (CFD) simulations, the research was carried out under supercritical conditions. The results of the analysis provided insights into the velocity patterns, which are essential for comprehending flow dynamics and optimizing reactor design.

4.2.1 Clockwise Arrangement

The contours of the axial velocity demonstrate a specific distribution for the clockwise arrangement, which is characterized by the wire spacers being arranged in a helical manner around the fuel rods. The findings indicate that the axial velocity is at its highest point (23 m/s - 25 m/s) in the middle of the plane, resulting in the formation of a prominent peak that extends down the central axis of the rod bundle.

This velocity peak is indicative of a powerful and concentrated flow channel in the core of the bundle. This flow channel can be attributed to the helical twisting effect carried out by the wire spacers, which direct the flow of coolant inward towards the centre of the bundle.

This pattern in Figure 4. 1, is depicted very clearly in the velocity contour maps, which show that the highest velocity values are concentrated in a small, cylindrical zone that runs down the axial length of the bundle. The flow structure is clearly defined and centralized, as seen by the steady decrease in velocity that occurs towards the periphery. This centralized flow has the potential to improve the effectiveness of convective heat transfer in the centre region; however, if it is not handled effectively, it may result in less effective cooling near the bundle walls.

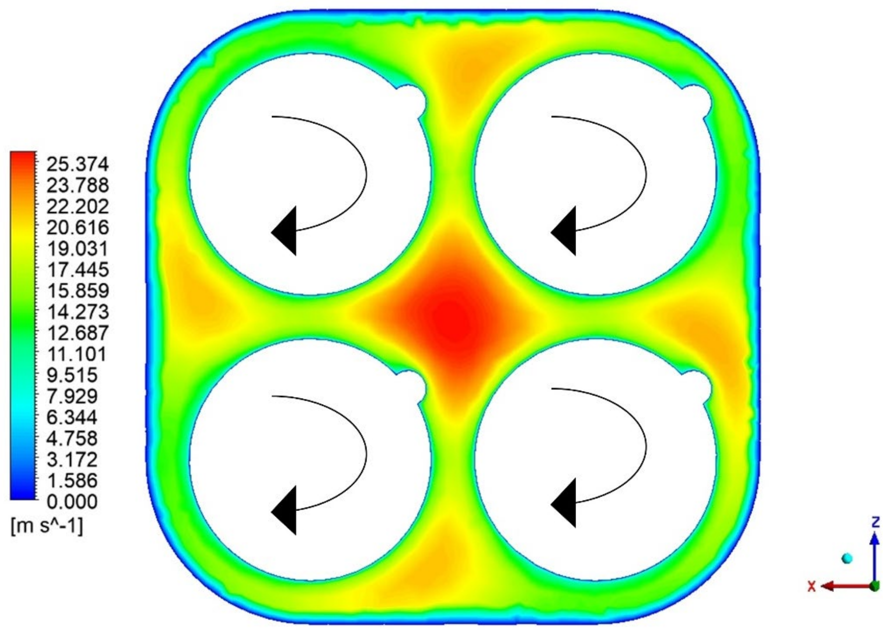

Figure 4. 1 Axial Velocity Field of Conventional Helical Wire

4.2.2 Mixed Arrangement (Clock and Anti-Clock Wrappings)

The mixed design, on the other hand, displays a more complicated velocity distribution since it mixes a variety of spacer orientations around the fuel rods. There are two separate zones of high velocity (22 m/s – 23 m/s) that are indicated by the axial velocity contours for this arrangement. One of these regions is located in the middle of the plane, and the other is placed closer to one side of the plane.

The contour maps of Figure 4. 2, show that the maximal axial velocity is not only along the centre axis of the rod bundle, but it also extends towards one side of the rod bundle. It is likely that the variable spacer orientations that interrupt the uniform flow and cause localized zones of acceleration are responsible for the construction of an asymmetric flow channel, which is suggested by the presence of this secondary high-velocity region's presence. This mixed flow pattern has the potential to result in a better distributed cooling impact across the rod bundle. This might potentially improve the overall thermal performance of the rod bundle by minimizing hot spots and promoting a more even temperature distribution.

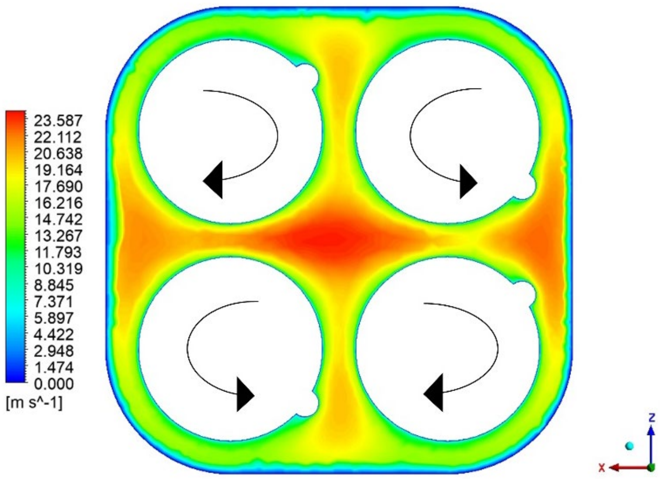

Figure 4. 2 Axial Velocity Field of New Pattern Helical Wire

There is a peak in the centre of the velocity profiles, which is comparable to the arrangement that is clockwise; however, there is an additional high-velocity zone on one side, which results in a flow field that is more complicated. In particular, this asymmetry in the flow can promote heat transfer from the fuel rods and enhance the mixing of the coolant. This is especially true in regions where the coolant velocity is otherwise lower.

4.2.3 Comparative Analysis

When comparing the two different spacer configurations as shown in Figure 4. 1 and Figure 4. 2, the clockwise layout produces a flow pattern that is more cantered and focused, whereas the mixed arrangement encourages a cooling technique that is more diffused and has the potential to be more successful. Considering that the mixed arrangement has the capability of producing many high-velocity zones, it is possible that it has a greater potential for lowering temperature gradients inside the rod bundle, which would result in an increase in the overall rate of heat removal.

A summary of the findings from the computational fluid dynamics (CFD) examination of the axial velocity contours is that the arrangement of wire spacers has a substantial impact on the flow dynamics that occur within the rod bundle. The mixed design supports a more complicated and diffused flow, which may give potential benefits in terms of heat management and reactor performance. The clockwise arrangement, on the other hand, creates a central flow channel that is concentrated. These findings offer useful insights that may be used to optimize the design of wire spacers in SCWR fuel assemblies in order to achieve higher thermal hydraulic performance.

4.3 Variation of Velocity Vectors

In this particular section, the outcomes of the velocity vector contours for two various wire spacer patterns that are contained within a 2×2 rod bundle has been presented. These patterns are the typical helical wire wrap and a new configuration geometry. In addition to shedding light on the implications of spacer design on the behaviour of coolant flow under supercritical circumstances, these simulations offer crucial insights into the flow characteristics and distribution throughout the bundle.

4.3.1 Conventional Helical Wire Wrap (Clockwise)

The contour plots for the traditional helical wire wrap show that there is a significant swirl flow pattern along the region of the duct wall. This prominent swirling effect is a direct consequence of the helical wire spacer, which creates a strong rotating motion in the fluid which flows down the axial length of the rod bundle. This motion causes the fluid to flow in a spiralling motion. The flow exhibits a spiralling direction that focuses around the edge of the duct, as shown by the velocity vectors in Figure 4. 3, which provide a clear illustration of this swirling motion.

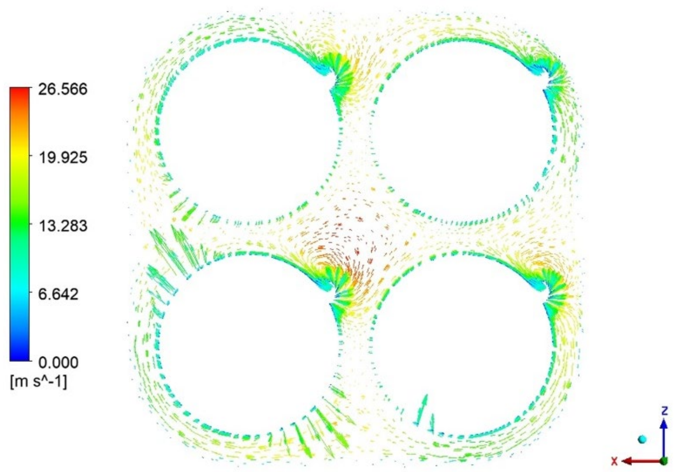

Figure 4. 3 Velocity Vector of Conventional Helical Wire

An insignificant portion of the coolant flow is able to pass through the tiny crevices that exist between the fuel pins, despite the fact that there is a large amount of swirl. Low flow velocities are indicated by the velocity vectors that are contained within these inter-pin gaps, which suggests that there is restricted fluid motion and the possibility of stagnation zones. This restricted flow can result in uneven cooling, with a greater possibility of hot spots appearing on the rod surfaces that are less exposed to the circulating coolant. This can reduce the rate of the cooling process.

4.3.2 New-Pattern Geometry (Clock and Anti-Clock Wrappings)

On the other hand, the new-pattern geometry, which was intended to enhance flow dispersion, displayed a velocity vector profile that was noticeably distinct from the previous one. According to the contour plots for this arrangement, the flow rate is dispersed in a manner that is consistent across all of the sub-channels that are contained within the rod bundle. The velocity vectors, which are depicted in Figure 4.4, demonstrate a flow pattern that is more balanced, with a sizeable percentage of the coolant traveling across the distances that exist between the pins.

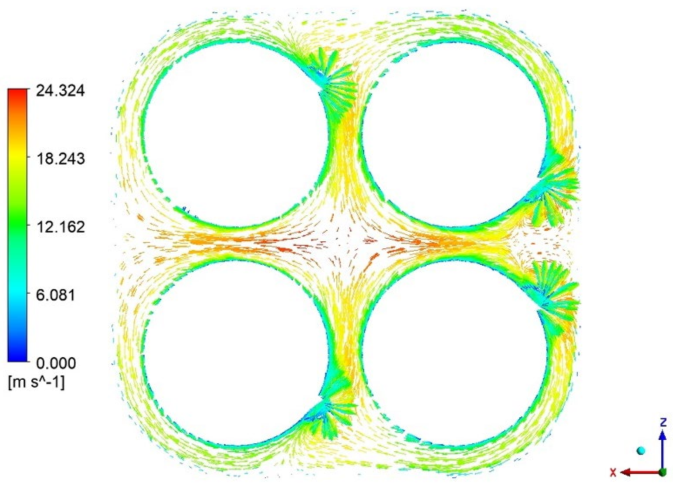

Figure 4. 4 Velocity Vector of New Pattern Helical Wire

The shape of the new pattern encourages a more effective usage of the flow area that is available, which in turn ensures that coolant is able to reach all of the rod bundle's components in a more efficient manner. These velocity vectors, which suggest consistent and relatively high velocities through the tight spaces between the fuel rods, are evidence of the enhanced flow distribution that has been achieved. By ensuring that the rod surfaces are cooled more consistently, this configuration enhances the total heat transfer performance and reduces the likelihood of stagnation, in contrast to the standard helical wrap.

4.3.3 Comparative Analysis

When the two configurations are compared as can be seen in Figure 4. 3 and Figure 4. 4, the typical helical wire wrap produces a powerful swirl that is primarily restricted to the outer areas of the duct. This restricts the rate of the helical wire wrap in cooling the middle sections of the rod bundle. The new-pattern geometry, on the other hand, is able to create a more even flow distribution and greatly improve the flow through the inter-pin gaps. This is an essential feature for ensuring that temperature profiles remain uniform and minimizing localized overheating.

It is clear from the velocity vector contours that the new-pattern geometry has a number of benefits, including improved coolant distribution and enhanced thermal performance. Through the facilitation of a faster flow rate through the short inter-pin gaps, this design reduces the likelihood of hot spots forming and enables more efficient heat evacuation from the fuel rods.

In conclusion, the findings of the velocity vector contours shed light on the significance of wire spacer design in terms of its ability to influence the behaviour of coolant flow within a rod bundle environment. In comparison to the traditional helical wire wrap, the new-pattern geometry displays a superior ability to distribute flow uniformly and assure adequate cooling across all sub-channels. This represents a considerable improvement over the previous method. In order to improve the thermal hydraulic performance of advanced nuclear reactor systems, these findings offer useful insights that may be used to optimize the designs of wire spacers.

4.4 Pressure Distribution

The findings for both the traditional helical wire wrap and a new-pattern geometry are presented in this part. The emphasis is placed on the manner in which these two types of wire wraps influence the pressure distribution and pressure drop across the bundle.

4.4.1 Conventional Helical Wire Wrap

There is a significant amount of pressure variation throughout the rod bundle, as seen by the static pressure contours for the standard helical wire wrap. As can be seen in Figure 4. 5, the existence of the wire wrap has a considerable impact on the static pressure, which results in fluctuations that are proportional to the positions of the wire spacers.

The presence of these spacers causes localized changes in the cross-sectional area of the sub-channel, which in turn causes differences in pressure as the coolant travels along the winding routes that are generated by the helical wraps as shown in Figure 4. 6 the cross-sectional pressure difference is (1059.093 Pa).

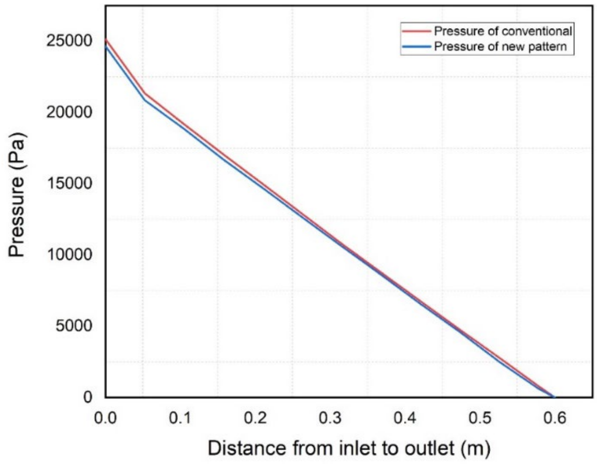

Figure 4. 5 Static pressure at the domain's centre

A significant disadvantage of the typical helical wire wrap is the greater pressure drop that it causes throughout the rod bundle. This is one of the key downsides. Because of the wire wrap, there is an increase in the frictional resistance and flow disruption, which ultimately results in a greater drop in static pressure from the rod bundle's intake to its outlet. This increased pressure drop can have a detrimental impact on the rate of the cooling process. Furthermore, it requires more energy to maintain the optimum coolant flow rate, which makes it less desirable for efficient reactor operation.

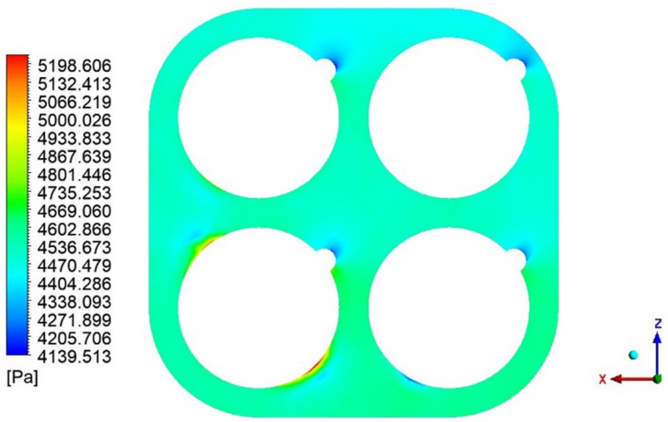

Figure 4. 6 Pressure Field Distribution of Conventional Helical Wire

4.4.2 New-Pattern Geometry

On the other hand, the new-pattern geometry, which has a decrease in the number of wire spacers at the centre of the domain and a design that is more streamlined, leads to a different distribution of static pressure difference (407.407 Pa). The contours of the static pressure for this structure exhibit a pressure profile that is more uniform as shown in Figure 4. 7, with much reduced variability in the axial direction. The revised pattern, which is represented in Figure 4. 5, demonstrates a reduced pressure differential from the inlet to the outlet, which highlights a flow that is smoother and more efficient through the rod bundle.

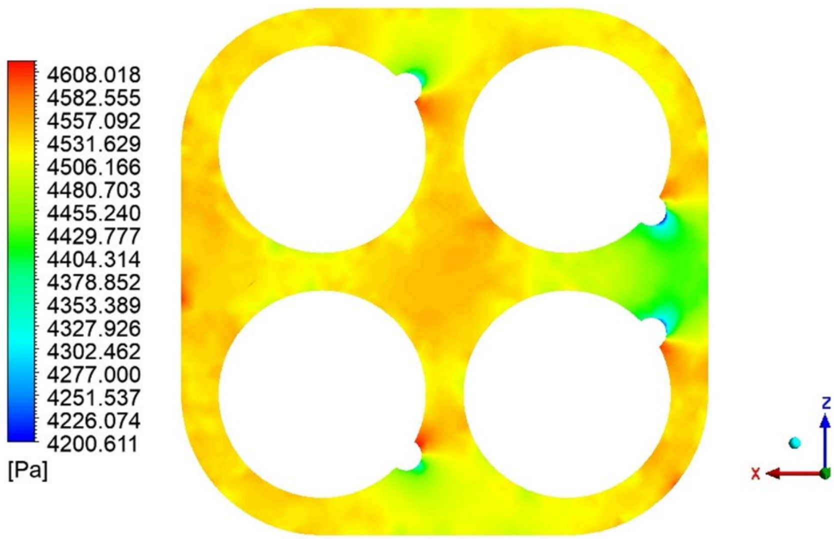

Figure 4. 7 Pressure Field Distribution of New Pattern Helical Wire

The new-pattern geometry has a number of advantages, the most important of which is that it has a little impact on the cross-sectional area of the sub-channel. This, in turn, decreases the pressure variations that are brought about by changes in the flow area. The new design has a reduced number of wire spacers, which minimizes the amount of flow obstruction and makes it possible to have a flow route that is more continuous and less turbulent. Therefore, the pressure drop across the rod bundle is greatly decreased, with a measured reduction of 1.9% in the axial pressure fluctuation at the centre of the domain in comparison to the typical helical wrap design. This is a major improvement.

4.4.3 Comparative Analysis

The advantages of the new-pattern geometry in terms of pressure management are brought to light by the comparative study of static pressure contours between the traditional helical wire wrap and the new-pattern geometry as can be seen in Figure 4. 5, Figure 4. 6 and Figure 4. 7. Due to the frequent changes in sub-channel area that are generated by the helical wraps, the traditional design suffers from increased pressure drop and higher pressure variations. This is despite the fact that it is effective in causing swirl and increasing heat transfer.

On the other hand, the new-pattern geometry provides a flow channel that is more optimal, with less flow resistance and a more uniform amount of pressure distribution. Because of the lower pressure drop that was seen in this design, the flow rate was increased, and the potential for lower operational costs was reduced. This is because the amount of energy required to pump the coolant through the rod bundle was reduced.

In conclusion, the findings of the static pressure contour highlight the significance of wire spacer design in terms of its ability to influence the hydraulic performance of rod bundles through its influence. The new-pattern geometry exhibits a distinct benefit in lowering pressure variations and minimizing pressure drop, which ultimately results in an improvement in the overall rate and effectiveness of the cooling system in advanced nuclear reactors. The findings of this study offer useful insights that may be utilized in the design and optimization of wire spacers for the purpose of achieving higher thermal hydraulic performance and energy efficiency in reactor installations.

4.5 Temperature Distribution

In the investigation of temperature distribution contours within a 2×2 rod bundle under supercritical conditions, it is observed that there are substantial discrepancies between the typical helical wire wrap and the new-pattern geometry. In addition to providing useful insights for the development of Supercritical Water Reactor (SCWR) designs, these variations bring to light the impact that wire spacer design has on thermal performance and safety.

4.5.1 Conventional Helical Wire Wrap

The contours of the temperature distribution for the standard helical wire wrap are shown in Figure 4. 8. These contours show that higher temperature zones (479°C – 479.4°C) are clustered in the side of the rod bundle, with the highest temperatures occurring close to the fuel rods. The helical wrap generates a whirling flow that, while it does improve mixing to some degree, leads to a distribution of coolant that is less uniform and temperature gradients that are more noticeable. This unequal flow distribution results in hot spots, which are areas with high temperatures that are confined and have the ability to raise the risk of material degradation or thermal stressors to the material.

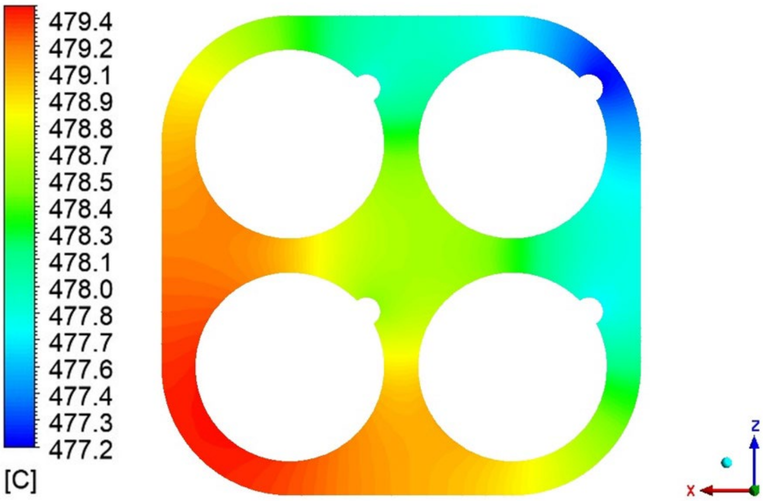

Figure 4. 8 Temperature Field Distribution of Conventional Helical Wire

It is difficult to achieve the appropriate level of thermal efficiency and safety margins when using the traditional design because the greatest temperature that can be measured at the section plane is substantially higher than it does in the typical design. Although the swirling motion of the coolant is beneficial in certain parts, it does not sufficiently disperse the coolant across all of the sub-channels. This results in poor cooling in certain areas, which in turn leads to higher rod temperatures overall.

4.5.2 New-Pattern Geometry

On the other hand, the temperature contours for the new-pattern geometry as shown in Figure 4. 9, demonstrate a temperature distribution that is more consistent across the rod bundle (470.7°C – 470.8°C). The exceptional capacity of the new design to keep temperatures at lower levels and to prevent localized overheating is highlighted by the fact that it brings the maximum temperature down by 8.5°C at the section plane. This decrease in peak temperature is a direct result of the enhanced flow distribution that is made possible by the new-pattern wire spacer. This spacer makes it possible for a flow that is more balanced and spread over all of the sub-channels.

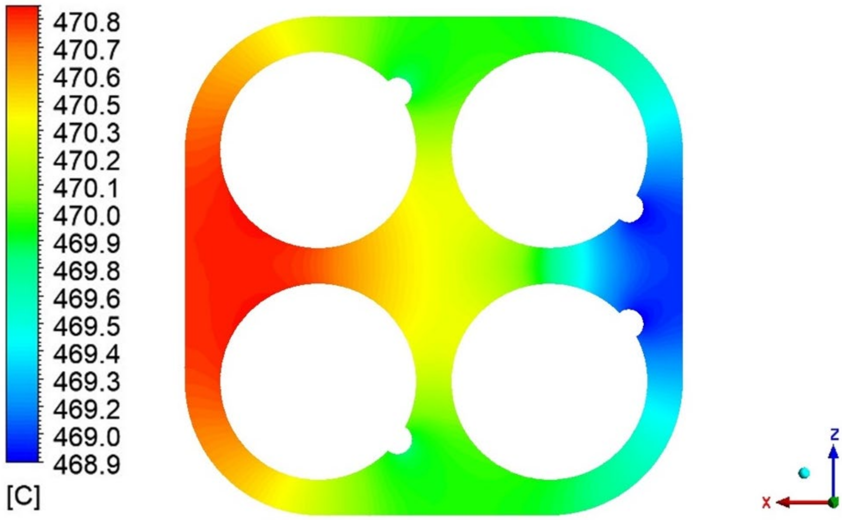

Figure 4. 9 Temperature Field Distribution of New Pattern Helical Wire

The examination of the velocity vector in tangential projection for the new-pattern geometry reveals that the flow rate is higher and more uniformly distributed across the various sub-channels. Within the rod bundle, temperature gradients are reduced as a result of this improved flow homogeneity, which also promotes more efficient heat transfer. The lower temperatures that were recorded in the revised pattern are indicative of improved thermal management, which in turn reduces the likelihood of high-temperature hotspots and improves the overall thermal hydraulic performance under typical operating settings.

As can be seen in Figure 4. 10, the overall rod temperature in the new-pattern geometry is consistently lower than the temperature in the conventional design. Since the temperature of the rod has been significantly lowered, the possibility of accidents that are caused by rod melting or thermally-induced failures has been significantly reduced. By ensuring that coolant is dispersed evenly and that heat is efficiently evacuated from all sections of the rod bundle, the new-pattern wire spacer is able to effectively control the temperature rise that is occurring.

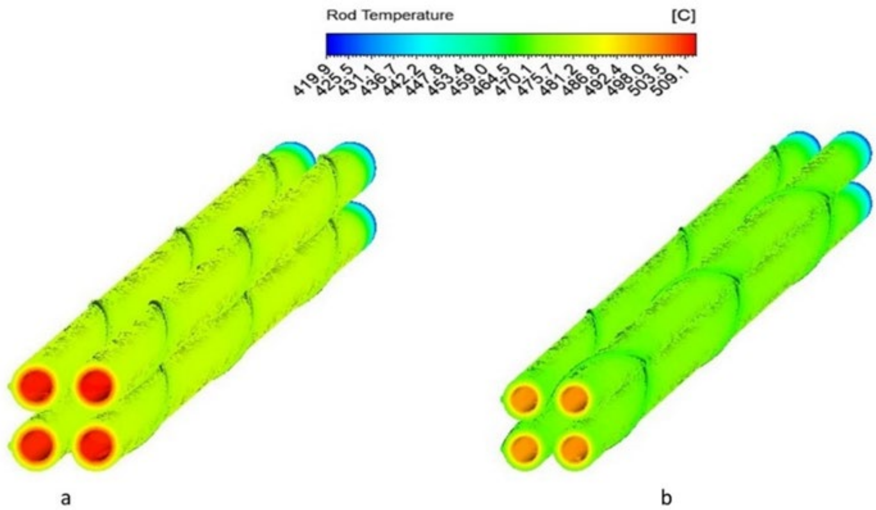

Figure 4. 10 Overall Temperature of Wired Rods (a) Conventional (b) New Pattern

4.5.3 Comparative Analysis

The advantages of the new-pattern geometry in terms of thermal performance are highlighted by the comparative analysis of temperature distribution between the traditional helical wire wrap and the new-pattern geometry as cab seen in Figure 4. 8, Figure 4. 9 and Figure 4. 10. In spite of the fact that the conventional design is capable of producing effective swirl, it is unable to disperse coolant in an adequate manner over all of the sub-channels. This leads to greater localized temperatures and thus increased thermal dangers.

On the other hand, the new-pattern geometry exhibits a glaring improvement in terms of temperature management. The success of the new design in improving thermal hydraulic performance is highlighted by the fact that the maximum temperature has been lowered and the temperature distribution across the rod bundle has become more uniform. Under typical operating conditions, the new pattern's lower total rod temperatures reduce the danger of material deterioration and enhance the reactor's safety and dependability. Furthermore, the new design reduces the risk of material degradation.

To summarize, the findings of the temperature distribution contour reveal that the new-pattern geometry offers considerable advantages over the typical helical wire wrap in terms of having the ability to maintain lower temperatures and to promote uniform cooling. In order to achieve better thermal management, greater safety, and higher operational efficiency, our findings lend support to the implementation of new-pattern wire spacers in SCWR designs.

4.6 Comparison of Straight Wire and Helical Wire Spacer

Wire spacers are an essential component in nuclear reactors, as they preserve the geometry of rod bundles and ensure that they continue to function effectively. It is important to note that the thermal hydraulic characteristics of the system are greatly influenced by the choice between helical and straight wire spacers. In spite of the fact that helical wire spacers have a greater pressure drop than straight wire spacers, the benefits that they offer in terms of improving flow dynamics and heat transfer are significant. Despite the fact that they have a higher pressure drop, the following are the primary advantages of utilizing helical wire spacers:

4.6.1 Enhanced Swirl and Turbulence

In order to create a swirling motion in the flow of coolant, helical wire spacers are planned to be constructed. This swirl increases the turbulence as shown in Figure 4. 11, which in turn improves the efficiency with which the coolant is mixed. The turbulent flow that is produced as a result causes the thermal boundary layers that are formed around the fuel rods to be disrupted. This results in an increase in the heat transfer coefficient and an improvement in the reactor's overall thermal efficiency. In supercritical water reactors (SCWRs), where effective heat transfer is essential to the maintenance of safe and stable operating conditions, this is an especially desirable feature.

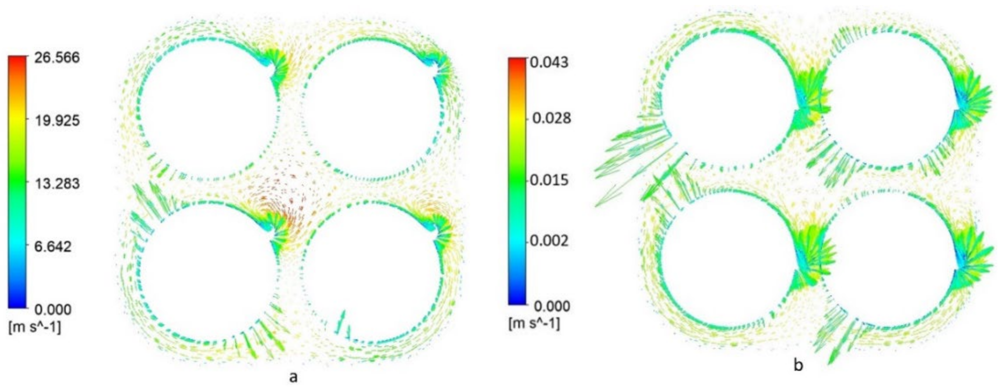

Figure 4. 11 Velocity Vector of (a) Helical Wire (b) Straight Wire

4.6.2 Improved Coolant Distribution

The helical arrangement ensures that the coolant is distributed more evenly across the rod bundle to achieve the desired effect. The spacers help in uniformly distributing the coolant around each rod by guiding the flow in a helical direction. This reduces the probability of hot spots emerging, which are areas of the fuel rod that experience localized high temperatures that could result in damage or failure of the fuel rod. This uniform distribution is essential for ensuring that the fuel rods remain intact during the period of time that the reactor is in operation and for maintaining a consistent level of cooling.

4.6.3 Reduced Risk of Flow Stagnation

Straight wire spacers have a tendency to channel the flow of coolant in a linear fashion, which can lead to regions of sluggish flow. This is especially detrimental in the areas where there are narrow spaces between fuel rods. It is possible for these stagnant zones to result in insufficient cooling, which in turn raises the risk of overheating. Helical wire spacers, on the other hand, encourage the continual circulation of the coolant around the rods. This reduces the likelihood of stagnant flow zones occurring and ensures that the rod bundle as a whole receives more consistent cooling.

4.6.4 Enhanced Structural Integrity

Even when subjected to high flow rates and turbulent circumstances, the fuel rods are able to maintain their position and spacing because to the helical arrangement of the wire spacers, which offers improved mechanical support for the fuel rods. This structural stability is critically important for preventing rod displacement and preserving the precise geometry that is necessary for the reactor to work at its highest possible level. Helical wires, which offer supplementary support, are a significant contributor to the overall longevity and safety of the reactor system.

4.6.5 Better Performance in Supercritical Conditions

In SCWRs, where the coolant runs at supercritical pressures and temperatures, the characteristics of the fluid undergo considerable changes, which makes the importance of effective heat transfer even more crucial. In spite of the challenging conditions of a supercritical reactor, the heightened turbulence and mixing that are created by helical wire spacers guarantee that the coolant will continue to be dispersed evenly and that heat will be adequately evacuated from the fuel rods. This results in a steadier operation of the reactor as well as an improvement in thermal efficiency.

Despite the fact that helical wire spacers have a higher pressure drop than straight wire spacers (4669 Pa versus 4235 Pa) as shown in Figure 4. 12, the advantages that they offer in terms of enhanced swirl, improved coolant distribution, reduced danger of flow stagnation, and improved structural integrity more than make up for the disadvantages. Helical wire spacers are a preferred choice in many nuclear reactor applications due to their improved thermal performance and increased safety margins. This is especially true for SCWRs, which are of the utmost importance in terms of efficient heat transfer and steady coolant flow.

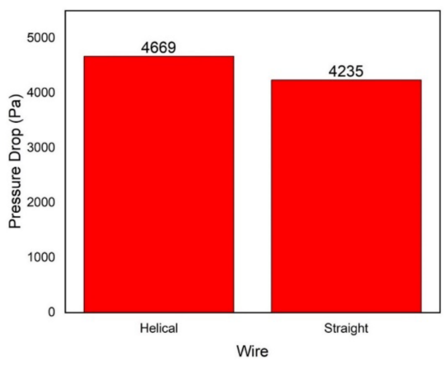

Figure 4. 12 Pressure Drop of Helical and Straight Wire Wrapped Rod Bundle

4.7 Summary

The purpose of this analysis was to examine the thermal hydraulic performance of 2×2 rod bundles under supercritical conditions. As part of this investigation, a comparison was made between standard helical wire wraps and a contemporary pattern geometry. The axial velocity, velocity vectors, static pressure, and temperature distribution are the primary areas of importance in this examination.

As a result of the helical design that directs coolant flow into the middle of the rod bundle, the standard helical wire wrap configuration generates a significant peak in axial velocity at the plane's centre. This peak is driven by the helical design. On the other hand, the new-pattern geometry, which features a spacer orientation that is variable, exhibits a velocity profile that is more complicated and contains many zones of high velocity. This results in an improved distribution of coolant and a more equal cooling across the bundle. This results are an increase in the effectiveness of convective heat movement and a decrease in the risk of any one area becoming overheated.

As a consequence of the helical spacers, velocity vector analysis reveals that there is a significant swirl flow along the duct walls in the traditional configuration. While this swirl flow does increase turbulence along the walls, it does not efficiently penetrate the thin inter-pin gaps. As a result, there is a possibility of stagnation zones and ineffective cooling. The new-pattern geometry, on the other hand, makes it possible to achieve a flow that is more consistently distributed, with higher tangential velocities over all of the sub-channel structures. As a result of this improved flow distribution, stagnant zones are reduced, and heat disposal is enhanced, which contributes to a lower overall temperature and higher thermal performance.

Because of the numerous changes in sub-channel areas that are caused by the helical wraps, static pressure contours reveal considerable pressure variations and a larger pressure drop in the standard helical wrap design. This indicates that the pressure drop is higher. These changes contribute to an increase in the amount of energy that is required to sustain the flow of coolant. The new pattern geometry, which has fewer wire spacers and a design that is more streamlined, has a more uniform pressure distribution and a pressure drop that is roughly 1.9% lower than the previous architecture. As a consequence, this leads to decreased amount of operational pressure and increased heat transfer rate.

The typical arrangement has greater peak temperatures, with hot spots concentrated near the fuel rods, as seen by the temperature contours obtained from the configuration. As a result of its efficient flow management and balanced cooling, the new-pattern geometry, on the other hand, exhibits a more even distribution of temperatures and a maximum temperature drop of 8.5 degrees Celsius. The overall safety of the reactor is improved as a result of this design, which reduces the likelihood of localized overheating.

When compared to the traditional helical wrap, the new-pattern geometry provides considerable gains in terms of axial velocity distribution, flow homogeneity, pressure control, and temperature regulation. In conclusion, the new-pattern geometry offers significant advantages. In Supercritical Water Reactors (SCWRs), these modifications result in improved thermal hydraulic performance, higher heat transfer rate, and enhanced safety. Overall, these improvements are beneficial.

Also a comparison of helical wire with straight wire was done. The advantages that the helical wire offer in terms of enhanced swirl, improved coolant distribution, reduced danger of flow stagnation, and improved structural integrity more than make up for the disadvantages of pressure loss.

Chapter 5

Design Optimization of Wire Spacer for 2×2 Rod Bundles

5.1 Introduction

The configuration of wire spacers in rod bundles plays a crucial role in influencing the thermal hydraulic efficiency of Supercritical Water Reactors (SCWRs). These spacers serve the purpose of preserving the structural integrity of the fuel rods and also have a crucial function in improving heat transfer, reducing pressure drops, and guaranteeing an even distribution of coolant. This chapter explores the optimization of wire spacers for 2×2 rod bundles, with a specific focus on the creative application of Radial Basis Function (RBF) surrogate models and the Non-Dominated Sorting Genetic Algorithm II (NSGA-II) to attain an optimal design that satisfies numerous performance criteria.

Wire spacers are essential elements in the construction of nuclear fuel assemblies. They guarantee optimal spacing between fuel rods, which is crucial for maintaining sufficient coolant flow and preventing the formation of localized areas of high temperature that might result in fuel rod malfunction. Within the framework of SCWRs, the configuration of these spacers assumes greater significance owing to the distinctive difficulties presented by the supercritical water surroundings, including elevated temperatures, pressures, and the requirement for effective heat transfer.

The traditional configuration of wire spacers, commonly utilizing helical wraps, has shown to be somewhat successful. Nevertheless, this frequently leads to an uneven distribution of flow and an elevated pressure drop, thereby jeopardizing the efficiency and safety of the reactor. Therefore, it is crucial to develop a spacer design that is more optimal in order to evenly distribute the coolant, improve thermal efficiency, and reduce pressure loss.

Conventional design methods frequently depend on trial and error or gradual enhancements, which are insufficiently efficient or thorough to explore the extensive design possibilities and choose the most ideal configurations. In order to surpass these constraints, sophisticated computational methods are utilized. The integration of RBF surrogate modelling and NSGA-II provides a robust framework for design optimization, enabling the effective examination of design alternatives and the determination of optimal solutions that effectively balance various conflicting objectives.

The RBF surrogate model is a mathematical method used to estimate complex computational simulations. It offers a rapid and precise method for forecasting system reactions to different design modifications without requiring exhaustive CFD simulations. This surrogate model is especially valuable for minimizing the computing load and enabling swift assessment of various design options.

NSGA-II is a multi-objective evolutionary algorithm known for its ability to efficiently identify Pareto-optimal solutions. These solutions represent the optimal balance between competing objectives, such as decreasing pressure drop and maximizing heat transfer rate. This technique is very suitable for effectively managing the intricate and non-linear characteristics of spacer design optimization. It allows for the identification of designs that provide substantial enhancements in performance.

The main goal of this chapter is to showcase the use of RBF surrogate modelling and NSGA-II in the optimization of wire spacers for 2×2 rod bundles. The chapter provides a detailed explanation of the procedure for developing a surrogate model, choosing sampling locations, and performing multi-objective optimization to determine the most effective spacer configurations. The results demonstrate the enhancements in thermal hydraulic performance attained through this sophisticated optimization approach.

This chapter attempts to offer a thorough guide on optimizing wire spacer designs by utilizing advanced computational approaches. The ultimate goal is to enhance the efficiency and safety of nuclear reactors. The knowledge acquired from this study can be utilized to improve the design of SCWRs and other advanced reactor systems, guaranteeing that they fulfil the rigorous requirements of contemporary nuclear energy production.

5.2 Dual – Objective Optimization of Wire Spacers

When it comes to improving the performance of nuclear reactors, optimizing the design of wire spacers in fuel assemblies is absolutely necessary. Achieving this optimization requires striking a balance between the rate of heat transfer and the reduction of pressure drop. The ratio of wire-wrap pitch to fuel rod diameter (H/D) and the ratio of wire-spacer diameter to fuel rod diameter (D_W/D) are two dimensionless design factors that are taken into consideration in this context. Discovering the best arrangement that maximizes heat transfer, thus minimizes the surface temperature of the rod while simultaneously minimizes friction factor (pressure drop) is the objective of this endeavour.

Both the length of the helical path and the distance between the coils are affected by H/D, which in turn has an effect on the swirl and turbulence that occurs in the coolant flow that occurs. D_W/D has an effect on the cross-sectional area that is accessible for coolant flow as well as the degree of blockage, which in turn has an effect on the overall pressure drop and flow distribution.

O_T, which stands for the objective function linked to heat transfer, is defined as the highest temperature divided by the area-averaged surface temperature of the fuel rod. Through the use of this measurement, one can gain an understanding of the uniformity of heat removal as well as the effectiveness of the cooling process. Values that are lower suggest that the heat transfer performance is better. The second objective function is concerned with pressure drop factor (O_P), which is an essential component in ensuring that coolant circulation is carried out in an effective manner. To reduce the amount of pumping power that is required and to increase the overall efficiency of the system, it is desirable to have lower pressure dips.

It was decided to use the L_{16} orthogonal array method in order to create sample points in order to achieve the dual goals of enhancing heat transfer and reducing pressure loss. With this strategy, which involves two factors (design variables) and four levels (various configurations), it is possible to assure a systematic investigation of the design space while simultaneously reducing the number of simulations that are being performed.

For the purpose of training a Radial Basis Function (RBF) surrogate model, the sample points that were generated were utilized. A computationally efficient method for evaluating various design configurations is provided by the RBF model, which provides an approximation of the complicated relationship that exists between the design variables and the objective functions.

In order to achieve optimal performance in the wire spacer design, the Non-Dominated Sorting Genetic Algorithm II (NSGA-II) was utilized once the surrogate modelling procedures were completed. Heat transfer factor and pressure drop factor are two competing objectives, and NSGA-II is a strong multi-objective optimization technique that identifies Pareto-optimal solutions. These solutions represent the optimum trade-offs between the two competing objectives.

A collection of Pareto-optimal solutions was produced as a result of the optimization process that utilized NSGA-II. Each of these solutions represents a different equilibrium between the objective functions. In order to improve heat transfer rate while simultaneously minimizing pressure drop, these solutions provide a variety of configurations of H/D and D_W/D. These configurations give useful insights that may be utilized in the design of high-performance fuel assemblies as used in nuclear reactors.

In order to successfully identify optimal wire spacer topologies that improve heat transfer and reduce pressure drop, the dual goal optimization approach that makes use of dimensionless design variables (H/D and D_W/D) is utilized. Incorporating the L_{16} orthogonal array method, RBF surrogate modelling, and NSGA-II optimization approach into the design space provides a thorough and effective investigation of the design space, which ultimately results in considerable enhancements to the thermal hydraulic performance of nuclear fuel assemblies.

5.2.1 Coefficient of Determination Score (R^2)

In this investigation, the Radial Basis Function (RBF) surrogate model was evaluated with regard to its performance and accuracy by means of the Coefficient of Determination (R^2) score. Using sixteen sample points that were generated by the L_{16} orthogonal array, the surrogate model was developed based on the data that was collected from simulations of computational fluid dynamics (CFD). Out of these sixteen points, twelve were utilized for the purpose of training the RBF model, while the remaining four were set aside for the purpose of the validation process. As shown in Figure 5. 3, the R^2 score, which is an important statistic for determining how accurate a model is, was computed for both the training dataset and the validation dataset.

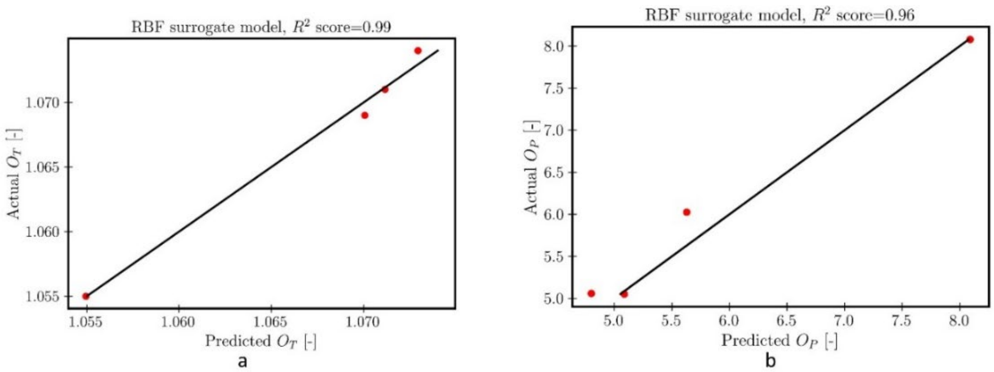

Figure 5. 1 Coefficient of Determination score of the RBF Surrogate Model (a) Heat Transfer Factor O_T (b) Pressure Drop Factor O_P

In order to generate a diverse set of sixteen sample points, the L_{16} orthogonal array approach was utilized. Each of these sample points was distinguished by a unique combination of the dimensionless design variables H/D and D_W/D. Through the usage of these sites, computational fluid dynamics (CFD) simulations were carried out in order to acquire the heat transfer and pressure drop data that were necessary for the RBF surrogate model to be successfully constructed.

A total of twelve of these sample points were subsequently utilized in the training of the RBF model. As part of the training phase, the model was fitted to the data in order to capture the link between the design variables and the objective functions, which were heat transfer and pressure drop during the training process. The model's prediction capabilities were validated by setting aside the remaining four sample points for the purpose of validation.

The R^2 score is a statistical measure that demonstrates how well the predictions made by the surrogate model and the actual data line up with one another. A score of 1.0 for R^2 indicates that there is a perfect correlation, which indicates that the model explains one hundred percent of the variability in the data. If, on the other hand, the R^2 value is closer to zero, it suggests that the model is not successful in capturing the underlying trends in the data.

The R^2 scores for both the training dataset and the validation dataset are displayed in Figure 5. 3. It was observed that the following results occurred:

- Heat Transfer Factor (O_T): The RBF surrogate model was able to attain an R^2 score of 0.99 when applied to the heat transfer dataset. The fact that the model gets a high score implies that it has a nearly perfect fit to the training data, which means that it has successfully captured the intricate correlations that exist between the design variables and the objectives.

- Pressure Drop Factor (O_P): The validation dataset had an R^2 score of 0.96, which indicates that it was an acceptable dataset. Despite the fact that this result is significantly lower than the O_T score, it still displays a good correlation, which indicates that the model is able to effectively generalize to new data points that were not included in the training set.

There is evidence that the RBF model is accurate and reliable, as demonstrated by the strong R^2 values for both datasets. A well-balanced model that avoids overfitting and maintains robust predictive performance across a variety of data subsets is shown by the slight difference that exists between the scores obtained during heat transfer factor and those obtained during pressure drop factor.

The results demonstrate that the RBF surrogate model is an effective instrument for forecasting the outcomes of design modifications in wire spacer configurations for 2×2 rod bundles for the purpose of analysis. The high R^2 scores (0.99 for O_T and 0.96 for O_P) demonstrate that the model is able to accurately reflect the data that was generated by the CFD, which ensures that the design optimization process will be guided by reliable guidance. These findings highlight the usefulness of the model in quickly exploring the design space and identifying optimal configurations that strike a balance between the performance of heat transfer and the pressure drop in Supercritical Water Reactors (SCWRs).

5.2.2 Pareto Optimal Front Analysis

Figure 5. 4 is a representation of the Pareto front, which was obtained during the process of optimization by utilizing the Non-Dominated Sorting Genetic Algorithm II (NSGA-II). This front is a representation of the set of optimal solutions that strike a balance between the competing goals of minimizing pressure drop factor (O_P) and minimizing the heat transfer factor (O_T) in the design of the wire spacer for 2×2 rod bundles. On the response surface that was produced by the Radial Basis Function (RBF) surrogate model, which was trained with the help of Computational Fluid Dynamics (CFD) data, the NSGA-II optimization was carried out.

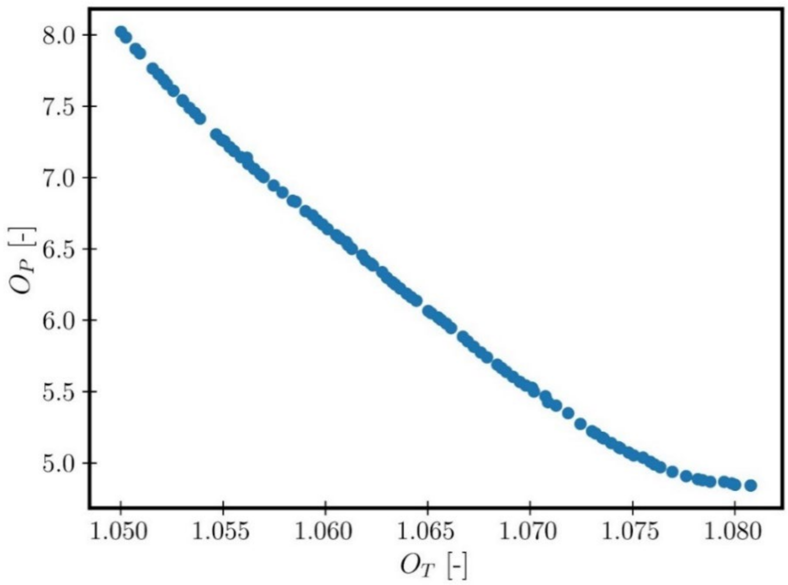

Figure 5. 2 Pareto Front of the Objective Function

5.2.2.1 Optimization Set up

- Population Size: 100
- Maximum Generations: 100
- Crossover Probability: 0.9

In order to guarantee a comprehensive exploration of the design space and convergence towards the global Pareto-optimal solutions, these parameters were selected through careful consideration.

5.2.2.2 Pareto Front Characteristics

As can be seen in Figure 5. 4, the Pareto front is a nonlinear curve that illustrates the intricate trade-offs that exist between the two goals. A number of important observations can be drawn from the Pareto front analysis:

1. Nonlinearity of the Pareto Front:

The fact that the curve is nonlinear is evidence that the connection between O_T and O_P is not a clear one. The presence of this nonlinearity means that the enhancement of one aim will have an impact that is both complex and non-proportional on the other objective.

2. Global Pareto-Optimal Front:

Note that not all regions of the response surface function represent the global Pareto-optimal front. This is an extremely important point to keep in mind. It is possible that certain regions only offer local optimal solutions, which do not give the most advantageous trade-offs between the objectives. This demonstrates how important it is to investigate the entirety of the design space in order to locate the global optimal solution.

3. Trade-Off Between Temperature and Pressure Drop:

Observations made from the Pareto front indicate that a decrease in the heat transfer factor (O_T) results in an increase in the pressure drop factor (O_P) over time.

There is a trade-off that is inherent in the design, which is that gains in heat transfer rate often result in higher pressure decreases due to increased fluid friction and turbulence.

4. Gradient Relationship:

In proportion to the increase in O_T, there is a discernible reduction in the gradient of O_P. This indicates that even minor increases in temperature reduction demand increasingly larger sacrifices in terms of pressure drop. As a result, it becomes increasingly difficult to accomplish considerable benefits in heat transfer without incurring major pressure losses.

5. Optimal Value Ranges:

Based on the findings of the analysis, it is recommended that the optimal range for O_P values in a balanced design is between 5.051 and 8.082, while the ideal range for O_T values is between 1.05 and 1.087. In order to reach a satisfactory compromise between the competing objectives, these ranges offer designers a practical guidance that they can follow.

5.2.2.3 Implications for Design

The Pareto front study offers designers a number of helpful insights, including the following:

1. Balanced Decision Making:

The Pareto front allows designers to make educated judgments about where to position their designs within the optimal range by providing them with the information they will need. Consequently, this contributes to the achievement of an acceptable equilibrium between the rate of heat transfer and the pressure drop.

2. Insight into Trade-Offs:

Understanding the repercussions of optimizing for one target over another is made easier by the clear portrayal of the trade-offs that exist between maximizing for one aim and optimizing for another. When it comes to making strategic design decisions that are in line with the overall performance goals of the reactor, having this knowledge is absolutely necessary.

3. Guidance for Future Designs:

The findings serve as a reference for the design of future wire spacers, offering a benchmark for the expected performance as well as the trade-offs that are involved.

Figure 5. 4 is an illustration of the Pareto front that was generated from the NSGA-II optimization. This front successfully illustrates the precise balance that must be maintained between the goals of decreasing pressure drop and increasing heat transfer rate in wire spacer design. It is essential to provide great consideration to design optimization because of the nonlinear character of the front, which exposes the complexity of the trade-offs and emphasizes the necessity for careful analysis. In order to achieve high-performance fuel assembly designs in nuclear reactors, the optimal ranges for O_P and O_T that have been identified provide an opportunity for practical guidance.

5.3 Summary

This chapter centres on the design optimization of wire spacers for 2×2 rod bundles in nuclear reactors. The goal was to improve the thermal-hydraulic performance by achieving a balance between two key objectives: improving heat transfer rate and minimizing pressure loss. The optimization framework combines the Radial Basis Function (RBF) surrogate model and the Non-Dominated Sorting Genetic Algorithm II (NSGA-II) to efficiently determine the best design configurations.

The optimization method commences with computer modelling, wherein the wire spacer and rod bundle's geometry is constructed, and a mesh is generated for meticulous study. The governing equations for fluid flow and heat transfer are utilized, and suitable boundary and operating conditions are specified to model the system's behaviour at supercritical circumstances. The quality and reliability of the CFD model are ensured by conducting a grid independence test and validating it against experimental data.

The RBF surrogate model was constructed using data acquired from CFD simulations to approximate the intricate correlation between design variables and objectives. A dataset with sixteen sample points is created using the L_{16} orthogonal array approach, ensuring diversity. The model was trained using twelve points, and its predicted performance was validated using the remaining four points. The RBF model has strong performance with R^2 scores of 0.99 for heat transfer factor and 0.96 for pressure drop factor, suggesting its ability to accurately capture the fundamental patterns in the data.

The optimization framework utilizes the NSGA-II algorithm to effectively explore the design space. NSGA-II is a robust multi-objective optimization method that produces a Pareto front, which demonstrates the compromises between heat transfer rate and pressure drop. NSGA-II, with a population size of 100, a maximum of 100 generations, and a crossover probability of 0.9, is able to identify the Pareto-optimal solutions that provide the most favourable trade-offs between the conflicting objectives.

The Pareto front, as illustrated in the chapter, is a non-linear curve that emphasizes the intricate correlation between the objectives. The analysis demonstrates that there is a direct correlation between the improvement in heat transfer rate and the increase in pressure drop, highlighting the inherent trade-offs involved in the design process. The ideal range for the pressure drop factor (O_P) is from 5.051 to 8.082, while the optimal range for heat transfer rate factor (O_T) is from 1.05 to 1.087. These findings offer pragmatic recommendations for creating wire spacers that achieve a harmonious equilibrium between performance and efficiency.

The results of the chapter illustrate the effectiveness of the RBF surrogate model and NSGA-II in optimizing wire spacer designs. The elevated R^2 scores validate the precision of the surrogate model, while the Pareto front analysis offers significant insights into the compromises between the objectives. The determined optimal ranges assist designers in attaining a harmonious design that enhances reactor performance while reducing operating limitations.

To summarize, utilizing the RBF surrogate model and NSGA-II optimization technique for the design optimization of wire spacers in 2×2 rod bundles provides a reliable and effective method for improving the performance of nuclear reactors. The combination of computational modelling, surrogate modelling, and advanced optimization algorithms offers a thorough framework for investigating intricate design spaces and determining the most effective solutions. The insights presented in this chapter provide a helpful point of reference for future endeavours in the design and optimization of nuclear reactors.

Chapter 6

Conclusions, Future Scope and Social Impact

6.1 Conclusions

The purpose of this thesis is to investigate the intricate thermal-hydraulic dynamics of 2×2 rod bundles in supercritical water reactors (SCWRs), with a particular emphasis on the function that wire spacers play in maximizing performance efficiencies. A number of significant findings have been made as a result of intensive computational fluid dynamics (CFD) simulations, as well as multi-objective optimization with surrogate models and genetic algorithms. This section provides a summary of the findings that were observed as a result of the investigation into the axial velocity, velocity vectors, pressure, and temperature distributions. Furthermore, it discusses the advantages of utilizing helical wires as opposed to straight wires, as well as the effectiveness of the optimization strategies that were utilized.

6.1.1 Axial Velocity

1. Velocity Distribution:

   The results of the computational fluid dynamics (CFD) analysis show that the configuration of wire spacers significantly affects the flow dynamics in the rod bundle. The mixed design which is wrapped in clockwise and anti-clockwise direction, facilitates a more intricate and dispersed flow, perhaps resulting in advantages in terms of heat transfer and reactor efficiency. The arrangement in a clockwise direction, however, provides a centre flow channel that is highly concentrated. These findings provide valuable insights that can be utilized to enhance the design of wire spacers in SCWR fuel assemblies, resulting in improved thermal hydraulic performance.

2. Implications for Thermal Performance:

The enhanced axial velocity in helical wire topologies with mixed arrangement promotes a more effective transfer of heat from the fuel rods to the coolant, hence minimizing the risk of localized overheating. The preservation of thermal stability and the prevention of material degradation in reactor components are both important reasons that's why this is so important.

6.1.2 Velocity Vectors

1. Swirl Flow Characteristics:

It can be observed that the conventional helical wire wrap generates a strong vortex that is mainly confined to the outer regions of the duct. This hampers the effectiveness of the helical wire wrap in cooling the central portions of the rod bundle. In contrast, the new-pattern geometry has the capability to achieve a more uniform distribution of flow and significantly enhance the flow through the spaces between the pins. This feature is crucial for maintaining uniform temperature profiles and preventing localized overheating.

The velocity vector contours clearly indicate that the new-pattern geometry offers several advantages, such as greater coolant distribution and improved thermal performance. This design enhances heat evacuation from the fuel rods by increasing the flow rate through the narrow inter-pin gaps, hence reducing the formation of hot spots.

2. Uniform Flow Distribution:

The new-pattern wire spacer revealed a flow distribution that was more consistent throughout all of the sub-channels than before. Because of this homogeneity, the coolant circulates evenly around each rod, which has the effect of improving the overall thermal performance and lowering the danger of temperature-induced structural problems.

3. Performance Benefits:

There is a correlation between the enhanced tangential velocity component in helical designs and the increased rate of coolant circulation and heat removal. Because of the large improvement in flow characteristics, the thermal-hydraulic performance of the reactor is significantly enhanced. As a result, helical wire spacers with mixed arrangement are a more advantageous option than straight wire spacers.

6.1.3 Pressure Distribution

1. Pressure Variation:

According to the findings of the static pressure contour analysis, standard wire wrap designs display a pressure variation that is impacted by the wire wraps that are closely adjacent to them. This variance can result in an increased pressure drop throughout the system, which will have an effect on the efficiency of the system as a whole.

The new-pattern wire spacer design, on the other hand, displayed lower pressure variance. This was due to the fact that there were less wire spacers. This streamlined design reduces the number of disturbances that occur along the flow channel, which ultimately leads to a more consistent and effective distribution of pressure throughout the regenerative reactor.

2. Pressure Drop Analysis:

In spite of the fact that helical wire spacers cause a larger pressure drop (4669 Pa) in comparison to straight wires pressure drop (4235 Pa), this increase is compensated for by the major gains in heat transfer and flow uniformity that they bring about. In order to fully optimize the performance of the reactor, it is essential to consider the trade-off between pressure drop and rate of heat transfer.

6.1.4 Temperature Distribution

1. Improved Thermal Performance:

According to the findings of the temperature distribution analysis, the new-pattern wire spacer design that incorporates helical wires results in a considerable reduction of 8.5 degrees Celsius in the peak temperature. It is possible to credit this reduction to the improved mixing and more even dispersion of the coolant, which prevented the formation of hot patches.

As a result of the lower peak temperatures that are found in helical wire topologies, the risk of material degradation and the possibility of rod melting is reduced, which both improves the safety of the reactor components and increases their longevity.

2. Uniform Temperature Distribution:

Because of the arrangement of the helical wire spacer, the rod bundle had a temperature distribution that is more consistent throughout. This consistency helps to reduce thermal strains and prevents localized overheating, both of which are essential for ensuring that the structural integrity of the reactor components is preserved under all circumstances, including those that are supercritical.

6.1.5 Helical vs. Straight Wire Spacers

1. Heat transfer rate:

When it comes to the rate of heat transfer, helical wire spacers perform better than straight wire spacers because of the induced swirl flow that they produce. This swirl effect improves the mixing of the coolant and optimizes the rate with which heat is removed from the fuel rods.

Even though helical wire spacers have a higher pressure drop than other types of spacers, their overall thermal performance is greatly enhanced, which makes them a more desirable option for SCWR applications rather than other options.

2. Flow Uniformity:

The uniform flow distribution that is accomplished with helical wire spacers guarantees that the coolant is dispersed equally around each fuel rod. This results in an increase in the overall thermal efficiency and a reduction in the risk of localized overheating.

Because the advantages of utilizing helical wire spacers, which include enhanced heat transfer and flow homogeneity, exceed the drawbacks of increased pressure drop, these spacers are an excellent option for improving the performance of reactors.

6.1.6 Surrogate Model and Optimization

1. High Accuracy of Surrogate Model:

A high level of accuracy was established by the RBF surrogate model, which received R^2 scores of 0.99 for heat transfer factor O_T and 0.96 for pressure drop factor O_P. By confirming that the surrogate model accurately captures the relationship between design factors and performance indicators, these scores make it possible to explore the design space in an efficient manner.

2. Efficient Optimization with NSGA-II:

With the use of the NSGA-II optimization technique, a nonlinear Pareto front was successfully established. This front serves as an illustration of the trade-offs that exist between optimizing heat transfer rate and minimizing pressure drop. In order to achieve high-performance wire spacer designs, it is essential to strike the best balance between these opposing objectives, as indicated by the outcomes of the optimization process.

The Pareto front analysis offers extremely helpful insights into the ideal ranges for pressure drop factor (ranging from 5.051 to 8.082) and heat transfer rate factor (ranging from 1.05 to 1.087). The design of wire spacers that maximize reactor performance while retaining operational stability can be guided by these ranges, which provide practical direction.

3. Design Implications:

The results of the optimization research illustrate how important it is for wire spacer designs to strike a balance between pressure drop and rate of heat transfer. The optimal ranges that have been determined serve as a basis for future design work, guaranteeing that the wire spacers improve the overall performance of the reactor without compromising either its safety or its efficiency.

6.2 Future Scope

A number of potential future research directions can be pursued in order to further develop the optimization and performance of wire spacers in SCWR applications. These directions can be pursued by building on the findings of this thesis. Among these directions are the investigation of additional design variables, the investigation of improved modelling approaches, and the validation of the conclusions through experimental studies.

6.2.1 Extended Design Parameters

1. Advanced Geometries:

It is possible for future research to investigate a greater variety of wire spacer geometries, such as adjusting the pitch-to-diameter ratio (H/D) and the diameter-to-fuel rod diameter ratio (D_W/D). It is possible to gain a more in-depth understanding of the effects that various helical patterns, wire shapes, and materials have on thermal-hydraulic performance by conducting research on these topics.

It is possible to further improve one's understanding of flow dynamics and heat transfer mechanisms at supercritical circumstances by investigating complex geometries, such as twisted tapes or spiral inserts.

2. Material Innovations:

It is possible that further performance enhancements could be achieved by doing research on alternate materials for wire spacers. These materials could offer improved thermal conductivity or reduced friction coefficients. In supercritical settings, the durability and efficiency of wire spacers can be improved by the use of advanced materials such as high-temperature alloys or composite materials.

3. Multi-Objective Design Optimization:

By extending the optimization framework to incorporate additional objectives, such as lowering material costs or increasing structural integrity, it is possible to provide a more comprehensive approach to the design of wire spacers. Utilizing multi-objective optimization approaches can assist in locating designs that strike a balance between various performance parameters, hence guaranteeing the best possible performance from the reactor.

6.2.2 Dynamic Flow and Thermal Analyses

1. Transient Behaviour:

By extending the scope of the study to incorporate transient simulations that take into account dynamic changes in flow and temperature conditions, it is possible to gain a more in-depth understanding of the operational stability and safety of the reactor under a variety of load scenarios. The investigation of transient flow characteristics, such as start-up, shutdown, or emergency scenarios, will contribute to a better understanding of reactor dynamics and will assist in optimizing design parameters for applications in the real world.

2. Advanced Flow and Heat Transfer Models:

By using modern flow and heat transfer models, such as large eddy simulations (LES) or direct numerical simulations (DNS), it is possible to obtain more in-depth insights into the intricate interactions that occur between coolant flow and wire spacers. The accuracy and dependability of the simulations are improved as a result of these models' capacity to represent flow structures and turbulence effects on a small scale.

6.2.3 Real-Time Optimization and Control

1. Adaptive Systems:

It is possible to guarantee continual performance gains and operational safety in real-world reactor situations by developing real-time monitoring and adaptive control systems that make use of optimization frameworks. By integrating sensor data with control algorithms, it is possible to provide dynamic optimization, which guarantees optimal performance and safety under a wide range of operating situations.

2. Intelligent Control Strategies:

In the future, research could investigate the possibility of developing intelligent control techniques, such as controllers based on machine learning or predictive maintenance algorithms, with the goal of improving the performance and reliability of SCWRs. These methods can assist in the detection and mitigation of possible problems in real time, thereby guaranteeing that the operation of the reactor is both safe and efficient.

3. Integration with Reactor Control Systems:

Incorporating the optimization framework into the control systems that are already in place for the reactor can result in a more comprehensive approach to the management of the reactor. This integration can make it possible to make adjustments to the operating settings in real time, which guarantees the highest possible level of performance and safety even when working in dynamic conditions.

6.2.4 Experimental Validation

1. Prototype Testing:

Putting together prototypes that are based on improved designs and testing them in experimental setups will validate the findings of the computational analysis and give empirical data that can be used to modify the models even further. Through the use of experimental validation, any inconsistencies that may exist between simulated and real-world performance can be identified. This helps to ensure that the optimized designs are able to fulfil the requirements for practical performance.

2. High-Temperature Testing:

In order to gain significant insights into the thermal and mechanical performance of wire spacers, it is necessary to conduct high-temperature testing of these components under supercritical conditions. Through the use of these experiments, it is possible to evaluate the longevity and dependability of various spacer designs, thereby assuring that they are capable of satisfying the stringent requirements of supercritical reactors.

3. Flow and Heat Transfer Experiments:

Performing flow and heat transfer experiments using advanced diagnostic techniques, such as particle image velocimetry (PIV) or laser-induced fluorescence (LIF), can provide detailed insights into the flow dynamics and heat transfer mechanisms in supercritical conditions. These experiments can help validate the computational models and provide data for further optimization.

6.2.5 Integration with Reactor Core Design

1. System-Level Optimization:

By extending the optimization framework to incorporate core-level concerns, such as fuel arrangement and coolant flow patterns, a holistic approach to improving the overall performance of the reactor will be provided. Optimization at the system level can assist in determining design configurations that maximize thermal efficiency while simultaneously minimizing operational expenses, so guaranteeing that the reactor performs at its highest possible level.

2. Reactor Core Design Integration:

It is possible to improve the thermal-hydraulic performance of the complete reactor system by integrating the optimized wire spacer designs with the overall design of the reactor core. In the future, research might investigate the impact of various wire spacer designs on core-level parameters, such as the distribution of neutron flux and fuel burnup. This would provide a more holistic approach to the design of reactors.

3. Coupled Thermal-Hydraulic and Neutronic Analyses:

It is possible to gain a more in-depth comprehension of the interactions that occur between thermal and neutronic processes in supercritical reactors by carrying out combined thermal-hydraulic and neutronic investigations. Through these analyses, it is possible to identify design choices that optimize both thermal and neutronic performance, so assuring that the operation of the reactor is both safe and efficient.

6.2.6 Economic and Environmental Impact Assessment

1. Cost-Benefit Analysis:

In order to offer a more comprehensive review of the design's practicality, it is necessary to conduct rigorous economic analyses in order to evaluate the cost implications of optimal designs and their potential consequences on operational costs and lifespan expenses. In the future, research might investigate the economic benefits of various wire spacer designs, such as the possible cost reductions that could result from greater thermal efficiency and lower maintenance requirements.

2. Environmental Impact:

It is important to evaluate the environmental benefits of better reactor designs, such as decreased fuel consumption and fewer emissions, in order to emphasize the advantages of optimized wire spacers in terms of sustainability. It is possible for future study to investigate the environmental effects of various wire spacer designs, including the potential of these designs to lessen emissions of greenhouse gases and to reduce the amount of radioactive waste produced.

3. Lifecycle Assessment:

By conducting lifecycle assessments of various wire spacer designs, it is possible to provide a more thorough review of the environmental consequences of these designs. This evaluation can include potential benefits and drawbacks that may occur over the course of the reactor's full lifecycle. Through these studies, it is possible to develop design configurations that optimize sustainability while simultaneously minimizing their impact on the environment.

By addressing these potential future study paths, the findings of this thesis can be utilized as a foundation for the development of nuclear reactors that are more effective, safe, and environmental friendly. This research will contribute to the advancement of nuclear technology and energy security, guaranteeing that SCWRs will continue to play an important role in the global energy landscape. The insights gathered from this research contribute to the improvement of nuclear technology.

6.3 Social Impact

The study presented in this thesis has profound societal ramifications, as it tackles crucial areas of nuclear reactor technology that can result in substantial advantages for society. This study aims to improve the thermal-hydraulic characteristics of supercritical water reactors (SCWRs), which are crucial for the global shift towards cleaner and more dependable energy sources. By doing so, it contributes to the advancement of safer, more efficient, and more sustainable nuclear energy. The societal implications of this thesis can be comprehended by examining its contributions to energy security, environmental sustainability, economic prosperity, and public health and safety.

6.3.1 Energy Security and Reliability

One of the foremost concerns in the 21st century is to guarantee a consistent and dependable energy supply to satisfy the increasing worldwide demand. This research has enabled significant advancements in SCWR technology, ensuring a reliable and consistent energy source. Supercritical water-cooled reactors (SCWRs), which have a higher thermal efficiency and lower fuel consumption compared to traditional reactors, offer a promising solution for ensuring long-term energy security. This thesis improves the thermal hydraulic characteristics and safety of SCWRs by optimizing the design of wire spacers, making them a more practical and environmentally friendly choice for generating power.

6.3.2 Environmental Sustainability

Climate change is a pressing worldwide issue, and it is essential to decrease the release of greenhouse gases in order to lessen its effects. Nuclear energy, including sophisticated Supercritical Water-Cooled Reactor (SCWR) technology, provides a low-carbon alternative to fossil fuels. The improvements in thermal-hydraulic performance described in this thesis lead to the development of more efficient nuclear reactors that emit less emissions per unit of energy produced. This is in line with international initiatives to decrease carbon emissions and shift towards cleaner forms of energy, ultimately making a positive impact on the environment for future generations.

6.3.3 Technical Safety

Ensuring safety is of utmost importance when operating nuclear reactors. The results of this thesis, specifically the enhanced heat transfer and decreased pressure drop produced by optimizing wire spacer designs, lead to the safer functioning of SCWRs. Implementing advanced safety protocols decreases the probability of incidents, thereby safeguarding both employees at the facility and the neighbouring communities. In addition, the decrease in structural failures caused by temperature reduces the likelihood of radioactive material being released, thereby protecting public health and ensuring public trust in nuclear energy.

6.3.4 Contribution to Global Energy Goals

The research is in line with the global energy objectives established by organizations such as the International Atomic Energy Agency (IAEA) and the United Nations. This thesis aims to enhance the efficacy and security of nuclear reactors, thereby contributing to initiatives aimed at broadening the availability of clean and cost-effective energy. This is especially crucial for emerging nations that aim to fulfil their energy requirements while minimizing ecological consequences and advancing sustainable growth.

6.3.5 Promoting Public Awareness and Acceptance

This thesis aims to enhance public awareness of nuclear energy by addressing crucial issues pertaining to nuclear reactor safety and efficiency. The proven advantages of optimized Supercritical Water-cooled Reactor (SCWR) technology, such as improved safety and minimized environmental footprint, will help foster wider public approval of nuclear energy as an essential element of the global energy portfolio. Enhanced knowledge and comprehension of the progress in nuclear technology can facilitate more knowledgeable deliberations regarding energy policy and the contribution of nuclear power in attaining sustainable energy objectives.

6.3.6 Addressing Future Energy Challenges

The technological developments in Supercritical Water-cooled Reactor (SCWR) discussed in this thesis provide society with the necessary means to tackle upcoming energy concerns. With the ongoing growth of the world population and the increasing energy demands, there is an urgent need for efficient and sustainable energy solutions. This research aims to advance nuclear technologies that can fulfill these requirements while minimizing their environmental footprint, therefore providing a dependable energy source for future generations.

To summarize, the study outlined in this thesis has substantial social ramifications, aiding in the development of a safer, more effective, and more environmentally friendly energy future. This work contributes to worldwide initiatives aimed at achieving energy security, environmental sustainability, economic growth, and public health and safety by enhancing the technology of supercritical water reactors. The knowledge acquired from this research will have a pivotal influence on influencing the future of nuclear energy and its societal consequences.

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Appendix I

User Defined Functions of Thermo-Physical Properties of Supercritical Water

1. Density

```
#include <stdio.h>
#include <math.h>
#define real double
real Density(real T)
{
real rho;
if ((200 <= T) && (T< 385.33))
  rho    =    (1293.92056-3.29661*T)/(1-3.86706*pow(10,-4)*T-5.2505*pow(10,-
6)*pow(T,2));
else
  rho = (1505.05919-4.18263*T)/(1+0.03488*T-9.9752*pow(10,-5)*pow(T,2));
return rho;
}
int main (void)
{
   real T;
   for (T=200.0;T<600.0;T+=10)
       printf("T: %f, rho: %f\n",T,Density(T));
   return 0;
}
```

2. Viscosity

```
#include <stdio.h>
#include <math.h>
#define real double
real Viscosity(real T)
{
real mu_lam;
if ((200 <= T) && (T< 385.33))
  mu_lam   =   (0.00367-9.36547*pow(10,-6)*T)/(1+0.12159*T-3.11757*pow(10,-
4)*pow(T,2));
else if ((385.33 <= T) && (T< 416))
```

```
  mu_lam = (2.1189*pow(10,-5)-5.57259*pow(10,-8)*T)/(1-0.00314*T+1.37453*pow(10,-6)*pow(T,2));
else
  mu_lam = (1.59838*pow(10,-5)+9.27962*pow(10,-9)*T)/(1-8.37242*pow(10,-4)*T+3.54498*pow(10,-7)*pow(T,2));
return mu_lam;
}
int main (void)
{
   real T;
   for (T=200.0;T<1000.0;T+=10)
      printf("T: %f, ktc: %f\n",T,Viscosity(T));
   return 0;
}
```

3. Thermal Conductivity

```
#include <stdio.h>
#include <math.h>
#define real double
real Thermal(real T)
{
real ktc;
if ((200 <= T) && (T< 384))
  ktc = (0.69885-0.00147*T)/(1-0.00242*T+1.86675*pow(10,-6)*pow(T,2));
else if ((384 <= T) && (T< 516))
  ktc = (0.05679-1.61755*pow(10,-4)*T)/(1-0.00316*T+1.3633*pow(10,-6)*pow(T,2));
else
  ktc = 11.47972-0.12239*T+5.89557*pow(10,-4)*pow(T,2)-1.67055*pow(10,-6)*pow(T,3)+3.06468*pow(10,-9)*pow(T,4)-3.76735*pow(10,-12)*pow(T,5)+3.09803*pow(10,-15)*pow(T,6)-1.64116*pow(10,-18)*pow(T,7)+5.0764*pow(10,-22)*pow(T,8)-6.97955*pow(10,-26)*pow(T,9);
return ktc;
}
int main (void)
{
   real T;
   for (T=200.0;T<1000.0;T+=10)
      printf("T: %f, ktc: %f\n",T,Thermal(T));
```

```
   return 0;
}
```

4. **Specific Heat**

```
#include <stdio.h>
#include <math.h>
#define real double
real specificHeat(real T)
{
real cp;
if ((200 <= T) && (T< 385.33))
  cp = (144.97013-0.28834*T)/(1+0.1767*T-4.62394*pow(10,-4)*pow(T,2));
else
  cp = (1.34808-0.00477*T)/(1-0.0028*T+4.82658*pow(10,-7)*pow(T,2));
return cp;
}
int main (void)
{
   real T;
   for (T=200.0;T<600.0;T+=10)
      printf("T: %f, cp: %f\n",T,specificHeat(T));
   return 0;
}
```

Appendix II

Python Code for Dual Objective Optimization using RBF surrogate Model and NSGA II

```
from pymoo.algorithms.moo.nsga2 import NSGA2
from pymoo.problems import get_problem
from pymoo.optimize import minimize
from pymoo.visualization.scatter import Scatter
from pymoo.core.problem import ElementwiseProblem
import numpy as np
import pandas as pd
from sklearn.model_selection import train_test_split
from sklearn.preprocessing import MinMaxScaler
from sklearn.metrics import r2_score
from matplotlib import pyplot
from smt.surrogate_models import RBF

#********************** RBF Surrogate Model ********************
df=pd.read_csv("Optimization.txt", delimiter='\t', skiprows=1,header=None)

X=df.drop([2,3],axis=1)
y1=df[2]     #Temperature
y2=df[3]     #Pressure

#Normalizing Data
X[0]= X[0]/75
X[1]= X[1]/0.15
X=np.array(X)
```

```
y1=np.array(y1)
y2=np.array(y2)

#****************************** y1 ************************************
#Splitting Data
X_train,X_test,y_train,y_test=train_test_split(X, y1,test_size=0.2)

sm1=RBF(d0=X_train.shape[0])
sm1.set_training_values(X_train,np.array(y_train))
sm1.train()

y_predicted=sm1.predict_values(X_test)

# Calculate R-squared score for the predictions
r2 = r2_score(y_test, y_predicted)
print(f"R-squared score: {r2}")



#plotting r2 score
pyplot.rcParams['text.usetex'] = True
pyplot.figure(figsize=(6.4,4.8),dpi=1000)
pyplot.rcParams["axes.linewidth"] = 2
pyplot.plot(y_test,y_test,color='black',lw=2)
pyplot.scatter(y_predicted,y_test,color='red',marker='o')
pyplot.xlabel('Predicted $O_T$ [-]',fontsize=16)
pyplot.ylabel('Actual $O_T$ [-]',fontsize=16)
pyplot.title(f'RBF surrogate model, $R^2$ score={r2:.2f}',fontsize=16)
```

```
pyplot.tick_params(axis='both',size=8,labelsize=15,direction='inout')
pyplot.tight_layout()

#****************************** y2
************************************
#Splitting Data
X_train,X_test,y_train,y_test=train_test_split(X, y2,test_size=0.2)

sm2=RBF(d0=X_train.shape[0])
sm2.set_training_values(X_train,np.array(y_train))
sm2.train()

y_predicted=sm2.predict_values(X_test)

# Calculate R-squared score for the predictions
r2 = r2_score(y_test, y_predicted)
print(f"R-squared score: {r2}")



#plotting r2 score
pyplot.rcParams['text.usetex'] = True
pyplot.figure(figsize=(6.4,4.8),dpi=1000)
pyplot.rcParams["axes.linewidth"] = 2
pyplot.plot(y_test,y_test,color='black',lw=2)
pyplot.scatter(y_predicted,y_test,color='red',marker='o')
pyplot.xlabel('Predicted $O_P$ [-]',fontsize=16)
pyplot.ylabel('Actual $O_P$ [-]',fontsize=16)
pyplot.title(f'RBF surrogate model, $R^2$ score={r2:.2f}',fontsize=16)
```

```
pyplot.tick_params(axis='both',size=8,labelsize=15,direction='inout')
pyplot.tight_layout()

#********************************** NSGA ll ***************
class MyProblem(ElementwiseProblem):

    def __init__(self):
        super().__init__(n_var=2,
                         n_obj=2,
                         n_ieq_constr=0,
                         xl=np.array([12.5,0.075]),
                         xu=np.array([75,0.15]))

    def _evaluate(self, x, out, *args, **kwargs):
        #f1 = 100 * (x[0]**2 + x[1]**2)
        #f2 = (x[0]-1)**2 + x[1]**2


        X_GA=np.array(x)
        X_GA=X_GA.reshape(-1,2)
        X_GA[0][0]=X_GA[0][0]/75.
        X_GA[0][1]=X_GA[0][1]/0.15
        f1=(sm1.predict_values(X_GA))
        f2=(sm2.predict_values(X_GA))

        out["F"] = [f1, f2]

problem = MyProblem()
```

```
algorithm = NSGA2(pop_size=100)

res = minimize(problem,
            algorithm,
            ('n_gen', 200),
            seed=1,
            verbose=False)

plot = Scatter(figsize=(11,7))
plot.add(problem.pareto_front(), plot_type="line", color="black", alpha=0.7)
plot.add(res.F, facecolor="none", edgecolor="red")
plot.show()

#Results
pyplot.rcParams['text.usetex'] = True
pyplot.figure(figsize=(6.4,4.8),dpi=1000)
pyplot.rcParams["axes.linewidth"] = 2
pyplot.scatter(res.F[:,0],res.F[:,1])
pyplot.xlabel("$O_T$ [-]",fontsize=16)
pyplot.ylabel('$O_P$ [-]',fontsize=16)
pyplot.tick_params(axis='both',size=8,labelsize=15,direction='inout')
pyplot.tight_layout()
#pyplot.ylim(top=8.1,bottom=5)
#pyplot.xlim(left=1.04, right=1.06)
```

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Review

Supercritical water flow in heated wire wrapped rod bundle channels: A review

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ARTICLE INFO

Keywords:
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ABSTRACT

Understanding the heat transfer features of supercritical water is one of the most critical issues in the development of Supercritical Water-Cooled Reactors (SCWRs). This review article provides a state-of-the-art description of the thermal-hydraulics performance of water flowing at supercritical pressures in tubes with wire-wrapped rod bundles and attempts to gain a fundamental understanding of the unique characteristics. Experimental and numerical research relevant to the supercritical water heat transfer in sub-channels with wire-wrapped rod bundles is reviewed. Supercritical pressure, thermo-physical properties of water, and relevant parametric effects (e.g., effects of mass flux, heat flux, pressure) are outlined on heat transfer performance. The reliability of turbulence models and the simulation of the buoyancy and turbulent heat flux remain major challenges. Therefore, the transfer of heat of supercritical water under SCWR operating conditions with helical wire as a spacer requires further research.

1. Introduction

The advantages of higher thermal efficiency of the supercritical water-cooled reactor (SCWR) and simplification of its system make this the most desirable as one of GIF's six Generation IV reactors (Abram and Ion, 2008) (Zhao et al., 2022) (Zang et al., 2020). SCWR designs seek to reduce the surface temperature of the cladding directly to guarantee the safety of fuel rod assembly, and the use of grid spacer is an important method of reducing the temperature of cladding (Ishiwatari et al., 2009). The two main types of spacers used in SCWRs are spacer grids and wire spacers (Xiao et al., 2018). The SCWR is a modern light water reactor (LWR) next-generation technology (US, 2014). To ensure it would fulfil the commercial needs for improved safety, thermal efficiency, and costs, its design incorporated technologies already in use at LWRs and fossil fuel plants. It is expected that the SCWR will operate under supercritical pressure settings, which makes heat transmission in the reactor core more difficult or significantly different than in conventional LWRs. Research efforts over the last decade concentrated on the essential characteristics of heat transfer in tubes and normal SCWR sub-channels with supercritical fluids (Chen et al., 2019). The SCWR concept's most critical feature is that the coolant's nominal pressure is usually 25 MPa. The pressure density reaches the critical water pressure value (22.064 MPa). The temperature of water (500 °C) at the outlet of the reactor pressure vessel is higher than the critical temperature, which is 373.95 °C. As a sub-critical temperature, but as a supercritical pressure fluid, the coolant approaches the reactor pressure vessel (RPV) and undergoes the pseudo-critical transition depending on the pressure (384.95 °C at 25 MPa) at the pseudo-critical temperature and leaves the reactor pressure vessel as a supercritical fluid (water in the case of SCWR) (Kiss and Mervay, 2019). The SCWR is a once-through type water-cooled reactor running above the critical water pressure (22.1 MPa) and providing the turbine system with supercritical pressure steam at a high temperature (Oka and Koshizuka, 2001). It is expected that the plant design would achieve greater thermal efficiency and a simple system than the current nuclear power plants (Mori et al., 2012) (Zhao and Gu, 2020). If we use supercritical water, we can boost our thermal efficiency from 33 to 35% to about 45%. As a result, many theoretical and experimental studies of supercritical water have been published (Huang et al., 2016).

SCWR designs are aimed at achieving a safe, reliable, and low-power cost operation. However, since beyond the critical pressure, the reactor coolant does not undergo the transfer of liquid vapor, the system of a power plant of SCWR will become smaller by removing the recirculation pump, steam generator, and separation of the steam-water system, which will further minimize capital costs (Wang et al., 2012). Some of the crucial technical challenges in designing an SCWR are not fully

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Numerical simulation of thermal hydraulics of supercritical pressure water with 2×2 rod assembly wrapped differently with a wire

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ARTICLE INFO

Keywords:
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ABSTRACT

Through the use of CFD, a thermal-hydraulic analysis was carried out on a set of 4 rod assemblies. The common wire-wrapped spacers on the fuel assemblies are wound in the same direction for all rods. Because of this design, it is anticipated that there would be an adverse flow in all of the sub-channels. In the course of this analysis, this flow was demonstrated to be real. In order to achieve desirable flow characteristics for coolant mixing, a novel configuration for wire-wrapped spacers is proposed here. This pattern designates a group of 4 rods, the rods have the wrapping system of clockwise and anticlockwise pattern. It was determined that this pattern had an exceptional mixing effect as well as a uniform flow. This pattern has a maximum temperature area that is approximately 8.5 °C lower than that of the conventional wire-wrapped fuel bundle arrangement. When compared to the regular pattern, the pressure drop caused by the new arrangement was approximately 1.9% less severe. The redesigned layout of the rod assembly improves heat transfer while also reducing the amount of pressure loss that occurs.

1. Introduction

The Supercritical Water-Cooled Reactor (SCWR) is one of the six advanced nuclear reactors which was come up with Generation-IV International Forum (GIF) (Zhao and Gu, 2020) (Hu et al., 2018) (bo Li et al., 2017) (US, 2014). Many countries like Canada, Europe, Japan, and China have studied the concept of Super Critical Water Cooled Reactors (SCWRs) to improve the current thermal efficiencies of nuclear power plants that are operating under Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs) (bo Li et al., 2017). The SCWRs could be cost-effective as they removed some of the thermal systems to use the supercritical water directly without any phase change.

From the designer's point of view, SCWRs are assumed to be operated around the critical pressure of 25 MPa, with inlet and outlet temperatures of 280 °C and 500 °C or more. As supercritical water is an fine coolant, the heat flux of the fuel rod surface is more than 1500 kW/m^2, while the average mass flux is around 1600 kg/m^{2}s to keep the peak temperature of cladding below 650 °C (bo Li et al., 2017).

The flow rate of the coolant in SCWRs is less than in conventional reactors, as the supercritical water has immense heat capacity in the pseudo-critical region. Because there is no phase change that occurs, the burning phenomenon does take place in SCWRs (Hu et al., 2018). The properties near the pseudo-critical area of supercritical water exhibit rapid variations (Bodkha and Maheshwari, 2021). Due to these rapid variations in the properties, the study of heat transfer characteristics gained importance. Many scholars (Yamagata et al., 1972) (Shiralkar and Griffith, 1968) (Koshizuka et al., 1995) (Wang et al., 2021, 2023; Yang et al., 2010; Su et al., 2013) have studied the unique heat transfer distinctive of supercritical water for its better understanding due to variations in its properties. Recently, some universities like Shanghai Jiao Tong University and Xi'an Jiao Tong University in China have studied the heat transfer characteristics in a 2 × 2 rod bundle with supercritical water to gain a better understanding of fuel rods with supercritical water as a coolant (Wang et al., 2014) (Gu et al., 2015) (Wang et al., 2016).

Heat transfer enhancement is a significant phenomenon that should be kept in mind while designing the reactor core. A proper heat transfer from the fuel assembly to the coolant can prevent the fuel rod from being melted out and restrict the accident from occurring. For the enhancement of heat transfer from the fuel assembly, spacers of different shapes are used. Many researchers have investigated the aftermath of spacers in determining the heat transfer phenomena from fuel rods to water (Wang et al., 2020; He et al., 2021; Chen et al., 2021; Li et al., 2020).

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MESRA, RANCHI

JHARKHAND, INDIA

International Conference on Thermo-Fluids and System Design (ICTFSD 2022)

March 22-23,2022

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SUMMARY

Gaurav Kumar is an accomplished academic with a Ph.D. in Mechanical Engineering with specialization in thermal hydraulics from Delhi Technological University, an M.Tech in Nuclear Engineering from the same institution, and a B.Tech in Mechanical Engineering from Punjab Technical University. His diverse educational background and expertise in Computational Fluid Dynamics (CFD) underscore his proficiency in both mechanical and nuclear engineering domains.

PROFESSIONAL EXPERIENCE

Delhi Technological University

Guest Assistant Professor (Department of Mechanical Engineering) *May 2022 to Present*

- Taught a range of courses for B.Tech and M.Tech programs, including subjects such as Engineering Graphics, Basic Mechanical Engineering, Product Design and Simulation, Fluid Mechanics, and Computational Fluid Dynamics (CFD).
- Developed and delivered comprehensive lecture materials, including presentations, notes, and hands-on lab exercises, to enhance student learning and engagement.
- Designed and assessed coursework, assignments, and examinations to evaluate student understanding and progress.
- Organized and supervised laboratory sessions, ensuring students gained practical experience with experimental techniques, **CAD** model development, **CFD** and **FEA** software.
- Engaged in professional development activities, including attending conferences and workshops, to stay updated on the latest trends and advancements in the field.

Delhi Technological University

Research Assistant (Department of Mechanical Engineering) *August 2017 to Dec 2024*

- Conducted advanced research in thermal hydraulics, focusing on the application of **CFD** models to analyze fluid flow and heat transfer phenomena.
- Developed comprehensive **CAD** models,**CFD**, and **FEA** simulations to study the behavior of coolant flow in nuclear reactors, and vibrations of the fuel rods, enhancing safety and efficiency.
- Applied various **AI** models to the generated data to optimize the existing design for best performance.
- Published research findings in reputable peer-reviewed journals and presented at international conferences.
- Collaborated with a multidisciplinary team of researchers to innovate and optimize thermal hydraulic systems.

TECHNICAL SKILLS

- Experienced in generating 3D CAD models and simulating physical phenomena of fluid flow, heat transfer, stresses, and vibrations using **ANSYS Workbench (Fluent, CFX, Structural, Modal, Thermal)** and **Solidworks**.
- Intermediate expertise in using **Deep Learning** and **Machine Learning** models to evaluate and understand data, improve CFD and structural simulations, and optimize engineering solutions by using Python Language.

SOFT SKILLS

- Leadership: Successfully led and mentored teams of B.Tech and M.Tech students on various projects, providing guidance and fostering a collaborative learning environment.
- Communication: Delivered lectures and presentations effectively, simplifying complex concepts for students and peers. Provided clear and constructive feedback to students.
- Analytical Thinking: Conducted in-depth research and data analysis, publishing findings in reputable journals and presenting at international conferences.

EDUCATION

- PhD, Mechanical Engineering (Thermal Hydraulics), Delhi Technological University, 2017-2024
- M.Tech, Nuclear Engineering, Delhi Technological University, 2013-2015 (7.59/10)
- B.Tech, Mechanical Engineering, Punjab Technical University, 2008-2012 (68.43%)

CERTIFICATIONS

- Python for Everbody, by the University of Michigan, on Coursera.
- Machine Learning, by the University of Washington, on Coursera.
- Deep Learning, by Andrew Ng, on Coursera.
- Applied Data Science with Python, by the University of Michigan, on Coursera.

ACHIEVEMENTS

- Honored by commendable research excellence award at Delhi Technological University in 2024.
- Published **11** research papers (**6** research papers in international journals, **4** of which are indexed in SCI, and **5** research papers in international conference proceedings indexed in Scopus).
- Presented **6** research papers in international conferences.
- Act as an organizing member in **3** international conferences and **1** short-term training program at Delhi Technological University in 2017-2024.

- Participated in **17** different workshops, seminars, training, and faculty development programs.
- Completed **3** different industrial training on site of total **32** weeks in thermal power plants and tractor assembly plant.
- Life Member of the Indian Society for Heat and Mass Transfer (ISHMT).

RESEARCH PROFILE

Gaurav Kumar is a highly skilled researcher with a Ph.D. in Thermal Hydraulics and extensive expertise in Computational Fluid Dynamics (CFD). His research focuses on developing advanced CFD models to analyze fluid flow and heat transfer phenomena, particularly in nuclear reactors. He has a strong background in thermal hydraulics, with experience in conducting simulations to enhance the safety and efficiency of reactor cooling systems. He has successfully collaborated with multidisciplinary teams and mentored B.Tech and M.Tech students on various CFD-related projects. He is also proficient in applying AI and machine learning models to optimize simulations and engineering solutions. His research contributions are well-recognized, with publications in peer-reviewed journals and presentations at international conferences.

PAPERS IN INTERNATIONAL JOURNALS

1. Mandia, V., Sharma, V., Chandra, Y., **Kumar, G.**, & Singh, R. K. (2024). Optimizing Winglet Cant Angle For Enhanced Aircraft Wing Performance Using CFD Simulation and Hybrid ANN-GA. International Journal of Numerical Methods in Fluids. (Published) **(SCIE)**
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5. **Kumar, G.**, & Singh, R. K. (2023). Supercritical water flow in heated wire wrapped rod bundle channels: A review. Progress in Nuclear Energy, 158, 104620. **(SCIE)**
6. Tyagi, A., **Kumar, G.**, & Singh, R. K. (2023).Thermal-hydraulic Behaviour of Corrugated Pipe Configurations. Journal of Polymer & Composites. **(ESCI)**

PAPERS IN INTERNATIONAL CONFERENCE PROCEEDINGS

1. Y. Sharma, A. Tyagi, **G. Kumar** and R. K. Singh, "Comparative Analysis of Sensor Configurations for Blood Flow Predictions via Physics-Informed Neural Networks," 2024 4th International Conference on Innovative Practices in Technology and Management (ICIPTM), Noida, India, 2024, pp. 1-6, doi: 10.1109/ICIPTM59628.2024.10563919. **(Scopus)**
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4. M. Dwivedi, **G. Kumar** and R. K. Singh, "A Numerical Study of Thermal Management of single cylindrical LiFePO4 battery," 2023 3rd International Conference on Innovative Practices in Technology and Management (ICIPTM), Uttar Pradesh, India, 2023, pp. 1-6, doi: 10.1109/ICIPTM57143.2023.10118228. **(Scopus)**
5. U. Verma, S. Yadav, **G. Kumar** and R. K. Singh, "Comparison of the Heat Transfer Capability of Various Nanofluids in a Flat Plate Solar Collector," 2023 9th International Conference on Electrical Energy Systems (ICEES), Chennai, India, 2023, pp. 366-370, doi: 10.1109/ICEES57979.2023.10110182. **(Scopus)**

OTHER SKILLS

Software Ansys Workbench (Fluent, CFX, Structural, Modal, Thermal); Solidworks; NX CAD; Autodesk Inventor and AutoCAD.
Programming Python.
Languages *English*: Professional Proficiency, *Hindi*: Native.