# HEAT TRANSFER OF FUEL ASSEMBLIES WITH FOUR RODS

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Heat transfer regimes are observed with water at supercritical conditions flowing in vertical channels such as rod bundles are analyzed. The test section consists of two channels separated by a square steel assembly box with rounded corners. Water flows downward in the first channel and then turns upward in the second channel to cool the 2 × 2 rod bundle installed inside the assembly box. The bundle consists of four heated rods. Effects of various parameters on heat transfer behavior inside the 2 × 2 rod bundle are similar to those observed through Ansys. In this paper, we will study the thermohydraulic processes of rod bundles in the region of near-critical and supercritical parameters. The purpose of the work is to compare our calculated results with the experimental data and the results obtained at Ansys, to provide recommendations on the choice of criterion equations Nu for calculating the coefficients of heat transfer and heat transfer from the rod to water at supercritical pressure.

Currently, there is an international GIF IV program (Generation IV International Forum), which formulates the basic concepts for the development of six new types of IV generation reactors (mainly fast neutron reactors with the possibility of implementing a closed fuel cycle). Of greatest interest are fast reactors cooled by liquid sodium or lead, and water-cooled reactors at supercritical pressure of 25 MPa (SCWR), which allow combining the design of a reactor with pressurized water (VVER) and a boiling reactor (RBMK) in a single concept and increase the efficiency (44 % and more). Developments of this type of reactor are underway in 15 countries, including Russia. Development of a reliable method for prediction of heat transfer to fluids with highly variable properties is among the critical engineering problems to be solved in designing promising nuclear power installations with supercritical pressure water [1]. An examination [2] of distributed exploratory information and the specifics of conveyances of the wall temperature and the Heat transfer coefficient along a Heated tube enbaled an solution to be made that the flow and heat transfer regimes appeared in tries different things with liquids at close basic circumstances could be separated into typical, deteriorated, and further developed heat transfer regimes. Kurganov et al. [3] thought about that the distinctive feature of heat transfer regimeswas that conditions of the heat transfer coefficient on the administering boundaries. [3, 4] of numerous publication on heat transfer regimes to the critical point exhibit that, as of now, the typical Heat transfer systems are concentrated on best of all. Taking into account the previously mentioned conduct of Heat transfer in fluids under supercritical circumstances correlations for anticipating heat transfer in these systems are typically gotten from experimental information utilizing the strategy used in [5–7]. The summed up correla-tions are created in view of the demonstrated heat transfer coefficient for turbulent flow a fluid with consistent properties. The utilization of supercritical liquids in various cycles isn't new, and is really not a human development. (Stringently talking, a supercritical fluid is a fluid at pressure and temperatures that are higher than the thermodynamic critical values. The SCWR is a once-through type water cooled reactor working over the critical pressure of water (22,1 MPa) and providing supercritical pressure steam at a high temperature to the turbine framework. The plant framework is supposed to accomplish higher thermal efficiency and an easier framework than the current nuclear power stations. Research exercises are progressing overall to foster high level nuclear power stations with SCWR (Oka and Koshizuka2000 [8]; Yoo et al., 2005 [9]; Kamei et al., 2006 [10]; Oka et al., 2007 [11]. One of the primary elements of supercritical water is areas of strength of its thermal physical properties nearby the pseudo-critical line. This enormous variety of thermal physical properties brings about a not usual flow and heat transfer conduct. So the dependable information on the thermal hydraulic conduct at reactor pertinent circumstances is vital for the plan of the SCWR core. Investigations of thermal hydraulic driven conduct of supercritical liquids have been performed since the 1950s. The exploratory and hypothetical investigations on heat transfer at supercritical pressure circumstances were inspected by a few authors.

The test segment comprises of two channels isolated by a square steel gathering box with adjusted corners. Water flows downward in the first channel and then turns upward in the second channel to cool the 2 × 2 rod bundle introduced inside the assembly box. Impacts of different boundaries on heat transfer behavior inside the 2 × 2 rod bundle are alike to those observed in tube or annuli.

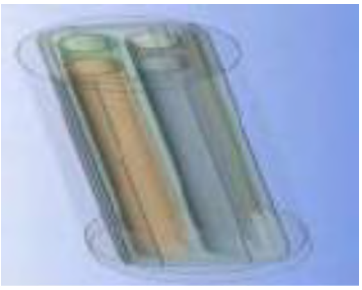

*Fig. 1. 3D Model of test section*

A stainless tube with 38 mm in inner diameter fills as pressure vessel. There are two channels in the test segment, isolated by a square assembly box with round corners. Water flows downward in the first channel between the pressure vessel and the assembly box, and then flows upward in the second channel inside the assembly box after mixed at the bottom of the test section as shown in figure 1. In many cases it's comfortable to have basic equations for assessment of heat transfer coefficients. Below is a assortment of suggested condition equations for our geometries as well as a few equations for heat transfer processes with change of phase [12, 13]. To decide attributes and conditions of existence of different heat transfer regimes in supercritical pressure water, we analyzed the experimental data for rod bundles. In view of the information estimated in the experiments at various sections along the length of heated channels, we preliminary predicted local Nusselt numbers *Nu* which has been utilized below. The following data ought to be prepared as a result of the calculation.

Heat transfer along the length of the rod

$$Q = m\, c_p\,( t_f' - t_f), \tag{1}$$

where, $m$ is the mass flow rate (kg/s), $c_p$ is specific heat, (J/kg·K), $t_f$ is the fluid temperature, (°C), $t_f'$ is the fluid temperature after a certain interval (°C).

$$Q = q \cdot S\,, \tag{2}$$

where, $q$ is the heat flux, W/m$^2$, $S$ is the size of the rod, mm$^2$.

$$t_f' = t_f + \frac{Q}{m \cdot cp}\,. \tag{3}$$

Average temperature

$$t_{avg} = \frac{t_1 + t_2}{2}\,. \tag{4}$$

The following are treated and there is equivalent diameters used for estimating the Nusselt and the Reynolds number, beside$D_h$. Here is one taken.

Hydraulic diameter

$$D_h = \sqrt{\frac{4 \cdot F_a}{\pi}}\,, \tag{5}$$

where, $F_a$ is Flow area, (m$^2$)

Flow area (cross sectional area)

$$F_a = \frac{\pi \cdot D^2}{4}, \tag{6}$$

where, $D$ is the diameter of the pipe in meters, (m)

Reynolds number is given by

$$Re = \frac{VD_h}{v}, \tag{7}$$

where, $Re$ is the Reynolds number, which is unitless, $V$ is the velocity in meters-per-second, (m/s), $D_h$ is the hydraulic diameter of the pipe in meters, (m),ν is kinematic viscosity, (m$^2$/s).

Mass flow rate

$$m = m_f \cdot F_a\,, \tag{8}$$

where, $m_f$ is the mass flux rate (kg/m$^2$s)

Velocity

$$V = \frac{m_f}{\rho}, \tag{9}$$

where, $\rho$ is the fluid density, (kg/m$^3$)

Prandtl number

$$Pr = \frac{\mu\, c_p}{\lambda}. \tag{10}$$

where, $\mu$ is dynamic viscosity, (N·s/m$^2$), $\lambda$ is the thermal conductivity, (W/m·K).

Nusselt number Correlations

The final expression recommended for predicting heat transfer has the form

$$Nu = 0.027\, Re^{0.8} Pr_{t_f}^{\frac{1}{3}} \left(\frac{\mu_{t_f}}{\mu_{t_w}}\right)^{0.14}, \tag{11}$$

where, $\mu_{t_f}$ is dynamic viscosity at $t_f$, , (N·s/m$^2$), $\mu_{t_w}$ is dynamic viscosity at $t_w$ , (N·s/m$^2$)

The Nusselt number is calculated as

$$Nu_D = \frac{\alpha \cdot D_h}{\lambda}, \tag{12}$$

where, α is the heat transfer coefficient, (W/m²K).

Heat transfer coefficient is calculated by using equation

$$\alpha = \frac{\lambda \cdot Nu_D}{D_h}. \tag{13}$$

The range of parameters for the calculation of efficiency was taken as pressure $P$= 25 MPa, heat flux of $q$ = 600 kW/m$^2$ and mass flux of 800 kg/m$^2$s. The determined efficiency fits well with the experimental data. The level of absorbed heat by the water in the primary channel is shown in figure 2. The heat transfer between the two channels decreases with the increment of inlet temperature. This is brought by the little temperature difference between the two channels when the fluid temperature at the inlet. The determined results shows close outcomes with published data [14].

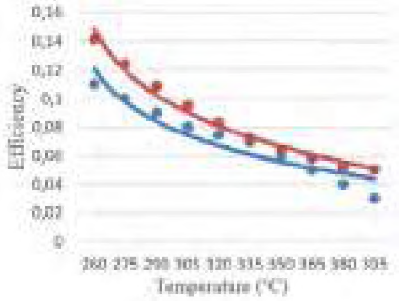


*Fig. 2 Comparison of Heat transfer through the channel (Published ● and ● calculated )*


*Fig. 3. Variation of wall temperatures (● ,●) and fluid temperature(●) along the length*

Figure 3 shows wall temperature in the two channels along the length with fluid temperatures. The functioning pressure is 25 MPa. The mass flux is 800 kg/ m² s and the heat flux is 600 (kW/m²). The water entering the test area goes downward in the first channel. It is blended in the mixing chamber at the lower part of the test section. Then, it turns and goes upward on the subsequent channel. The wall temperatures in the top and lower part of the channel shows close understanding in the inner channel while in external channel shows a slight deviation. The wall temperature ranges from 286 °C to 368 °C and the fluid temperature ranges from 300 °C to 340 °C


*Fig. 4. Variation of HTC (●, ●)along the length of the rod*

The variations of heat transfer coefficient along the length is displayed in figure 4. The heat transfer coefficient shows a ordinary way of behavior. The heat transfer coefficient increases steadily in the inner channel and decreases in the outer channel. The ascent of HTC in inner channel is exceptionally mild contrasted with the HTC in outer channel. The HTC ranges from 9,8 (kW/m²K) to 12,46 (kW/m²K).


*Fig. 5. Comparison of ANSYS Fluid temperature ● and calculated ●*

Table 1. Results

| Z | $t_{f1avg}$ | $t_{f1}$ ansys | $t_{f1}$ Error% | $t_{f2avg}$ | $t_{f2}$ ansys | $t_{f2}$ Error % |
|---|---|---|---|---|---|---|
| 800 | 300,5258 | 300 | 0,00175 | 340,5931 | 361,2 | 0,057051 |
| 720 | 301,519 | 302,63 | 0,003671 | 336,6651 | 349,09 | 0,035592 |
| 640 | 302,4018 | 303,96 | 0,005126 | 333,2253 | 344,08 | 0,031547 |
| 560 | 303,1807 | 305,88 | 0,008825 | 330,1901 | 340,92 | 0,031473 |
| 480 | 303,8558 | 306,6 | 0,00895 | 327,5596 | 339,88 | 0,036249 |
| 400 | 304,4271 | 308,99 | 0,014767 | 325,3338 | 334,92 | 0,028622 |
| 320 | 304,8944 | 310,44 | 0,017864 | 323,5128 | 332,96 | 0,028374 |
| 240 | 305,258 | 311,88 | 0,021233 | 322,0963 | 325,72 | 0,011125 |
| 160 | 305,5176 | 312,48 | 0,022281 | 321,0846 | 320,18 | -0,00283 |
| 80 | 305,6734 | 312,44 | 0,021657 | 320,4776 | 316,21 | -0,0135 |
| 0 | 305,7833 | 312,57 | 0,020345 | 319,7342 | 315,42 | -0,0120 |

## Conclusion

In this work we have compared our calculated results with the experimental data and the results received with Ansys, to provide give suggestions on the model involving criterion equations *Nu* for calculating the coefficients of heat transfer and heat transfer from the rod to water at supercritical pressure. The results received through calculation estimation show close relations with the ones acquired through ANSYS.

In comparison with the fraction of heat transfer from the second channel to the first channel with the test conditons, there is a slight deviation. But, it decreases with the inlet water temperature. The impacts of systerm parameters including heat flux, mass flux and pressure on the heat transfer of supercritical water in the bundle are close and very like to those observed.

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