Analysis of the results of thermophysical experiments with multi-rod bundles. Elaboration of the SCP thermalhydraulic test facility design

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Abstract. The present report is devoted to the analysis of the thermophysical properties of supercritical water in the cores of supercritical water-cooled reactors (SCWR) and results of the experiments with multi-rod bundles cooled with supercritical fluid (mainly – water) flows carried out in different countries. The most important results and specific heat transfer regimes observed in the experiments were summarized. On the basis of the experimental data analysis the design of a thermophysical test facility for full-scale experiments for the determination of the heat and mass exchange regimes and supercritical water hydraulic parameters in multi-rod bundles was elaborated. The need for the experimental data for the proper description of the said processes was stated, and the importance of obtaining them was underlined. The list of the controlled parameters and experiments was created.

Introduction

In 2001, the leading countries in nuclear industry launched a project 'Generation IV' on the creation of a revolutionary Generation IV nuclear reactor. Taking into account the advantages of the water-cooled reactors technology, most of the countries participating in the GIF project, apart from developing other concepts, are engaged in the development of the supercritical (SC) water-cooled reactors. Since the end of the 1960-s complex studies focused on the creation of such reactors have been carried out by many Russian research centers under the supervision of the NRC 'Kurchatov Institute' that launched two SCP reactor projects, and to the current moment a lot of planned works under the project have been completed.

Initial transition from the subcritical coolant parameters to SCP ones was realized at thermal power plants operating on fossil fuels. However, despite considerable financial supporting in different countries, the process of TPPs transition to the SCP regime took more than 10 years. The main reason for that is a complex dependence of the near-critical fluid thermophysical properties on the regime parameters.

The experiments on heat transfer in pipes demonstrated a rapid temperature increase (100 °C and more) across a short length of the test section that in some cases caused its destruction. Such regimes were called 'deteriorated heat transfer (DHT) regimes'. It was established that transition to the
DHT regime cannot be explained similarly to the departure from nucleate boiling (as it was expected by some researchers) [1]. Moreover, in some experiments pressure pulsations with significant amplitudes were observed along the channel length. Such a phenomena was called 'pressure pulsation regime' (PPR). Furthermore, the experiments demonstrated that heat transfer was significantly affected by the depositions on the inner surfaces of the pipes. In order to solve that problem many research and development activities were performed, and the proposed key idea constituted in the shifting of the pseudo-phase transition point into the region of low heat flux values.

For NPPs transition to the SCP regime only a small volume of the thermophysical experimental results for SCP boilers appeared to be useful because of the difference in the coolant channels configurations. In the modern SCP boilers heat transfer takes place in pipes 20–40 mm in diameter while the hydraulic-equivalent diameter of fuel assembly (FA) subchannels is much smaller (from 3 to 10 mm), that results in a significant divergence between the hydrodynamics of coolant flows in the subchannels formed by fuel rods in FAs and hydrodynamics of coolant flows in pipes. Another problem typical for nuclear reactors (as compared to conventional boilers) is that the pseudo-phase transition point cannot be shifted into the region of low heat flux values comparable with those for boilers. All these facts highlight the need for carrying out the relevant experiments. The results of the experiments with multi-rod bundles carried out in Russia and abroad demonstrated that heat transfer processes in them were very complicated.

Therefore, determination of the data on heat transfer in supercritical and pseudo-phase transient regions employing geometry models which simulate the cores of the designed reactors becomes especially relevant. The said data can be obtained using a thermophysical test facility, and elaboration of its design represents one of the key challenges.

First of all, for elaborating the thermophysical test facility design the simulated parameters are to be specified. The said parameters, in their turn, should simulate the specific parameters of the reactor cores.

Supercritical water-cooled reactors can be classified into the following main groups [2]:

- reactors operating at near-critical parameters with the temperature at the core outlet slightly higher than the critical temperature;
- reactors operating at subcritical and supercritical parameters. For such reactors the coolant temperature at the core outlet is much higher than the critical temperature;
- reactors which operating parameters fall within the region above the pseudo-phase transition.

**Analysis of the previous thermophysical experiments with multi-rod bundles**

By 2008, only two experiments with multi-rod bundles were carried out worldwide. The said experiments included ones conducted at the JSC 'All-Russian Thermal Engineering institute' (VTI) and at NRC 'Kurchatov Institute'.

In the experiment carried out at VTI in 1977 [3] a flow-through test facility filled with the fluid bled from the TPP SCP boiler was used. The test facility had a vertical channel with a bundle of fuel rod simulators located therein. The test section represented a 7-rod bundle of ribbed tubes with an outer diameter of 5.2 mm and length of 500 mm close-packed into a hexagonal jacket with thin walls. Each of the tubes spaced at 400 mm intervals was wrapped with four ribs 0.6 mm in height and 1 mm in width. Heating of the channel was provided by passing constant current. The outer tubes of the bundle were used as gauge lines for measuring the pressure drop between the six bleeds. The wall temperature was measured only for the central tube at ten test points along its length using a movable thermocouple probe. The parameters of the experiment are given in Table 1.

| Table1. Main parameters of the experiments (VTI) |
|-----------------|-----------------|-----------------|
| Pressure, MPa | Heat flux q, MW/m² | Mass flow rate ρw, kg/(m²·s) |
| 24.5 | 4.7 | 500 – 4000 |
In the VTI experiments the following results were obtained:

- even for high \( q/\rho_w \) values no DHTs were detected;
- for mass flux values over 2,000 kg/(m²·s) at relatively high heat fluxes regimes with low-frequency (period 25–30 s) pressure pulsations 5 MPa in amplitude were observed.

Another experiment was carried out at the NRC 'Kurchatov Institute' using the test facility (TPZ) [4]. The test facility represented a closed loop with a test section inside it. The maximum length of the bundle of fuel rod simulators was 880 mm. The bundles with a circular cross-section differed from one another in the number (7 and 19) and diameter of fuel rod simulators (5.6 and 4.0 mm respectively). The specially shaped simulators represented rods with spherical recesses (1 mm in diameter) pressed therein in a staggered manner. The main parameters of the experiments carried out at the NRC 'KI' are given in Table 2.

| Pressure, MPa | Heat flux, MW/m² | Mass flow rate \( \rho w \), kg/(m²·s) |
|--------------|------------------|--------------------------------------|
| 23.5; 30.5   | 0.18 – 4.5       | 350 – 5000                            |

In the NRC 'KI' experiments the following key results were obtained:

- for \( t_f > t_m \) and \( q/\rho w > 0.8 \text{kJ/kg} \), the measured wall temperature (averaged along the bundle length) was higher than the calculated value with a wide variation in the experimental data for different fuel rod simulators, that can be caused by the accumulation of depositions on the simulators surfaces and their removal therefrom;
- in some cases for high mass flux values in the region \( q/\rho w > 0.6 \text{kJ/kg} \) at constant regime parameters the simulators temperature increased with time;
- in the regimes with growing \( t_w \) after discharging water immediately upon disconnecting the electrical load, iron and copper oxides were observed on the fuel rod simulators surfaces;
- sometimes the temperature distribution over the simulators changed smooth, while sometimes its changing was abrupt;
- for the tested regime parameters range at the mass flux value from 350 to 5,500 kg/(m²·s) and \( q/\rho w \) up to 1.7 kJ/kg no DHTs were detected.

There were some experiments focused on the investigation of heat transfer to supercritical water and hydraulic resistance of the bundles, which were conducted abroad. Many studies using vertical bundles with a square cross-section (2x2) were carried out in China.

A research group from the Xi’an Jiaotong University conducted the experiment for the determination of the heat transfer coefficient in a bundle of fuel rod simulators cooled with upward flow of supercritical water. In 2014 a paper on the experiments with a smooth bundle was issued [5], and in 2016 a paper on the experiments with a bundle with wire-wrap spacers was issued [6]. The parameters of the experiments are given in Table 3.

| Pressure, MPa | Heat flux, MW/m² | Mass flow rate \( \rho w \), kg/(m²·s) |
|--------------|------------------|--------------------------------------|
| 23 – 28      | 0.2 – 1          | 350 (400)\(^a\) – 1000 |

\(^a\)for a bundle with wire-wrap spacers.

The test section represented a pipe in which an electrically insulating channel with a square cross-section and rounded edges was located wherein the bundle was installed. The bundle geometry
simulated that of a fuel assembly for a HPLWR-type reactor. The outer diameter of a fuel rod simulator was 8 mm, the simulators were spaced 9.44 mm apart, the P/D coefficient was equal to 1.18, hydraulic-equivalent diameter was 4.32 mm and heated length was 600 mm. Heating was provided by supplying alternating current to the upper and lower electrodes. The arrangement of the test section and thermocouples positions are shown in Fig. 1.

The test section represented a closed loop. Prior to its supply into the test facility, water passed through a regenerative heat exchanger for its preliminary heating using the heat of water discharged downstream of the test section. Then water was heated up to the operating inlet temperature by means of a preheater.

![Fig. 1. Longitudinal and transverse cross-sections of the test section (on the left) and thermocouples positions (on the right) [6].](image)

The experiments demonstrated a non-uniform temperature distribution over the simulators perimeter with its maximum on the subchannel side adjacent to the channel angle and its minimum on the side facing the central subchannel. The difference between the minimum and maximum temperatures changed depending on the heat flux and mass flux. For a bundle with wire-wrap spacers the result generally was the same, excepting that the peak of heat transfer coefficient shifted towards lower coolant enthalpies, wall temperature decreased (especially in the subcritical region) and average heat transfer coefficient increased [6].

A non-uniform temperature distribution along the perimeter of the simulators wall was associated with the non-uniformity in the coolant mass flux distribution over the subchannels cross-sections: as the hydraulic-equivalent diameters of the central and outer subchannels were different it caused the said non-uniformity [7].

The experiments with bundles with a triangular lattice were performed at the Kyiv Polytechnic Institute (KPI) in 2009 – 2013. The studies were carried out for a single fuel rod simulator wrapped with four ribs and for bundles with three and seven such simulators [8 – 11] (see Fig. 2). For each bundle geometry configuration fuel rod simulators represented a cylindrical rod with a heated length of 485 mm. Four helical ribs were wound over the rods with 400 mm pitch. Apart from the ribs stainless steel fins of 0.1 mm thickness were welded to the simulators. The fins acted as spacers and provided a gap of 1 mm between the simulators and dielectric inserts in the channel. The dielectric inserts performing the functions of displacers and forming the channel shape were installed into the metal pipe bearing the total coolant pressure.
The main parameters of the experiments are given in Table 4.

![Fig. 2. Longitudinal cross-section of the test channel with 1, 3 and 7 fuel rod simulators [11, 12].](image)

| Pressure, MPa | Heat flux, MW/m² | Mass flow rate ρᵢw, kg/(m²⋅s) |
|---------------|------------------|-------------------------------|
| 22.6, 24.5, 27.5 | 0.5 – 4.58       | 800 – 3000                    |

The test facility represented a loop designed for the maximum operating pressure and temperature of 28 MPa and 700 °C (for the test section located inside a pressure pipe filled with nitrogen at working pressure) respectively. Chemically desalinated water was used as coolant. Heating of the simulators was provided directly by alternating current.

The experiments demonstrated that in case of DHT the decrease of heat transfer coefficient for the outer fuel rod simulators was the smallest. The similar effect was observed for the simulators wall temperature (for the outer fuel rod simulators it was lower). The dependence of the local heat transfer coefficient on the flow history was established as well.

From the comparison of the experimental data for 4- and 7-rod bundles it can be concluded that the maximum and minimum temperatures on the fuel rod simulators perimeter have different distributions. Such results are, probably, associated with the influence of the cold non-heated channel wall on the outer subchannels.

Another noteworthy experiment is that carried out at the State Scientific Centre of the Russian Federation Leypunsky Institute for Physics and Power Engineering (IPPE JSC). In the experiment freon R-12 with relatively low critical parameters was used as coolant. The experiment was performed with the 7-rod bundle with three plate spacer grids (SG) (see Fig. 3).

The experiment yielded the following results [13]:

- detection of DHTs in 15 out of 20 cases that again proved that deteriorated heat transfer regimes were typical not only for pipes but also for more complex geometry configurations (such as bundles);
- DHTs mostly occurred at the heating zone outlet (at a distance of 0.8 – 1 m from the inlet into the heating zone);
- in several cases two regions of DHT establishment were observed (one spaced approximately 0.2 m apart from the heating zone inlet and another near its outlet);
- spacer grids improved heat transfer conditions;
- the currently used one-dimensional correlations are not applicable for the prediction of the obtained data on DHT with an error lower than ±50%.
Summarizing the abovementioned data, to the present moment two experiments with SCP water were carried out in Russia using different (flow-through and circulation) test facilities. Different anomalies of heat transfer processes were detected. The said anomalies can be caused, e.g., by the fact that SCP-water in the NRC ‘KI’ experiment circulated in a circulation test facility and accumulated corrosion products (mainly iron from the loop and copper from current-carrying elements) that further resulted in the formation of the oxide depositions on the fuel rod simulators surfaces. The data obtained from the IPPE experiment can again suggest the fact that the experiment should be carried out with bundles which geometry configuration provides the fullest possible compliance between its hydrodynamics and that in the core of SCP-reactors. For instance, length is one of the critical geometry parameters, because hydrodynamic stabilization of the flow should take place within the heated length of the bundle as this process potentially affects the DHT occurrence (especially in case of installing intensifying spacer grids suggested by the Russian projects), that is substantiated by the KPI experience.

In accordance with the European design concept, heat transfer intensification and spacing are provided by means of wire-wrap spacers, while in the SCP-1700 reactor project the use of intensifying spacer grids in combination with the displacement elements of different shapes installed in the subchannels in-between the fuel rods is recommended. Therefore, for testing the Russian design solutions the simulation of hexagonal geometry is required.

**Required parameters of the thermophysical test facility**

From the analysis of the cores of the designed SCP-reactors the basic parameters important for the determination of the thermophysical test facility parameters can be obtained.
Table 5. Average basic parameters of the designed SCP reactors important for simulation.

| Reactor                | Inlet/outlet temperature, °C | Heat flux, kW/m² | Subchannel hydraulic-equivalent diameter, mm | Fuel rod outer diameter, mm | Interval between fuel rods, mm |
|------------------------|------------------------------|------------------|---------------------------------------------|-----------------------------|-------------------------------|
| V-670 SKDI             | 365/395                      | 9.47             | 10.965                                      | 8.0                         | 3.7                           |
| PSKD-600               | 388-500                      | 15.0             | 8.4 – 11.5                                  | 8.4 – 11.5                  | >1                            |
| Single-circuit SCP     | 280-540                      | 9.7 – 15.0       | 8.0-10.7                                    | 8.0-10.7                    | ~1 – 1.4                      |

Thus, to simulate all the designed reactors the test facility should have the following parameters:
- coolant pressure 23.5 – 25 MPa;
- coolant temperature 360 – 540 °C;
- average heat flux per a surface unit from a fuel rod simulator 8 – 15 kW/m²;
- maximum linear heat flux 40 kW/m;
- mass flux 800 – 2,000 kg/(m²s);
- maximum heated length of a fuel rod simulator 4,200 mm;
- upward and downward coolant flows.

It is important to carry out the experiments at both upward and downward coolant flows as the flow direction is of great significance. DHT can occur upon the transition to another flow regime with the formation of two DHT regions along the channel length. In previous studies [10] it was established that the effect of free convection is different in upward and downward coolant flows, and heat transfer coefficient for downward coolant flow is smaller.

The heat network of the test facility should include means (such as preheater and regenerative heat exchanger) for preliminary coolant heating in order to ensure consistently high temperatures at the test section inlet.

The test bundle should have the following parameters:
- hydraulic-equivalent diameter 3 – 11 mm;
- diameter of fuel rod simulators 8 – 11.5 mm;
- minimum interval between fuel rod simulators 1 – 4 mm.

The need for the experiments with multi-rod bundles is associated with the efforts to reduce the wall influence on heat transfer and hydrodynamic processes in the central subchannels of the bundle. Establishment of deteriorated heat transfer regimes (or departure from nucleate boiling) mostly takes place in the inner subchannels adjacent to the guide channels for absorbing rods (AR) of the control and protection system (CPS) as these passages are the narrowest.

For single-row bundles (with 2x2 rod array or triangular lattice) the subchannels are inner and outer at the same time. For a bundle with 3x3 rods array the inner subchannels are affected by the channel wall.

In the presence of heat exchange intensifying spacer grids axial coolant flow swirls form. In the bundles with small amount of fuel rods it causes coolant flow redistribution over the subchannels with its increasing in the outer subchannels and decrease of the mass flux in the central subchannel, that can promote the occurrence of DHT (departure from nucleate boiling).

Conclusion
The analysis of the experimental results leads to the conclusion that experiments with multi-rod bundles (with at least two fuel rod rows and the length ensuring hydrodynamic stabilization of the
flow within the heated length) as well as the installation of several spacer grids with the axial interval therebetwen similar to that for a commercially used FA are strongly required.

In the experiments on heat transfer the recommended number of the wall temperature measurement points is 20 per a cross-section of fuel rod simulators at the height level under consideration, and the determination of the local temperature at the point corresponding to the minimum distance between fuel rod simulators within the same cross-section is recommended as well.

Special attention should be paid to accurate temperature monitoring because in addition to the abovementioned deteriorated and improved heat transfer regimes and pressure pulsations discovered in the VTI experiments, other two specific regimes [15] (pseudo-boiling and pseudo-film boiling) can take place.

The importance of the studies on the fuel rod bundle hydraulics is explained by the need for the justification of the implementation of a hydraulic-equivalent diameter for the channels formed by close-packed fuel rods [16]. In case of turbulent flow with rapidly changing properties, in the channels with complex geometry configurations, a secondary flow can be formed (this case requires a separate study).

Thus, the following priorities for experimental and numerical studies focused on the analysis of the thermal and hydraulic processes critical for the SCP reactors operation can be formulated.

At the first stage, thermal and hydraulic characteristics of the core should be investigated for two different cases:

- for a multi-rod bundle with heat flux uniform along the test section length;
- for a multi-rod bundle with heat flux non-uniform along the test section length (similar to the heat flux from the fuel rods).

The need for the experiments with the bundles with non-uniform axial and radial power density distributions is explained by the fact that the non-uniformity of power density distribution over the bundle can significantly affect the ranges of the establishment of specific heat transfer regimes.

Another very significant aspect is coolant chemical composition monitoring. SCP water represents a perfect solvent, however, below the pseudo-phase transition point the solubility rapidly decreases that leads to the precipitation of impurities on the simulators surfaces that, in its turn, causes changes in heat transfer within that area. Moreover, the presence of impurities in water changes its pseudo-phase transition temperature that can result in changes of the DHT occurrence range.

The second stage of thermal and hydraulic studies is related to coolant cooling processes in a steam generator (investigations on fluid condensation in the steam generator intertubular space are important for the justification of the condensation process stability).

At both investigation stages the temperature and chemical composition of coolant and concentration of the substances dissolved therein should be equal to the corresponding values expected in the reactor.

Thus, carrying out the experiments with bundles of fuel rod simulators cooled with SCP water is required in order to determine the following characteristics:

- conditions and ranges for the occurrence of deteriorated and improved heat transfer regimes and pressure pulsations in multi-rod bundles;
- effect of the coolant flow direction and spacer grids or intensifying spacer grids on the conditions and ranges of the occurrence of the said specific heat transfer regimes;
- hydraulic resistance for SCP water flows in cases of complex geometry configurations and presence of SGs or IGs.

Providing that the said characteristics are unknown, the creation of the cores of new-generation nuclear reactors ensuring thermal-hydraulic compatibility and thermal-mechanical reliability is impossible.
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