Compilation of Supercritical Heat Transfer Data through the IAEA Coordinated Research Project

L.K.H. Leung[1] and K. Yamada[2]
[1] Atomic Energy of Canada Limited, Chalk River, Ontario, Canada
(leungl@aecl.ca)
[2] International Atomic Energy Agency, P.O. Box 100, 1400 Vienna, Austria

Abstract

A large database on supercritical heat transfer has been compiled through the collaboration of 15 organizations under an IAEA-organized Coordinated Research Project on "Heat Transfer Behaviour and Thermo-hydraulics Code Testing for SCWRs". It covers heat-transfer data obtained with tubes, annuli, and small bundle assemblies of various orientations in water and surrogate fluids over a wide range of flow conditions. Part of the database has been applied in developing advanced prediction methods, while others are used for validation of existing prediction methods. Some separate effects on supercritical heat transfer are also described.

1. Introduction

Innovative design concepts of Super-Critical Water-cooled Reactor (SCWR) are of main interest in both developing and industrialized countries. Their interest attributes to the high thermal efficiencies (44-45%), improved economic competitiveness, and enhanced safety characteristics of these concepts. The International Atomic Energy Agency (IAEA) has initiated a Coordinated Research Project (CRP) on "Heat Transfer Behaviour and Thermo-hydraulics Code Testing for SCWRs" in 2008 to facilitate collaboration and interaction among its member states. Through this collaborating effort, all participants share innovative ideas and R&D information to expand the R&D capability and minimize unnecessary duplication in each member state. This has significantly expanded various databases for assessing existing prediction methods and/or developing advanced prediction methods to support the SCWR design.

Thermalhydraulics characteristics at supercritical water-flow conditions are required in support of the SCWR design and qualification of the fuel bundle and safety analyses. The lack of qualified experimental data on heat transfer for supercritical water flow has been identified as a significant risk to the SCWR design. This is due to the possible drastic deterioration of heat-transfer characteristic at the vicinity of the pseudo-critical point. Consequently, developing SCWR designs requires experimental data for the convective heat transfer from fuel to coolant covering relevant ranges of flow rates, pressures and fluid temperatures. The collection, evaluation and assimilation of existing and new data are necessary to establish accurate methods and techniques for the prediction of heat transfer in SCWR cores. The objective of this paper is to present the supercritical heat-transfer databases compiled from participants of the IAEA CRP.

2. Heat Transfer in Supercritical Pressure Water Flow

Experimental data obtained with tubes and bundles at supercritical water conditions are required for the development and validation of heat-transfer correlations. Several literature surveys (e.g., [1]) were performed and identified many sets of supercritical heat-transfer data with water flow in support of boiler applications. Correlations were developed using these data.

2.1 Tube Data

Tube data provide a fundamental understanding of supercritical heat transfer phenomena. These data were applied in developing most supercritical pressure heat-transfer correlations, which have been implemented into subchannel codes in support of the SCWR fuel design. Previous reviews of published information concluded that available information is limited for SCW flow in tubes [1]. In particular, most experimental data were obtained with large diameter tubes (some with ribbed tubes). Justification and validation are required to extend the application of correlations, based on these data, to the SCWR design and safety analyses. As indicated, a large supercritical heat-transfer database for water flow in tubes has been compiled from the CRP. Experimental parameters for individual dataset are tabulated in Appendix A for information.

The database of supercritical heat transfer in water contributed by Atomic Energy of Canada Limited [2] contains primarily experimental data obtained from studies listed in [1]. These databases are applicable for

- Assessing various heat transfer correlations and identifying suitable correlations for the required operating range of SCWRs,
- Developing prediction methods (in the form of either correlations or a look-up table) for supercritical heat transfer,
- Identifying experimental needs to fill missing ranges among available data, and
- Developing and validating fluid-to-fluid modeling techniques for heat transfer in the near critical and supercritical pressure regimes.

The ranges of parameters covered in the database are: tube diameters from 1.27 to 38 mm, pressures from 10.0 to 41.3 MPa, mass fluxes from 50 to 5424 $kg{\cdot}m^{-2}{\cdot}s^{-1}$, and heat fluxes from 60 to 9464 $kW{\cdot}m^{-2}$. Ranges of fluid temperatures and enthalpy, when described, appear similar to those of the previous database. Figure 1 illustrates the range of reduced pressure and reduced temperature covered in these data sets. Several data sets cover a wide range of conditions.

An extensive literature survey has been performed at the Shanghai Jiaotong University (SJTU) to establish a valid database for heat transfer investigations with fluids at supercritical pressures in vertical upward flow in vertical tubes and other channels [3]. Up to now, a total of 14150 data points for upward flow of water and surrogate fluids have been collected, checked, analyzed, categorized and put into a data bank. The water database consists of 8624 data points covering the typical parametric range of SCWR for some main thermo-hydraulic parameters. However, for certain combined conditions, existing data distribution is still not satisfactory for SCWR applications, especially for certain transient and accident conditions. For example, at the typical SCWR pressure of 25 MPa, the range of heat fluxes and mass fluxes covered in the database remains inadequate for transient or postulated accident analyses (see Figure 2). In addition, only few data are available at large ratios of heat flux to mass flux, where heat transfer deterioration is often encountered (see Figure 3). The distribution of data is considered to be "non-homogeneous"; this may lead to imbalance in information synthesis and data correlation.

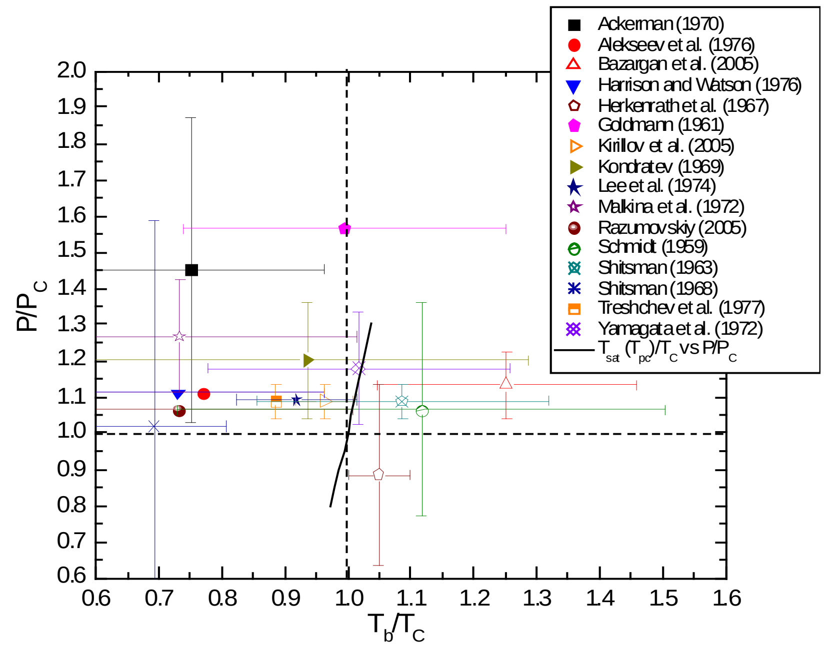

Figure 1 Range of Selected AECL Supercritical Heat-Transfer Database for Water Flow (see reference [1] for details).

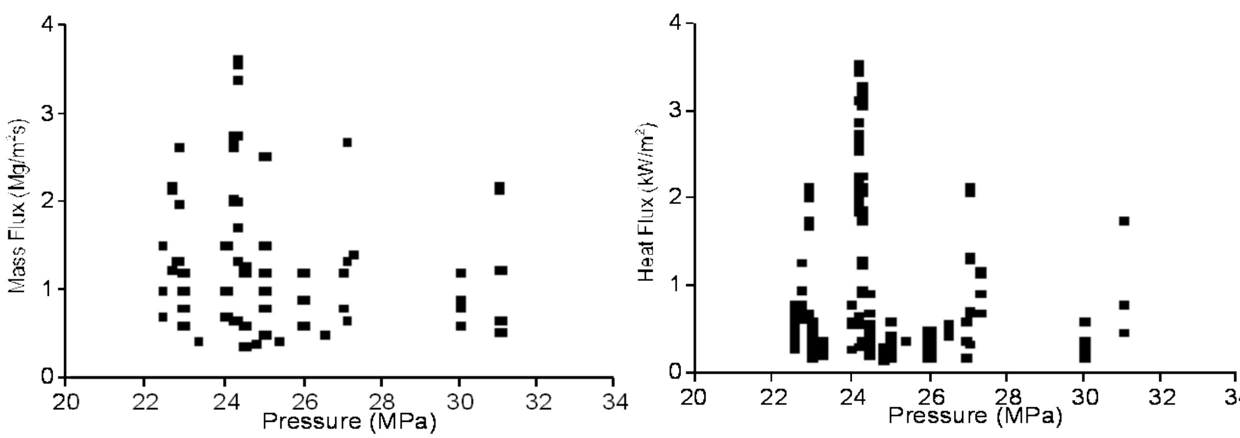

Figure 2 Ranges of Mass Flux, Heat Flux and Pressure Covered in the SJTU Supercritical Heat-Transfer Database for Water Flow.

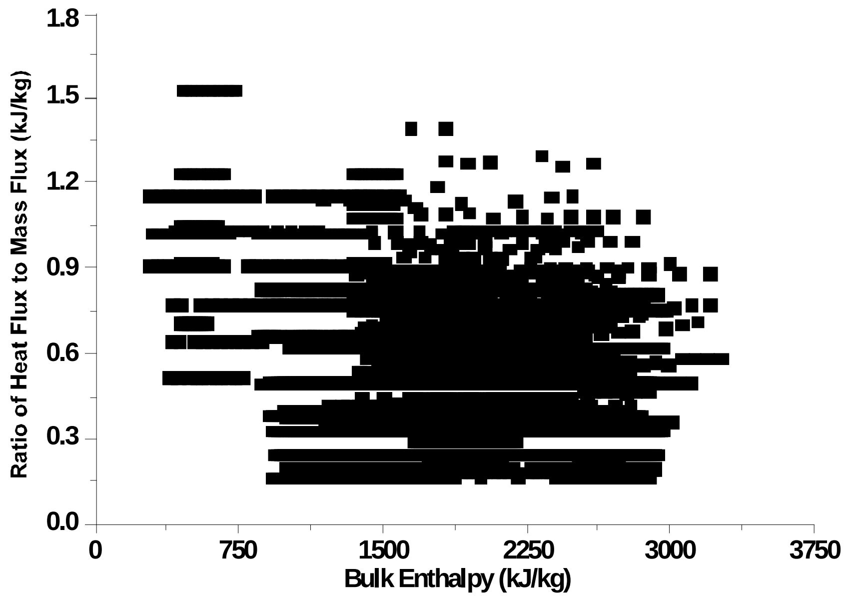

Figure 3 Ratios of Heat Flux to Mass Flux in the SJTU Supercritical Heat-Transfer Database for Water Flow.

The Institute of Physics and Power Engineering (IPPE) reviewed experimental research to supercritical water heat transfer in vertical and horizontal tubes, which was carried out in Russia from 1950 to 1980 [4]. Most of the research work was introduced to support thermalhydraulic calculations for the supercritical pressure boilers of fossil fuel power plant. Most publications presented experimental data in charts giving the wall temperature, T_w, along the heated length or as a function of bulk enthalpy. The IPPE database covers the range of pressures from 0.4 to 29.4 MPa, bulk-fluid temperatures from 20 to 500°C, bulk-fluid enthalpies from 400 to 3350 $kJ \cdot kg^{-1}$, mass fluxes from 0.17 to 10 $Mg \cdot m^{-2} \cdot s^{-1}$, and heat fluxes from 0.13 to 3.4 $MW \cdot m^{-2}$.

During early 1960s, the Nuclear Engineering Research Group at Manchester University became interested in heat transfer to fluids at supercritical pressures and initiated a series of experimental investigations, firstly using carbon dioxide and later water. As part of that research program, two large-scale test facilities were designed and constructed for detailed experimental studies using uniformly heated vertical tubes with upward and downward flows [5]. The water dataset was obtained with upflow and downflow inside a vertical 25.4-mm ID tube covering the mass-flux range from 132 to 1062 $kg \cdot m^{-2} \cdot s^{-1}$, inlet-fluid-temperature range from 150 to 320°C, and heat-flux range from 170 to 450 $kW \cdot m^{-2}$ at the pressure of 25 MPa.

2.2 Annuli Data

Recently, an experiment was performed at Xi'an Jiaotong University (XJTU) using an annulus test section to obtain supercritical heat-transfer data for validation of the supercritical heat-transfer correlation [6]. Both the inner and outer tubes of the test section were made of stainless-steel tubes.

The inner tube has an outer diameter of 8 mm and a wall thickness of 1.5 mm. It was heated with electric current over a heated length of 2 metres. The outer tube has an outer diameter of 16 mm and a wall thickness of 2 mm. Ceramic cylindrical spacers were installed at several axial locations to maintain the spacing between the inner and outer tubes. Thermocouples were installed inside the test section to measure the inner-wall temperature. Figure 4 illustrates schematically the annulus test section design at XJTU for supercritical pressure heat-transfer tests.

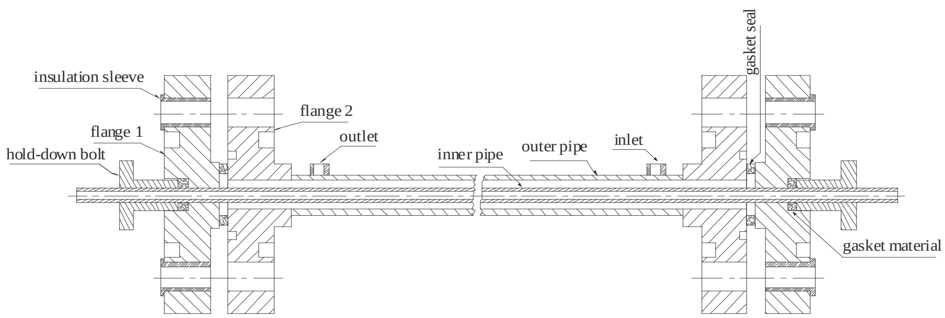

Figure 4 Schematic Diagram of the XJTU Annulus Test Section Geometry.

The test section was installed vertically in the loop and tested with an upward flow of water. Inlet and outlet fluid-temperatures, outlet pressure, and pressure drop over the test section were measured. Several fixed thermocouples were installed to measure wall temperature along the heated length. Wall temperature measurements have been obtained over a range of mass fluxes and heat fluxes at an outlet pressure of 25 MPa. Figure 5 illustrates variation of wall-temperature with local enthalpy and heat flux. The wall temperature increases with increasing fluid enthalpy and increasing heat flux. The temperature increase becomes more gradual at the pseudo-critical enthalpy.

3. Supercritical Heat Transfer with Surrogate Fluids

Heat-transfer experiments with supercritical pressure water flow are difficult to design and perform and are expensive due to the high pressures and temperature and the high level of heating power required. Surrogate fluids (such as carbon dioxide and refrigerants) have been used instead. Such fluids have previously been utilized in studies of critical heat flux and film-boiling heat transfer at subcritical pressure conditions. Using surrogate fluids reduces costs, simplifies test-section design, reduces operation risk, and increases the testing flexibility. These benefits stem from the fact that supercritical conditions for surrogate fluids are less severe than those for water. Table 1 summarizes the critical properties of water, carbon dioxide and refrigerant R-134a.

Heat-transfer experiments using surrogate fluids have considerably expanded the database and produced data at water-equivalent conditions that could lead to test-section damage in corresponding water tests. However, experimental data obtained with surrogate fluids are not directly applicable to water flow. Transformation of thermalhydraulics parameters for surrogate fluid experiments is required to establish equivalent values for water applications. Fluid-to-fluid modeling parameters have been established for critical heat flux and film-boiling heat transfer at subcritical conditions and for heat transfer at supercritical conditions.

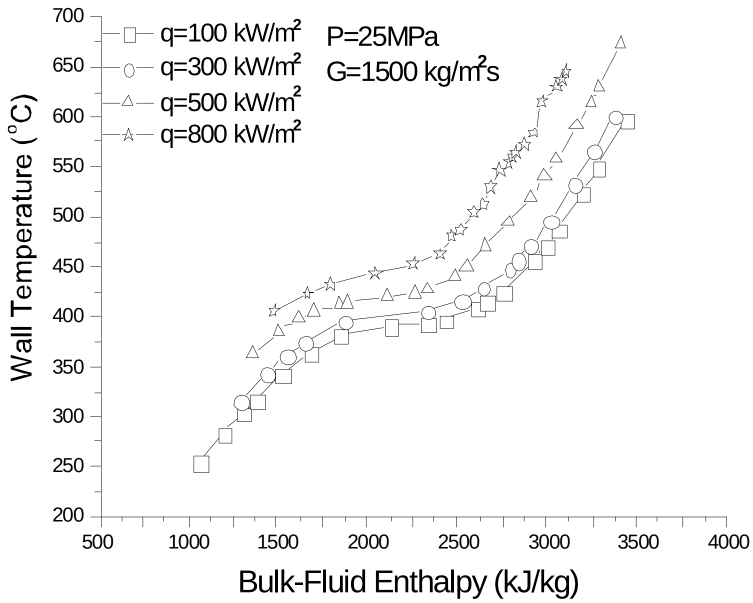

Figure 5 Wall Temperature Measurements Obtained with Vertical Upflow of Supercritical Water inside an Annulus.

Table 1 Critical Parameters for Water, Carbon Dioxide and Refrigerant R-134a

Parameter	Unit	Water	CO_2	R-134a
Critical pressure, P_c	MPa	22.1	7.38	4.06
Critical temperature, T_c	K (ºC)	647.3 (374.1)	304.2 (31.0)	374.2 (101.0)
Critical density, ρ_c	$kg{\cdot}m^{-3}$	315	468	512

Participants of the CRP have contributed supercritical heat-transfer data for carbon dioxide, Refrigerant-12, Refrigerant-22, Helium, n-Heptane, Nitrogen, and Toluene. Only the database for carbon dioxide is summarized in this paper. AECL supercritical carbon dioxide heat-transfer database covers the following range of parameters: tube diameters from 0.5 to 22.8 mm, pressures from 7.3 to 12 MPa, mass fluxes from 78 to 7520 $kg{\cdot}m^{-2}{\cdot}s^{-1}$, and heat fluxes from 0.8 to 1540 $kW{\cdot}m^{-2}$. The range of fluid temperature (or enthalpy) is difficult to specify due to differences in referencing. Figure 6 illustrates the range of reduced pressure and reduced temperature covered in the AECL carbon-dioxide datasets [2]. Nearly all data were collected at supercritical pressures.

SJTU collected 4881 data points for supercritical heat transfer in tubes with upward flow of carbon dioxide [3]. The majority of SJTU carbon dioxide data are based on the same sources as those in the AECL database.

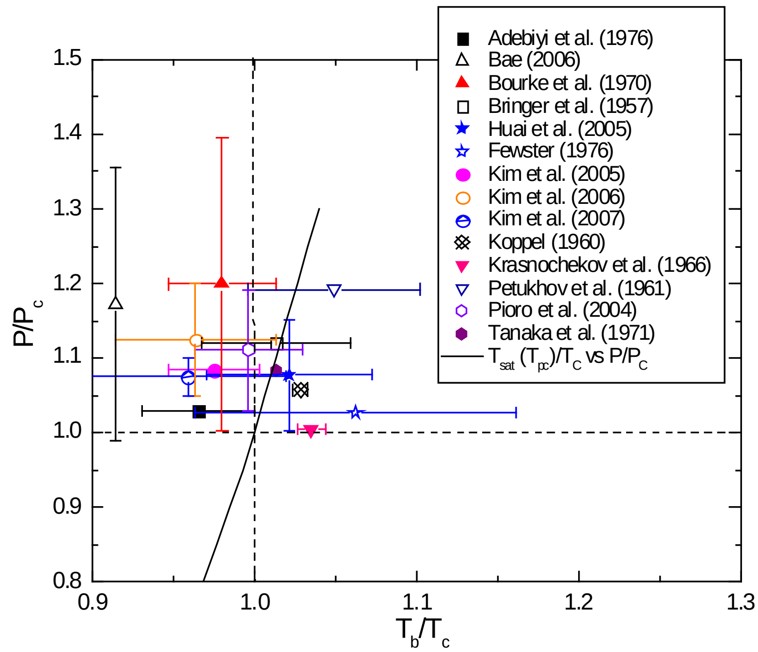

Figure 6 Range of Selected AECL Supercritical Heat-Transfer Database for Carbon Dioxide Flow (see [2] for reference details).

Three sets of supercritical heat-transfer experiments were performed with carbon dioxide flow in vertical tubes at Manchester University [7], [8], [9]. These experiments covered a wide range of flow conditions in both upward and downward flow (see Table 2).

Table 2 Summary of Experimental Conditions with Supercritical Pressure Carbon Dioxide at 7.504 MPa at Manchester University

Flow Direction	Tube Diameter (mm)	Fluid Inlet Temperature (°C)	Mass Flux ($kg\cdot m^{-2}\cdot s^{-1}$)	Heat Flux ($kW\cdot m^{-2}$)
Downflow	18.97	20, 23	0.068 - 0.16	50.2, 52.3
Upflow	18.97	7 – 31.5	0.041, 0.058, 0.16	2.56 - 59.2
Upflow / downflow	18.97	9.8 -27.63	0.068, 0.082, 0.16	9.21 – 56.7
Upflow	18.97	0, 4.5, 8, 10, 15, 20	0.029, 0.041, 0.058, 0.082	1.7 – 26.8
Upflow / downflow	18.97	10, 15, 20	0.029, 0.041, 0.058, 0.082	9.21 – 56.7
Upflow	0.51, 0.82	8 - 28	0.0058, 0.0129, 0.02	17.7 – 64
Upflow / downflow	0.51, 0.82, 18.97	5.5 - 30	0.0058 – 0.16	0.73 – 455

4. Separate Effects on Supercritical Heat Transfer

SCWR design and safety analyses must take into account separate effects on supercritical heat transfer. Such effects represent the heat-transfer changes as a consequence of departures from the reference geometry, which has been used in experiments to obtain the experimental data for correlation development. As illustrated in several literature surveys (e.g., Pioro and Duffey [1]), most experimental data were obtained using with tubular test sections. Hence the correlations are only strictly applicable for tubes (and need to be extended for use in subchannel applications). Experimental data for rod bundles are scarce and hence only a limited number of correlations have been proposed. These are strictly only applicable for the specific geometry, power profiles, and flow conditions, which were used in the experiments. From the point of view of design, the following separate effects are of main interest:

- Diameter (change in flow area);
- Geometry (tubes, annuli, and bundles);
- Axial power profile (uniform and non-uniform with various shapes);
- Radial power profile in bundles (uniform and non-uniform with various shapes);
- Circumferential power profile in fuel rods (uniform and non-uniform with various shapes);
- Spacing devices (wrap-around wires, spacer grids, and spacers);
- Gap size (small and large);
- Flow direction (upward and downward);
- Transient (postulated accident scenarios).

Experimental data covering these separate effects are very limited. Only some selected effects are discussed in following sections since no experimental data are available for the others.

4.1 Effect of Diameter

Subchannel sizes in a SCWR fuel bundle are optimized to minimize the flow and enthalpy mal-distributions and enhance heat transfer leading to cladding temperature reduction. Change in subchannel size affects the subchannel flow area, thus impacting on the heat-transfer characteristics. Most heat-transfer correlations implicitly take account of the flow area change through the mass flux and diameter (which together determine Reynolds number). The applicability of these terms in accounting for the flow-area change has been examined with heat-transfer data obtained in tubes of various diameters.

Kuang et al. [10] compared wall-temperature measurements obtained with two tubes of 12-mm and 26-mm diameter. Figure 7 shows the effect of tube diameter on wall temperature and heat transfer. For the same flow and heat flux conditions, there appears to be no effect of tube diameter on wall temperature at bulk-fluid enthalpies lower than the pseudo-critical enthalpy. The wall temperature becomes lower for the small-diameter tube with increasing bulk-fluid enthalpy beyond the pseudo-critical point. Heat-transfer coefficients are higher for the small-diameter tube at high bulk-fluid enthalpies but the difference diminishes with decreasing enthalpy.

4.2 Effect of Spacing Devices

Spacing devices are essential to maintain the subchannel flow area and gap size and to reduce fuel pin vibration in bundles. Various types of spacing devices have been used in nuclear reactor cores; grid spacers in light-water reactors, wire-wrapped spacers in gas-cooled reactors, and appendage-type spacers in pressurized heavy-water reactors. The spacer type for SCWR fuel bundles has not yet been finalized, but wire-wrapped spacers are proposed because of the similarity between gas-cooled reactors and SCWRs.

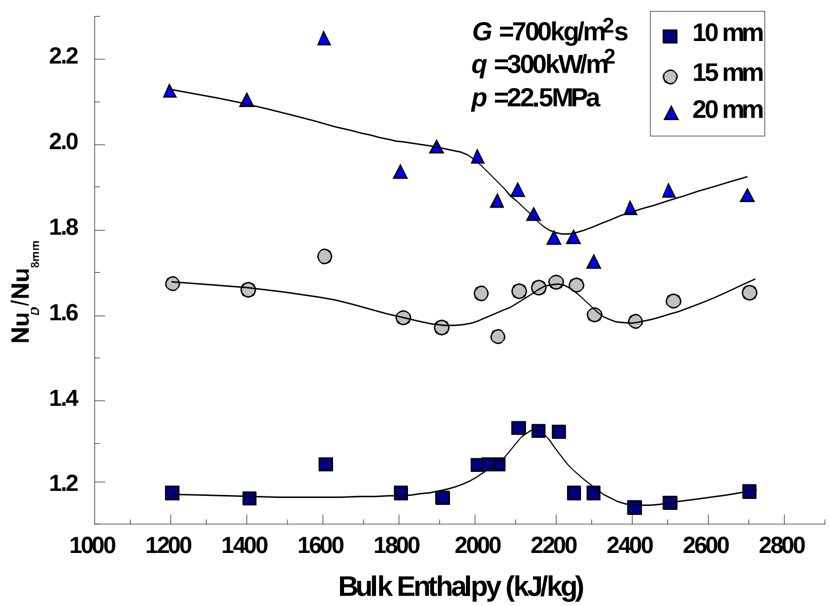

Figure 7 Effect of Diameter on Nusselt Number in Tubes.

A ceramic spacer was installed to the annulus test section at Xi'an Jiaotong University [6]. It was not attached to the heated surface minimizing the stagnation flow effect that could potentially lead to high surface temperature. Figure 8 shows the observed surface temperature distribution along the test section. The temperature increases with axial distance, but drops rapidly at the spacer location. This illustrates the heat-transfer enhancement effect of the spacer. There is no measurement upstream of the first spacer.

4.3 Effect of Flow Direction

The flow direction of the supercritical pressure coolant in current SCWR design concepts can be upward, downward, or both upward and downward (in a multi-path core). While most experimental data have been obtained with upward flow in tubes, quite a number of experiments have been performed to examine the heat-transfer characteristics with downward flow.

Figure 9 compares wall-temperature measurements for upward and downward flows at a pressure of 5.5 MPa, a mass flux of 400 kg·m^{-2}·s^{-1}, and a heat flux of 30 kW·m^{-2}. Wall-temperature measurements are higher for upward flow than for downward flow in the deteriorated heat-transfer region. Beyond this region, the wall-temperature measurements are similar for both upward and downward flows. One noticeable difference is the absence of a deteriorated heat-transfer conditions with downward flow in the low enthalpy region.

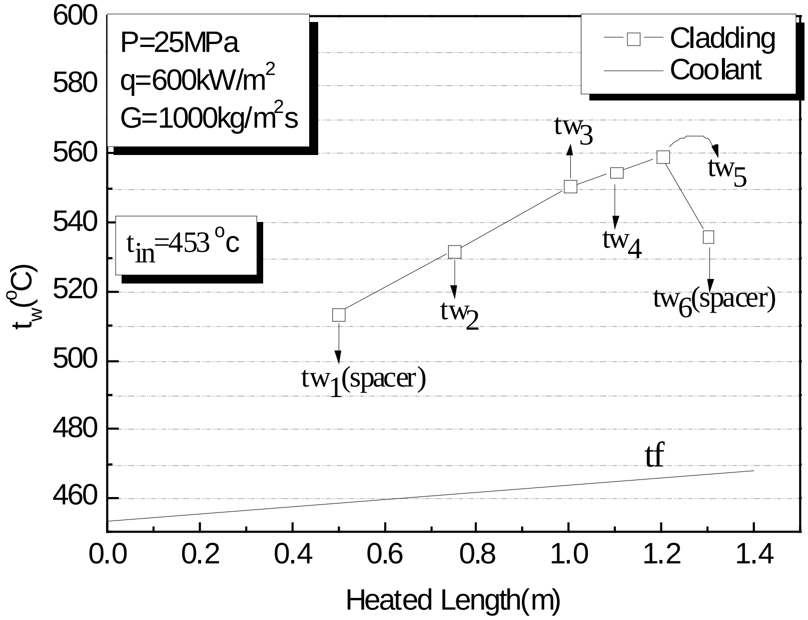

Figure 8 Wall Temperature Distribution in the Spacer-Equipped Annulus Test Section.

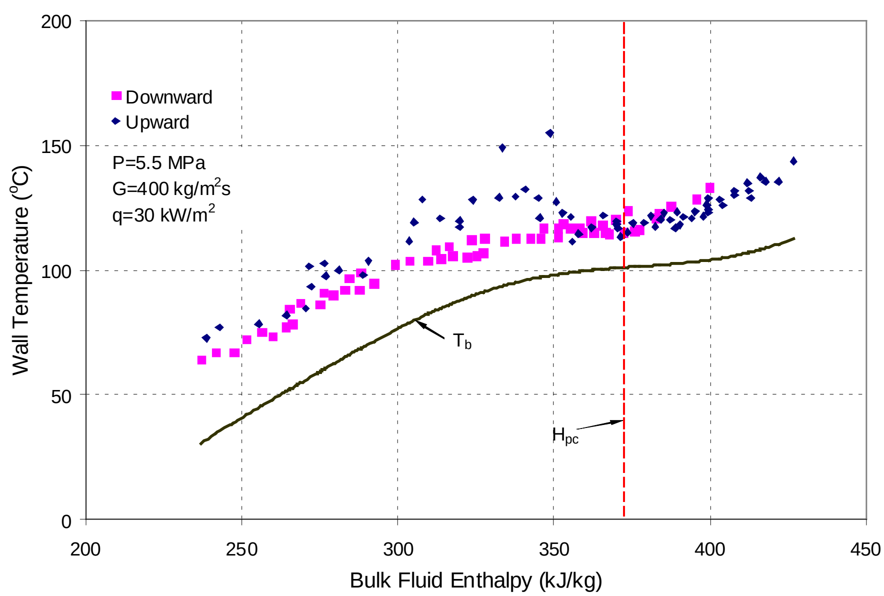

Figure 9 Effect of Flow Direction on Supercritical Heat Transfer in a 3-Rod Bundle.

5. Conclusion

Large amounts of experimental data are available for vertical upward flow of water and surrogate fluids in tubes. These data covered a wide range of flow conditions. However, the amount of data remains limited at conditions of interest in connection with the current SCWR design concepts. Nevertheless, these data are applicable for developing prediction methods for supercritical heat-transfer coefficients.

Separate effects on supercritical heat transfer have been identified for SCWR fuel and core analyses. Surveys of experimental studies revealed that experimental data are available for several different separate effects. Most of these data were obtained with tubes and annuli, and the observed effect may not be applicable in bundle analyses. Therefore, additional experimental data are required to understand such phenomena better and for the development of calculation methods for describing observed behaviour.

6. Acknowledgments

The authors would like to thank the following IAEA CRP participants for their contribution to this paper:

- Prof. X. Cheng of Shanghai Jiaotong University
- Prof. J.D. Jackson of University of Manchester
- Prof. P.L. Kirillov of Inst. of Physics and Power Eng.

The authors also would like to thank OECD/NEA for hosting the database, which has been a very useful tool for the CRP participants to share their data and to accelerate their research.

7. References

[1] I.L. Pioro and R.B. Duffey, "Experimental Heat Transfer to Supercritical Water Flowing inside Channels (Survey)," Nucl. Eng. Des., **235,** 22, 2407-2420 (2005).

[2] D.C. Groeneveld, S. Tavoularis, P.R.S.S. Gudla, S.K. Yang and L.K.H. Leung, "Analytical and Experimental Program of Supercritical Heat Transfer Research at the University of Ottawa," Nucl. Eng. Techn., **40**, 2, 107-116 (2008).

[3] B. Kuang, Y.Q., Zhang and X. Cheng, "A New, Wide-Ranged Heat Transfer Correlation of Water at Supercritical Pressures in Vertical Upward Ducts," Proc. of the 7th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety (NUTHOS-7), Seoul, Korea, October 5-9 (2008).

[4] P.L. Kirillov, R.S. Pomet'ko, A.M. Smirnov and V.A. Grabezhnaia, "Investigation of Heat Transfer to Water at Supercritical Pressures in Tubes and Rod Bundles," IPPE report FEI-3051, Obninsk (2005).

[5] J.D. Jackson, “Supercritical Pressure Water Heat Transfer Study at Manchester”, Report produced for AECL in connection with the database on SCWR Heat Transfer being established by IAEA, Ref. JDJ/IAEA/CRP/ Report No. 2, October 2009.

[6] W. Gang, Q. Bi, X. Zhu, Z. Yang and H. Wang, “Experimental Investigation on Heat Transfer of Supercritical Pressure Water in Annular Flow Geometry,” Proc. of the 5th Int. Sym. SCWR, Vancouver, Canada, March 13-16 (2011).

[7] J. Fewster, “Mixed Forced and Free convective Heat Transfer to Supercritical Pressure Fluids Flowing in Vertical Pipes”, Ph.D. Thesis, University of Manchester (1976).

[8] K.O.J. Evans Lutterodt, “Forced Convection Heat Transfer to Carbon Dioxide at Near-critical Pressure Conditions,” Ph.D. Thesis, University of Manchester, (1968).

[9] R.S. Weinberg, “Experimental and Theoretical Study of Buoyancy Effects in Forced Convection Heat Transfer to Carbon Dioxide at Supercritical Pressure,” Ph.D. Thesis, University of Manchester (1972).

[10] Kuang, B., Zhang, Y.Q. and Cheng, X., “A New, Wide-Ranged Heat Transfer Correlation of Water at Supercritical Pressures in Vertical Upward Ducts”, Proc. 7th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety (NUTHOS-7), Seoul, Korea, October 5-9, Paper 189 , 2008.