STATUS AND CHALLENGES OF NUCLEAR THERMAL-HYDRAULICS RESEARCH IN CHINA

X. Cheng [1-3], Y.H. Yang [2-3]
1) Karlsruhe Institute of Technology (KIT), Germany
Email: xu.cheng@kit.edu

2) Shanghai Jiao Tong University (SJTU), China

3) State Nuclear Power Research Institute (SNPRI), China

ABSTRACT

Nuclear energy has been considered as an important part in the medium- and long term Chinese energy mixing. Accompanying with the nuclear power development program extensive R&D program was initiated. Nuclear thermal-hydraulics has been recognized as a key subject in the nuclear technology development, because the target of nuclear thermal-hydraulics is to remove heat safely from the fuel. The thermal-hydraulic activities in China cover mainly three areas, i.e. reactor core thermal-hydraulics, safety related thermal-hydraulics and fundamental thermal-hydraulics.

This paper gives a brief summary of the ongoing R&D activities in these three areas. Focuses are put on a few selected topics, i.e. (a) transversal mixing and non-uniform heat transfer distribution in fuel assemblies, (b) two-phase distribution at ERVC conditions, (c) heat transfer of supercritical fluids. The state-of-the-art of the ongoing works and the challenging aspects are presented and discussed.

KEYWORDS: China, R&D, Thermal-hydraulics, Overview

1. INTRODUCTION

In the last 10 years, the primary energy production becomes double and a more significant increase in the installed electricity generation is observed. In the present energy mix in China about 75% of electricity is generated via coal fired power plants [1]. This results in serious problems, such as climate change and environmental pollution. Limitations in renewable energy development are obvious. Therefore, nuclear energy has been considered as an important portion in the medium- and long term Chinese energy mixing.

One of the lessons leant from the Fukushíma accident is that the design and construction of future Chinese nuclear power plants will obey the highest safety standards available worldwide. Only nuclear power plants of GEN-III or beyond will be constructed in China. Furthermore, it was agreed that R&D activities should be strongly enhanced, to improve nuclear safety and other related technologies.

In China significant efforts are made in one hand to realize the self reliance of nuclear technology based on GEN-III large scale PWR, and in the other hand to develop next generation nuclear reactor systems with improved features of safety, sustainability and economical competitiveness. R&D activities on five of the six GEN-IV reactors as well as on transmutation systems are ongoing in China. Related to both GEN-III technology self reliance and future nuclear systems, main focuses of R&D activities are put on advanced reactor and safety technologies.

Thermal-hydraulics is recognized as a key scientific subject in the development of advanced reactor cores and safety technologies, because their overall target is the safe removal of heat at normal conditions and accident conditions. Therefore, in the last years R&D activities on nuclear thermal-hydraulics have been growing rapidly and cover a wide range of topics. They can be roughly divided into three areas, i.e. reactor core thermal-hydraulics, nuclear safety related thermal-hydraulics and fundamental thermal-hydraulics.

It is impossible to provide a complete list of the research topics in this paper. In March 2015, the 1st Sino-German Workshop on Fundamentals of Advanced Nuclear Safety Technologies (SG-FANS) took place in Shanghai [2]. At this workshop, nearly all relevant institutions from both countries presented their R&D activities on nuclear safety with focus on thermal-hydraulics. This paper is based on the information presented at the SG-FANS workshop, and summarizes briefly the status and challenges of research activities in China with focuses on three selected subjects, i.e. (1) reactor core thermal-hydraulics; (2) In-vessel retention (IVR) related thermal-hydraulics; (3) supercritical fluid (SCF) thermal-hydraulics.

2. RESEARCH METHODOLOGY

The main research activities in China are financed by the Chinese government. Therefore, they are well coordinated. The nuclear thermal-hydraulics community consists of universities and research institutions. At the SG-FANS workshop 7 universities and 6 research institutions from China presented their research activities on nuclear safety related thermal-hydraulics. They are:

Universities

Harbin Engineering University (HEU)

North China Electric Power University (NCEPU)

Shanghai Jiao Tong University (SJTU)

Sun Yat-sen University (SYSU)

Thinghua University (THU)

University of Science and Technology of China (USTC)

Xi'an Jiao Tong University (XJTU)

Research Institutions

China Institute of Atomic Energy (CIAE)

China Nuclear Power Research Institute (CNPRI)

Institute of Nuclear Energy Safety Technology (INEST)

Nuclear Power Institute of China (NPIC)

Shanghai Nuclear Engineering Research and Design Institute (SNERDI)

State Nuclear Power Research Institute (SNPRI)

These 13 institutions belong to the key members of the Chinese nuclear thermal-hydraulics research community, which, however, is not limited to the above mentioned institutions. The methodology used in Chinese thermal-hydraulics R&D is indicated in Figure 1. At first important processes and phenomena are identified and tasks are defined. This is achieved through discussions among experts from industries, research institutions and universities. The final objective of the R&D activities is to develop models or numerical tools to describe the phenomena and processes. To achieve this target, experimental studies are carried out, to provide experimental evidence for a better understanding of the phenomena and more important to establish experimental data base, which is inevitable for model development and code validation. Emphasis is also put on international collaboration, to extend the experimental data base, wherever necessary and possible. At the same time, numerical analysis is performed to support experimental work, including the definition of test matrix, the selection of measurement techniques and test data analysis. Based on experimental data base, new models and numerical tools will be developed and validated. All three classes of thermal-hydraulics codes are considered, i.e. system thermal-hydraulics (STH) codes, sub-channel codes and CFD codes.

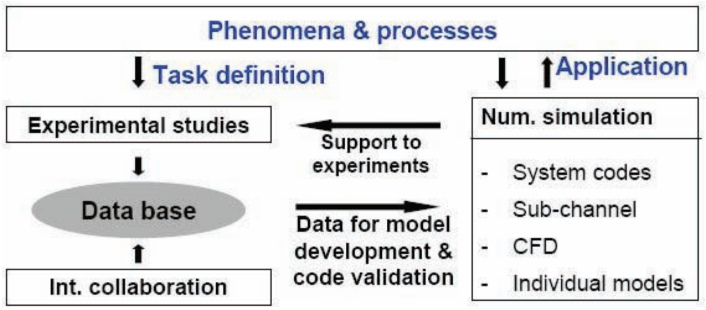


Figure 1: Methodology used in Chinese thermal-hydraulics R&D

3. SELECTED RESEARCH ACTIVITIES

3.1 Reactor core thermal-hydraulics

Table I summarizes some topics and institutions involved in reactor core thermal-hydraulics.

Table I: Selected R&D activities on reactor core thermal-hydraulics

Selected topics	*Experimental studies*	*Numeric studies*
Sub-channel codes	CNPRI, NPIC, SJTU, SNPRI, XJU	
Transversal mixing	NPIC, SJTU	SJTU
CHF, Post-CH	CIAE, CNPRI, NPIC, SJTU, XJTU	SJTU, XJTU
Non-uniform heat transfer distribution	SJTU	SJTU

The most important target in the reactor core thermal-hydraulics is to develop and to improve the accuracy of sub-channel codes for various reactors. Transversal mixing is the key process affecting the accuracy of sub-channel analysis codes. There are various driving mechanisms for transversal mixing and they can be well divided into five groups, i.e. turbulent mixing due to turbulent velocity fluctuation, diversion cross flow due to pressure difference, void drift due to bubble diffusion, flow scattering and flow sweeping due to spacers. Local non-uniform heat transfer distribution on fuel rods is another crucial phenomenon in reactors with tight or semi-tight fuel assemblies cooled with single phase fluids, such as liquid metal or supercritical water [3]. For water cooled reactors, CHF remains a key topic. R&D activities both fundamental feature and engineering application oriented are ongoing. In relation to incident conditions, post-CHF prediction becomes an important topic and is still far not reliable. In the following, two topics, i.e. turbulent mixing and non-uniform heat transfer distribution are presented.

(A) Turbulent mixing

Velocity fluctuation at the gap between two neighboring sub-channels is an important parameter affecting turbulent mixing of mass, momentum and energy. Therefore, velocity fluctuation at the gap region has been intensively investigated both experimentally and numerically [4], [5]. At the MIRFF test facility of SJTU, as photographically shown in Figure 2, three-dimensional flow field and turbulence quantities were measured in rod bundles. LDA measurement was carried out at different cross sections downstream the spacer grid. In addition, numerical simulation with CFD codes was carried out. Figure 3 compares the measured velocity fluctuation with the numerical results. It is seen that qualitatively there is good agreement between the experimental data and the numerical results. These test data can be well used for code validation and the results on velocity fluctuation in the gap can be applied for the development of turbulent mixing correlations.

Fig. 2: MIRFF test facility at SJTU [4]

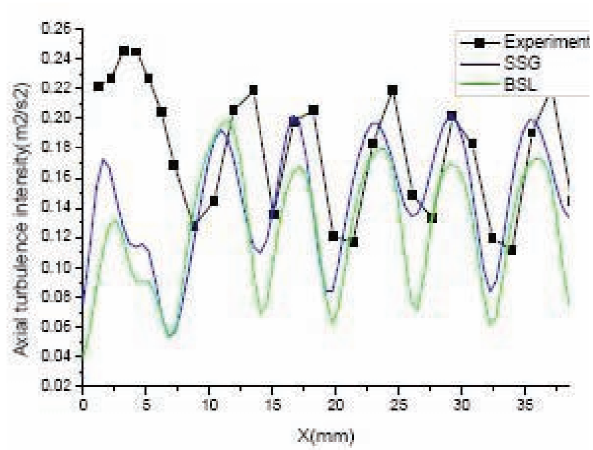


Fig. 3: Comparison of turbulence intensity [4]

In addition to experimental studies, systematic numerical analysis was performed to investigate the turbulent velocity fluctuation [5]. Under the assumption that the velocity fluctuation obeys the Gaussian profile, the velocity fluctuation can be derived from the Reynolds stress, which is obtained from CFD simulations

$$\varepsilon = \overline{|v'|} = \frac{\sqrt{\overline{v' \cdot v'}}}{\sqrt{\pi}} \tag{1}$$

Here $\boldsymbol{v'}$ stands for transversal velocity fluctuation. Figure 4 shows the average velocity fluctuation divided by the stress velocity. It is found that the relative velocity fluctuation is hardly dependent on Reynolds number. Based on systematic CFD analysis, empirical equation was then derived for the so called turbulent mixing, defined as $\boldsymbol{\beta=\varepsilon/u}$, with $\boldsymbol{u}$ the average axial flow velocity.

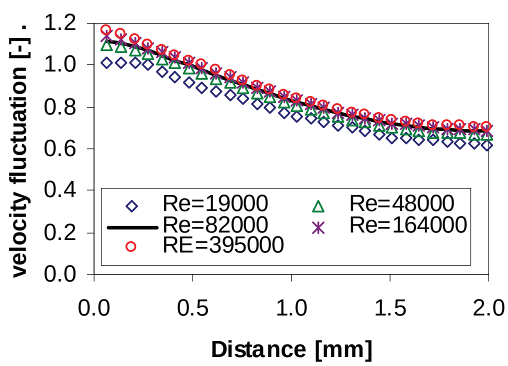


Figure 4: Dependence of velocity fluctuation on Reynolds number [5]

Velocity fluctuation can be determined either experimentally or numerically. However, the main challenging task is to establish the relationship between the velocity fluctuation and the transversal turbulent mixing of mass, momentum and energy. Nowadays, no reliable models are available to describe these three turbulent mixing processes.

(B) Non-uniform heat transfer distribution

It is well agreed that the temperature distribution is not uniform around the cladding surface, especially in a tight lattice, which is widely applied in the fuel assembly design of innovative nuclear reactors, in order to enhance the fuel utilization and economics. Since the maximal cladding temperature is one of the key design criteria, the circumferential heat transfer difference should be taken into consideration in the fuel assembly design. However, the circumferential heat transfer variation is not taken into consideration by the current thermal-hydraulic codes, such as sub-channel codes.

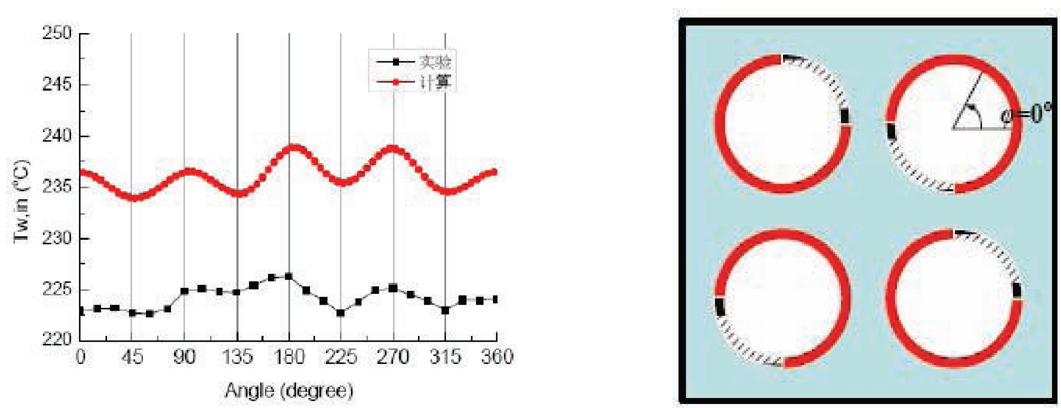
To consider this non-uniform heat transfer distribution in the sub-channel code, efforts have been made at SJTU to look into the mechanisms of the local non-uniformity of heat transfer, to derive correlations for the local heat transfer and to improve sub-channel codes [6].

At first, based on the similarity principle and the governing equations for the computational domain, seven dimensionless parameters affecting heat transfer are identified. The effect of these seven parameters on heat transfer is then detailed analyzed using CFD approach and the importance of each dimensionless parameter is clarified. It shows that heat transfer distribution depends mainly on the following four parameters, i.e. circumferential angle φ, pitch-to-diameter ratio $\frac{p}{d}$, Prandtl number Pr and ratio of the cladding thermal conductivity to the coolant thermal conductivity $\frac{\lambda_{clad}}{\lambda_{coolant}}$. Analytical analysis indicates that the heat transfer profile is a cosine series. By keeping the first order cosine term, the following structure of correlation is proposed for heat transfer coefficient (HTC) in rectangular arrangement of rod bundles:

$$htc_N = 1 - \frac{\cos(4\varphi)}{1 + c_0 \cdot \left(\frac{p}{d} - 1\right)^{n1} \cdot Pr^{n2} \cdot \left(\frac{\lambda_{clad}}{\lambda_{coolant}}\right)^{n3}} \tag{2}$$

Here $\boldsymbol{htc_N}$ stands for the local heat transfer coefficient (HTC) normalized by the average heat transfer coefficient. The coefficient $\boldsymbol{c_0}$, $\boldsymbol{n_1}$, $\boldsymbol{n_2}$ and $\boldsymbol{n_3}$ are determined for various types of sub-channels such as central, wall and corner sub-channels. Similar correlations are also derived for triangular lattices.

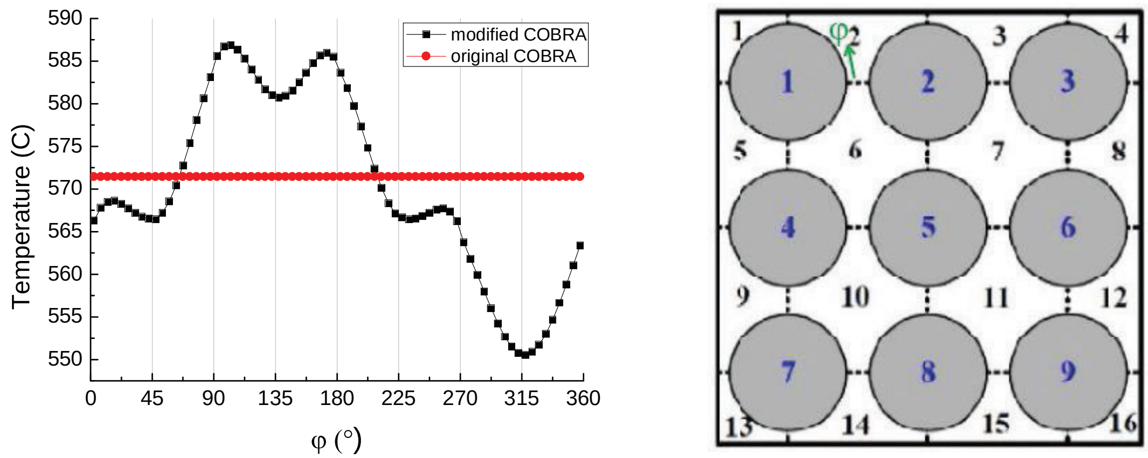
Figure 5 compares the results predicted with equation (2) with measured experimental data obtained in a 2x2 rod bundle.



(a) Comparison of temperature (b) cross section of 2x2 rod bundle

Figure 5: comparison of prediction with experimental data (parameters: d=8 mm, p/d=1.18, P=7.78 MPa, G=1.25 Mg/m^{2}s, q=0.46 MW/m^2, T_B=164°C)

To consider the non-uniform heat transfer distribution in sub-channel codes, a separate program module of 3-dimensional heat conduction has been developed with equation (2) for local heat transfer distribution. This module is then implemented into the sub-channel code COBRA. Figure 6 compares the cladding temperature of the rod No.1 in a 3x3 rod bundle cooled with supercritical water. It is seen that the maximum temperature occurs in the gap between the rod and the wall, whereas the lowest temperature appears in location directing to the center of the central sub-channel. The difference between the maximum and the minimum temperature is as high as 40°C.



(a) Temperature on rod 1 (b) Cross section of 3x3 rod bundle

Figure 6: Comparison of sub-channel results with and without local heat transfer models

3.2 In-vessel retention related activities

Severe accident (SA) mitigation is one of the key features of GEN-III LWRs, compared to those of GEN-II. Fukushima accident emphasizes the importance of SA mitigation system. The objectives of the SA mitigation system are to minimize the consequence of severe accidents, to guarantee the integrity of containment and to avoid large release of radioactive materials. In the GEN-III LWRs, SA mitigation concepts can be divided into two classes. In one class, the mitigation system focuses on the retention of the core melt inside the RPV, the so called in-vessel retention (IVR) concept. The other concept allows the failure of RPV and has a core catcher outside the RPV, the so called ex-vessel core catcher concept. Both concepts exist in the Chinese GEN-III PWRs. The AP1000 and the further developed CAP-series applies IVR concept, whereas VVER and EPR in China use ex-vessel core catcher concept.

Figure 7 shows schematically the principle of IVR and the related main phenomena. In case of severe accident, reactor core is fully or partially melt down, relocates in the lower plenum and forms a melt pool. The melt mass and composition is dependent on accident scenarios. From thermal-hydraulic point of view, the melt pool is a multi-component, multi-phase problem with phase transition. Decay heat released in the melt pool is transferred through the conduction of the RPV wall to coolant flowing outside the RPV, the so called external reactor vessel cooling (ERVC). Depending on heat flux and other flow conditions, boiling crisis may occur at the RPV outer surface, which leads to a strong increase in surface wall temperature.

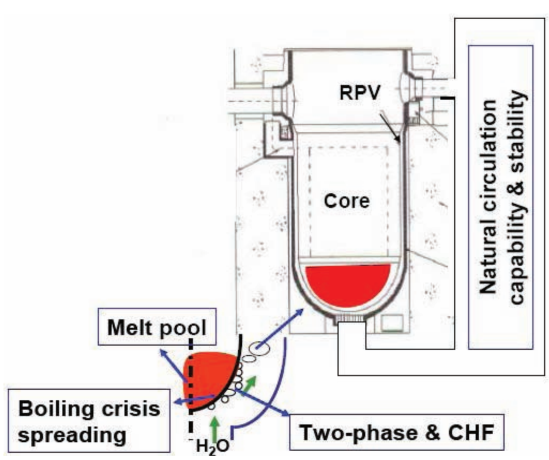


Figure 7: Schema showing the main phenomena involved in the IVR concept

It is well known that the feasibility of the IVR concept is one bottlenecking issue for large scale PWRs. Thus, extensive research activities were initiated in the last years to investigate the physical phenomena involved and to provide engineering confirmation of the IVR-ERVC concept. Table II summarizes some activities ongoing in China. In the following, some results related to fundamental ERVC two-phase flow are presented.

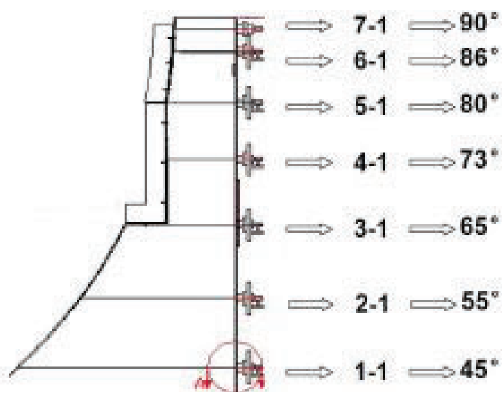
Table II: Selected R&D activities on IVR-related thermal-hydraulics

Selected topics	***Experiments***	***Numeric studies***
Melt pool behavior	NPIC, SNPRI, XJTU	SJTU, XJTU
ERVC two-phase flow & CHF	CNPRI, NPIC, SJTU, SNPRI, XJTU	SJTU, SNPRI, XJTU
Thermal stratification of water pool	NCEPU, SJTU	NCEPU, SJTU
ERVC System dynamics	CNPRI, SJTU	CNPRI, SJTU, SNPRI

(A) Two-phase distribution

Experimental studies on two-phase distribution in ERVC relevant flow channel with water-air flow were carried out at the REPEC-I facility of SJTU [7], schematically shown in Figure 8. The test section is 1/4 arc-shaped slice structure simulating the flow channel outside of the RPV lower head. The radius and the width of the test section are 2.1 m and 0.15 m, respectively. An air injection device made of porous materials is installed in the bottom of the test section. Void fraction was measured at seven cross sections of different elevations, as illustrated in Figure 8b.

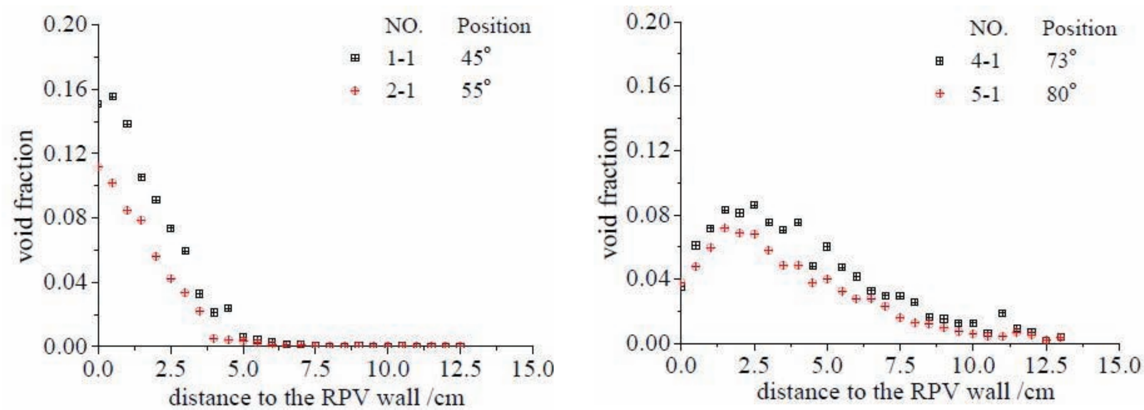
(a) Photo of the REPEC-I facility



(b) Positions of void measurement

Figure 8: REPEC-I Test facility at SJTU

Fig.9 shows the measured local void fraction distributions along the cross-sections 1-1, 2-1, 4-1 and 5-1. As seen in Fig.8b, the normal direction of the wall surface at cross sections1-1 and 2-1 are 45°, 35°, whereas the normal direction of the other cross section 4-1 and 5-1 is horizontal. The results show that the local void fraction at the first two cross sections with inclined wall surface exhibits a wall peak distribution. The local void fraction decreased rapidly as the distance to the wall increases. At cross sections 4-1 and 5-1 with vertical wall surface, the void peak occurs at a distance from the wall.



(a) With inclined wall surface

(b) With vertical wall surface

Figure 9: Void fraction distribution at various cross sections

In addition to experimental studies, efforts were also made to evaluate and to develop CFD approaches, to simulate the void distribution at ERVC conditions. Three approaches were assessed [8]. In the first approach, constant bubble diameter was taken. It was found that the results are very sensitive to the bubble diameter assumed. It is well known that the local void fraction in the ex-vessel flow channel varies over a large range, from small void fraction in bubbly flow to large local void fraction near the wall close to boiling crisis. The local bubble diameter changes also significantly. The second approach bases on interfacial area transport equation, which contains various source terms presenting bubble breakup, bubble coalescence, bubble condensation and nucleation. The challenging issue is to select reliable models for various source terms. Numerical results showed that the CFD simulation results are very sensitive to the selection of the models for the source terms.

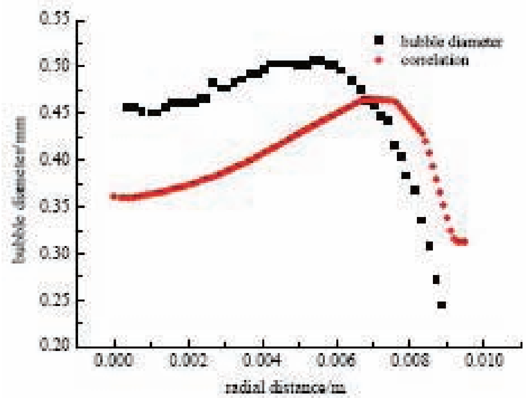
To overcome this difficulty and to enhance the robustness of the CFD simulation, the third approach uses empirical correlation to describe the bubble diameter. The basic idea is to introduce the following structure of equation for bubble diameter:

$$D_b \propto D_w F_{COND} F_{BK} F_{CO} \tag{3}$$

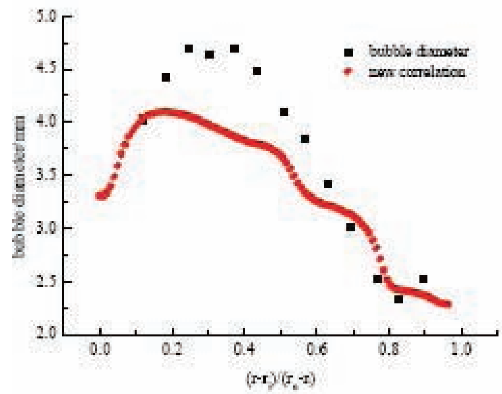
Here, $\boldsymbol{D_w}$ represents bubble diameter at its departure from the wall and is taken as the reference value. The three coefficients $\boldsymbol{F_{cond}}$, $\boldsymbol{F_{BK}}$ and $\boldsymbol{F_{CO}}$ account the effect of condensation, breakup and coalescence, respectively. Based on semi-analytical analysis and comparison with experimental data from open literature, the following equation is proposed:

$$D_b = 1.5 \times D_W \times e^{(-0.05 \times Ja)} \times (1 + \alpha^{0.5}) \times e^{(-0.1 \times We)} \tag{4}$$

Hear ***Ja*** stands for Jacob number, ***We*** for Weber number and $\boldsymbol{\alpha}$ for void fraction. Figure 11 compares the CFD simulation with equation (4) for the bubble diameter with the test data of DEBORA and of Lee. The results confirm the feasibility of the third approach.



Test data of DEBORA



Test data of Lee

Figure 10: Comparison of CFD results using equation (4) with test data [8]

3.3 Supercritical fluid thermal-hydraulics

The main thermal-hydraulic phenomena involved in supercritical water cooled reactors (SCWR) consist of heat transfer, flow stability, natural circulation and critical flow. Table III summarizes the related activities in China. Because of supercritical pressure conditions, maximum cladding temperature becomes one of the main design criteria instead of critical heat flux. It is well known that temperature increase of a few ten degrees could lead to significant damage of cladding due to corrosion. Therefore, accurate determination of heat transfer and cladding temperature is of crucial importance for the SCWR design. However, prediction of heat transfer at SCWR relevant conditions is still far not reliable. Thus, heat transfer is one of the most important thermal-hydraulic processes considered in the SCWR R&D programs and is dealt below.

Table III: Selected R&D activities on SCWR thermal-hydraulics

Selected topics	***Experiments***	***Numeric studies***
Heat transfer	CIAE, NPIC, SJTU	NPIC, SJTU, THU, XJTU
System dynamics and Flow stability	CIAE, NPIC, THU	SJTU, THU, XJTU
Critical flow	CIAE	CIAE

(A) Heat transfer

It was agreed in the open literature that the heat transfer coefficient at supercritical (SC) conditions may deviate strongly from the Bittus-Boelter equation, especially near the pseudo-critical line [9].

Figure 11 shows the ratio of the heat transfer coefficient to the reference value at zero heat flux. It increases from 1.0 with increasing bulk temperature and reaches its maximum at a bulk temperature still far from the pseudo-critical point T_{PC}. After then it decreases with the bulk temperature approaching the pseudo-critical point. It may show a minimum value at a bulk temperature around the pseudo-critical point. At a bulk temperature far beyond the pseudo-critical point, the ratio approaches to unity again.

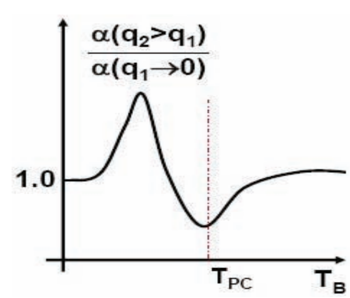


Figure 11: General behaviour of heat transfer in SC fluids

Experimental facilities

To investigate heat transfer of supercritical fluids experimentally, two test facilities have been constructed at SJTU, the SWAMUP facility with supercritical water and the SMOTH facility for supercritical Freon R134a. Figure 12 shows both test facilities. Detail information of both test facilities can be found in Zhao et al. [10] and Zhang et al. [11].

(a) SWAMUP facility

(b) SMOFT facility

Figure 12: Photos of both SCF heat transfer test facilities

Example of test results

Totally more than 15,000 data points were obtained. Figure 13 shows the ratio of the measured heat transfer coefficient (HTC) in circular tubes to the calculated HTC with the Dittus-Boelter correlation. In case the bulk temperature is far below the pseudo-critical point and the entrance effect is negligibly small, e.g. length-to-diameter ratio larger than 50, the HTC ratio is close to 1. That means HTC can be well predicted with the D-B correlation. The entrance effect leads to an increase in the HTC ratio. It is larger than 1 in the entrance region. The maximum HTC ratio in both test sets is about 1.2. The HTC ratio reduces strongly, as the bulk temperature approaches the pseudo critical value (384°C at P=25MPa). The HTC ratio reaches values lower than 0.2. It is clearly observed that the reduction in HTC ratio is continuous procedure and doesn't show any special behavior at the HTC ratio around 0.3. Therefore, definition of heat transfer deterioration (HTD) around the HTC ratio 0.3 is obviously random and doesn't represent any physical phenomenon.

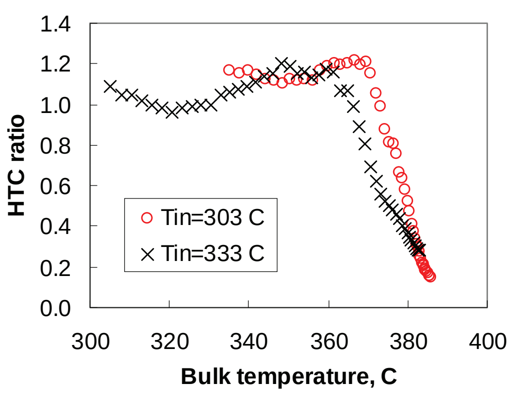


Figure 13: HTC ratio versus bulk temperature (D = 10 mm, P = 25 MPa, G = 1000 kg/m^{2}s, q = 0.70 MW/m^2) [10]

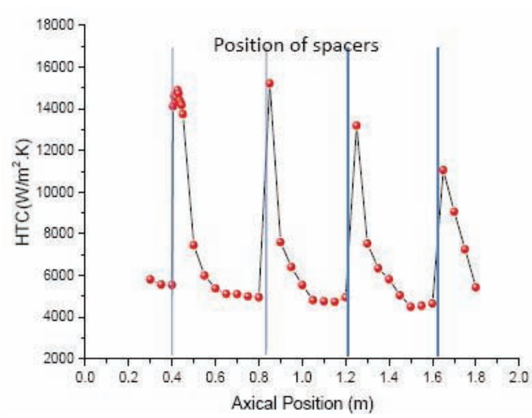


Figure 14: Measured heat transfer coefficient along axial position (P=25 MPa, G=800 kg/m^{2}s and q=0.6 MW/m^2) [11]

Figure 14 shows the measured heat transfer coefficient in annuli versus the distance from the test section entrance. It is seen that just after the spacer grids, heat transfer coefficient increases strongly. With the increase in the downstream distance from the spacers, heat transfer coefficient decreases at first sharply and then smoothly. Far away from the upstream spacer, e.g. 40 times of the hydraulic diameter, heat transfer coefficient approaches to a constant value. It can be concluded that at distance of more than 40 times hydraulic diameter, the effect of spacer on heat transfer coefficient is negligibly small.

Based on experiments performed at SJTU, an experimental data bank is established, as summarizes in Table IV. This data bank contains about 40,000 test data and is available for the development of heat transfer correlations and for the validation of numerical codes.

Table IV: Heat transfer test data in SC water

Channel geometry	D [mm]	P [MPa]	G [Mg/m^{2}s]	T_B [°C]	Q [MW/m^2]	Number of data points
Tubes	2.5 – 38	22.5 – 37.0	0.2 – 3.5	260 - 430	0.0 – 3.4	31062
Annuli	7.3	23.0 – 26.0	0.45 – 1.5	340 – 390	0.20 – 2.0	750
2x2 rod bundles	6.2, 9.2	23.0 – 26.0	0.43 – 1.8	330 – 390	0.40 – 1.5	5972

4 SUMMARY

This paper gives a brief summary of the thermal-hydraulics R&D activities in China. Examples of research results of a few selected topics are presented. The following conclusions are made:

- The thermal-hydraulics research activities in China are generally well coordinated, and the corresponding community is established.
- The extensive thermal-hydraulics activities are focused on three areas, i.e. reactor core thermal-hydraulics, nuclear safety related thermal-hydraulics and fundamental thermal-hydraulics. The main objectives of the Chinese thermal-hydraulics R&D are to develop physical models and numerical tools to describe the important phenomena occurring in the above mentioned three areas.
- Transversal mixing between sub-channels belongs to the key phenomena of the reactor core thermal-hydraulics. Several studies dealing with turbulent mixing were initiated and achieved very promising progress. However, the future challenge is the description of relationship between the local velocity fluctuation and the three turbulent mixing terms.
- The importance of the non-uniform heat transfer distribution was proved for the analysis and design of fuel assemblies with tight lattice, which are applied in several future reactors. A semi-empirical correlation was proposed based on strongly simplified assumptions. Future challenges are in the extension of this correlation to conditions with strong variation of fluid properties and larger parameter range of flow conditions.
- Critical heat flux determines the feasibility of the IVR concept. Due to the deficiency in reliable prediction methods, engineering confirmation is based on experimental measurement in scaled down test facilities. Efforts were made to understand and to reliably describe the main phenomena leading to boiling crisis. Both experimental and theoretical studies on local phase distribution and boiling crisis were carried out at several institutions. The main challenge of the CFD approach to predict CHF is the description of local equivalent bubble diameter and the source terms related to interfacial mass, momentum and energy exchange.
- Heat transfer of SC fluids belongs to the most investigated phenomena in the area of SCWR thermal-hydraulics. In the last years, several test facilities were established at various institutions. It is expected that China will become the most important country in the SCWR thermal-hydraulic studies, especially SC fluid heat transfer. Main focuses of the future activities will be on the extension of the existing test data bank, heat transfer in rod bundle geometries, the development of more reliable heat transfer correlations and the development of scaling laws.

According to the information presented at the SG-FANS workshop [2], future R&D activities of nuclear thermal-hydraulics in China will contain, but not limited to the following subjects:

- Codes development & validation

- Experimental facilities & data base
- Measurement techniques
- Specific CFD issues
- CHF & post-CHF
- Core degradation
- IVR & core catcher
- Containment TH & hydrogen safety
- Passive safety system & natural circulation
- In-containment fission product behaviour
- Education & training

ACKNOWLEDGEMENT

The authors gratefully acknowledge the contributions of the Chinese colleagues, especially those from SJTU and SNPRI.

REFERENCES

[1] State Council of the PRC, 2013. 12th Five-Year Plan for Energy, State Council of the PRC, January 1, 2013.

[2] SJTU, 2015. 1st Sino-German Workshop on Fundamentals of Advanced Nuclear Safety Technology (SG-FANS), www.sjtu.edu.cn/snse, March 9-12, 2015, SJTU, China

[3] X. Cheng, N.I.Tak, CFD analysis of thermal–hydraulic behavior of heavy liquid metals in sub-channels, Nuclear Engineering and Design, vol. 236, no. 18, pp. 1874–1885, 2006

[4] J.B. Xiong, Y. Yu, X. Cheng, 2014. Three-dimensional LDV Measurement of Turbulent flow in a 6x6 Rod Bundle, CFD4NRS-5 Workshop, Zurich, Switzerland, 9-11 Sept., 2014

[5] X. Cheng, N.I. Tak, Computational fluid dynamics analysis of heat transfer to heavy liquid metals in bare rod bundles. Nuclear Technology, 2007, vol.158 (2): 229-236

[6] T. Yang, X.J. Liu, X. Cheng, 2013. Investigation on heat transfer non-uniformity in rod bundle, Nuclear Engineering and Design, 2013. 265: p. 222-231

[7] Y.C. Li, 2014. Experimental and numerical investigations on temperature and two-phase distribution under ERVC conditions, Dissertation (in Chinese), Shanghai Jiao Tong University, 2014

[8] W. Chen, 2015. Private communication, State Nuclear Power Research Institute, 2015

[9] X. Cheng, T. Schulenberg, Heat Transfer at Supercritical Pressures – Literature Review and Application to an HPLWR, Wissenschaftliche Berichte, FZKA-6609, Forschungszentrum Karlsruhe, May 2001

[10] M. Zhao, H.Y. Gu, X. Cheng, 2014. Experimental study on heat transfer of supercritical water flowing downward in circular tubes, Annals of Nuclear Energy, 2014, 63: 339-349

[11] S.Y. Zhang, 2015. Investigation on flow and heat transfer of supercritical fluids and the scaling feasibility, Dissertation (in Chinese), Shanghai Jiao Tong University, 2015