日本原子力学会　熱流動部会　　AESJ Thermal Hydraulics Division

THERMAL HYDRAULICS

熱流動部会ニュースレター（第 70 号）

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研究開発の現状紹介

Nuclear Thermal-hydraulic Research Laboratory (XJTU-NTRL),
School of Nuclear Science and Technology, Xi'an Jiaotong University
Suizheng Qiu, Guanghui Su, Wenxi Tian, Yingwei Wu, Daling Zhang

I) GENERAL　INFORMATION

The School of Nuclear Science and Technology (SNST) of Xi'an Jiaotong University (XJTU) can be traced into 1958, about at the same time with China nuclear industry and it is one of the oldest nuclear energy departments in China. The Nuclear Thermal-hydraulic Research Lab (NTRL) of XJTU was established simultaneously. There are now two professors, one associated professor, two assistant professors, two engineers, 20 doctor students and 30 master students in XJTU-NTRL.

Prof. Suizheng Qiu

Prof. Guanghui Su

Associate Prof.
Wenxi Tian

Assistant Prof.
DaLin Zhang

Assistant Prof.
Yingwei Wu

The main research fields of XJTU-NTRL are nuclear reactor thermal-hydraulics, safety and related issues. The main research activities are listed as below:

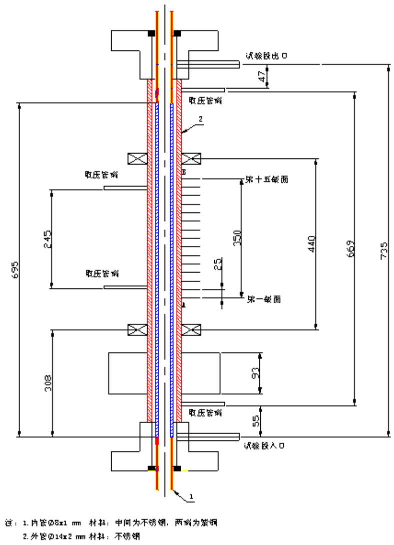
- Fundamental Research on Flow And Heat Transfer In Narrow Channels
- Application of Advanced Numerical Method In Nuclear Thermal-Hydraulics
- Instability Analysis in Complex Nuclear Systems
- Code Development on Thermal-Hydraulics and Safety Analysis of Nuclear Reactors
- Severe Accident and Safety Analysis of Advanced Light Water Reactors
- Research and Design of Generation Ⅳ Nuclear Systems(SFR And MSR)

II)BRIEF INTRODUCTON TO RESEARCH ACTIVITIES

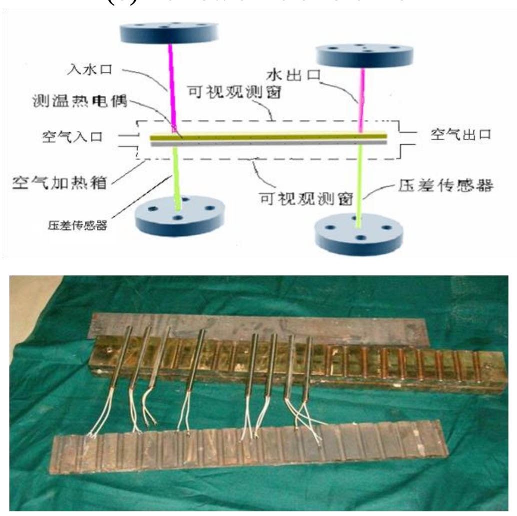
1. Fundamental Research on Flow And Heat Transfer in Narrow Channels

Experimental investigations have been comprehensively conducted by XJTU-NTRL to evaluate the thermal-hydraulic performance of new type heat transfer element in nuclear systems such as once-though steam generator(OTST), integral plate type fuel element etc. The flow and heat transfer behaviors in narrow annular channels and rectangular channels have been experimentally studied. Fig. 1 shows the diagram of high pressure water test loop and test sections.

(a) Diagram of water test loop



(b) Narrow annular channel

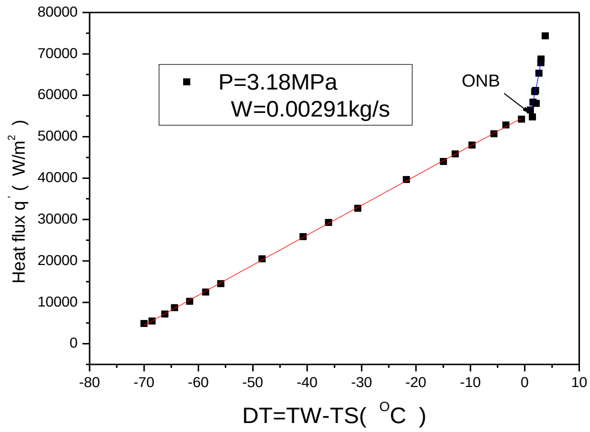


(c) Narrow rectangular channel

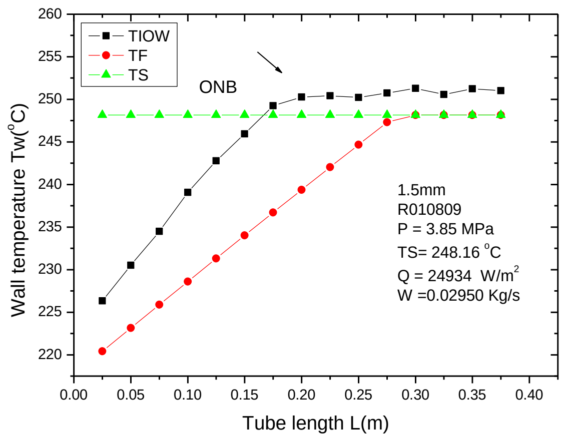
Fig. 1 Water test loop and test sections

1.1 Flow and heat transfer characteristics in annular narrow channels

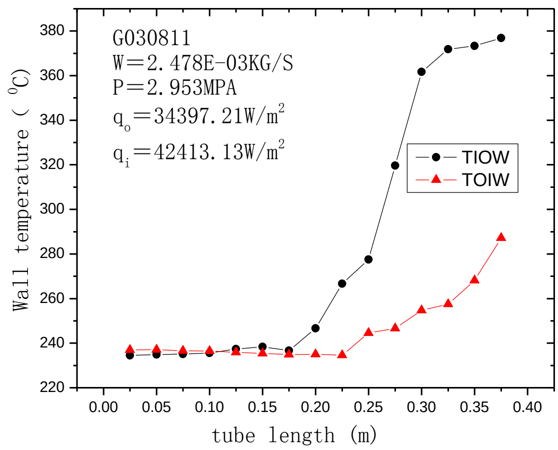
The characteristics of Onset Nucleate Boiling(ONB) and Dryout(DO) were first investigated in narrow annuli with the gap size of 1.0, 1.5, 2.0 mm, respectively. Then the flow and heat transfer behaviors in regions of subcooled water, subcooled boiling, saturated boiling, post-dryout, superheated vapor were analyzed. The results show that laminar flow turned to turbulent flow at a smaller Reynolds number. The decrease of the gap size reduced the friction factor and the coefficient of heat transfer. The ONB occurs at a higher heat flux compared with conventional tubes.



(a) Measurement of ONB-I



(b) Measurement of ONB-II



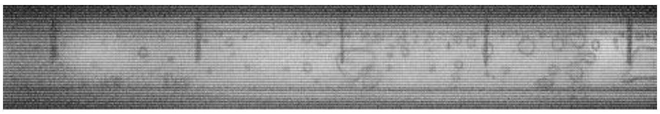
(c) Measurement of dryout point

Fig. 2 Measurement of ONB and Dryout point in narrow alular channel

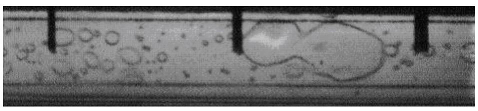
The coefficient of boiling heat transfer, which was less influenced by mass flow rate and quality, increased with heat flux. The evaporation, which related to heat flux, and nucleate boiling were believed play important roles on boiling heat transfer. The coefficients of boiling heat transfer in 1.0 and 2.0mm gap size annuli are large than in 1.5 mm gap size annuli. The decrease of gap size does not enhance nucleate boiling heat transfer. The ratio of heat flux on inner and outer tubes is an important parameter on ONB and CHF. The forced convection between the heated wall and the vapor plays the major role(with the percent not less than 90%) in post dryout region. And the heat transfer by droplets direct contact to the tube wall and the interfacial heat transfer of droplets in superheated vapor also have an indispensable contribution. The radiation heat transfer could be neglected.

1.2 Flow patterns and transition in rectangular narrow channels

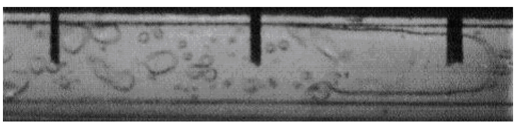
Visual two-phase thermal-hydraulic experiments have been performed in rectangular quartz glass tubes (1.0×20×500mm and 2.0×20×500mm) heated by high temperature air at atmospheric pressure. With visualization methods, four flow patterns, namely, dispersed bubble flow, slug flow, churn flow and annular flow, are recognized and captured. The flow patterns and characters in rectangular narrow channels are shown in Fig.3. The void fraction at transition point from bubble flow to slug flow is larger than that of general channel. However, the void fraction at transition point decreased with heat flux. The water flow pattern classifications in rectangular narrow channels have been deduced.



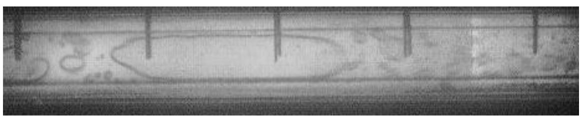
(a) Bubble flow



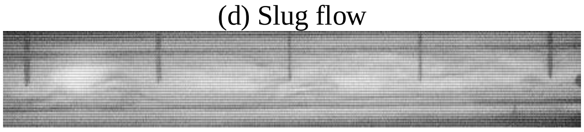
(b) Coalescence behavior of bubbles



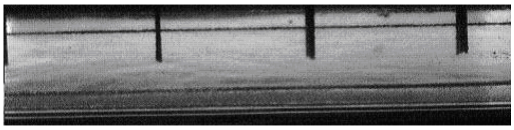
(c) Formation of slug flow



(d) Slug flow



(e) Churn flow



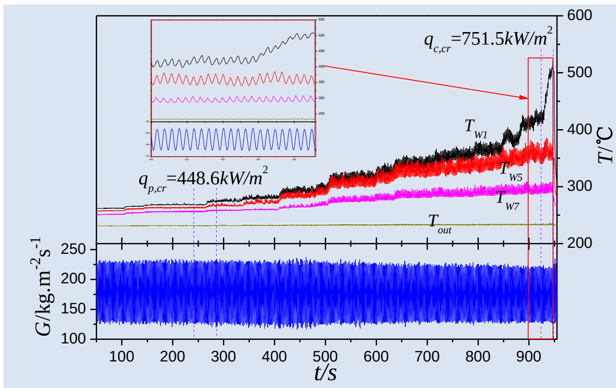
(f) Annular flow

Fig.3 Flow patterns and characters in rectangular narrow channels

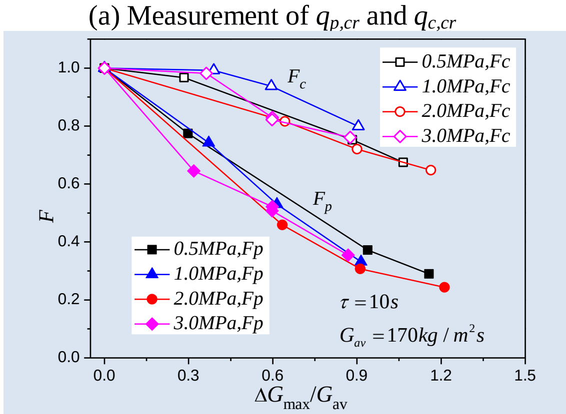
1.3 Flow and heat transfer in rectangular narrow channels

The single-phase and two-phase thermal-hydraulic experiments have been performed in rectangular narrow channels (1.0×60×800mm, 1.8×60×800mm and 2.5×60×800mm) at 1.0-6.0MPa.

For single-phase flow, laminar flow turned to turbulent flow also at a smaller Reynolds number. The Nusselt number was smaller than in general channel and it increased with gap size. The heat transfer got worse in narrow rectangular channels compared with conventional channels. In two-phase flow region, the pressure drop increased with the decrease of gap size at the same outlet quality. The frictional pressure drop of gas-liquid two-phase flow was restricted by the ratio of aspect ratio. The boiling heat transfer in rectangular narrow channels was affected by both nucleate boiling and forced-convection. Nucleate boiling was dominant at narrower channels. The decrease of gap size caused heat transfer deterioration.



(a) Measurement of $q_{p,cr}$ and $q_{c,cr}$



(b) Influence of ($\Delta G_{max}/G_{av}$) on $q_{p,cr}$ and $q_{c,cr}$

Fig. 4 Periodic dryout($q_{p,cr}$) and continuous dryout($q_{c,cr}$)

1.4 Experimental study of CHF under oscillatory flow condition

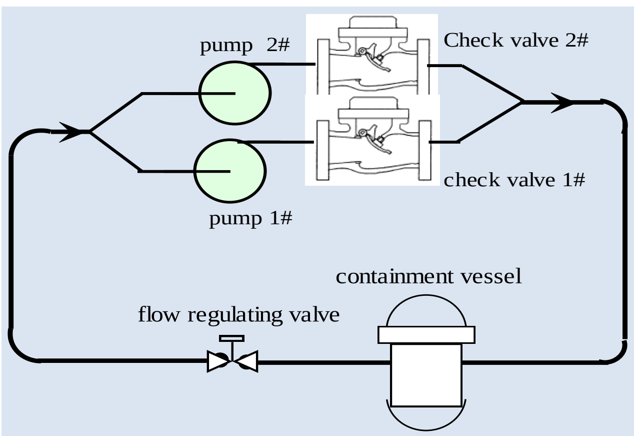
The transient critical heat flux (CHF) experiments under oscillatory flow condition have been carried out in vertical tube under forced sinusoidal inlet flow oscillation condition (oscillation period in 1, 2, 5 and 10s, normalized amplitude of flow oscillation, $\Delta G_{max}/G_{av}$, in 0-3.5). Both periodic dryout($q_{p,cr}$) and continuous dryout($q_{c,cr}$) have been captured as shown in Fig. 4a. The periodic dryout occur at a much lower wall heat flux compared with stable-flow CHF. Both $q_{p,cr}$ and dryout $q_{c,cr}$ decrease obviously with the increase of oscillation amplitude of mass flux($\Delta G_{max}/G_{av}$) as shown in Fig. 4b.

2. Application of Advanced Numerical Method In Nuclear Thermal-Hydraulics

Many advanced methods have been used by XJTU-NTRL to analyze thermal-hydraulic problems. These methods include Artificial Neural Network (ANN), Genetic Algorithm (GA) and Wavelet Analysis (WA), Method of Characteristics (MOC) line, Moving Particle Semi-implicate (MPS) method and Computational Fluid Dynamics (CFD) method.

Su et al. applied ANNs to predict CHF under low pressure and oscillation conditions at natural and forced circulation flow. A genetic neural network (GNN) model has been set up to apply genetic algorithm to optimize BP neural network weight and threshold. The GNN was applied successfully to predict both the characteristic points of flow boiling curves and CHF. Based on wavelet analysis theory of signal singularity detection, Su et al. successfully applied the wavelet modulus maxima to detect the characteristic points of flow boiling curves. The detection results by wavelet modulus maxima detection have a good agreement with experimental data.

The method of characteristic line (MOC) has been used to investigate the check valve-induced water hammer phenomena in a parallel pump feedwater system during the alternating startup process. The typical results of the check valve-induced water hammer phenomena are shown in Fig. 5.



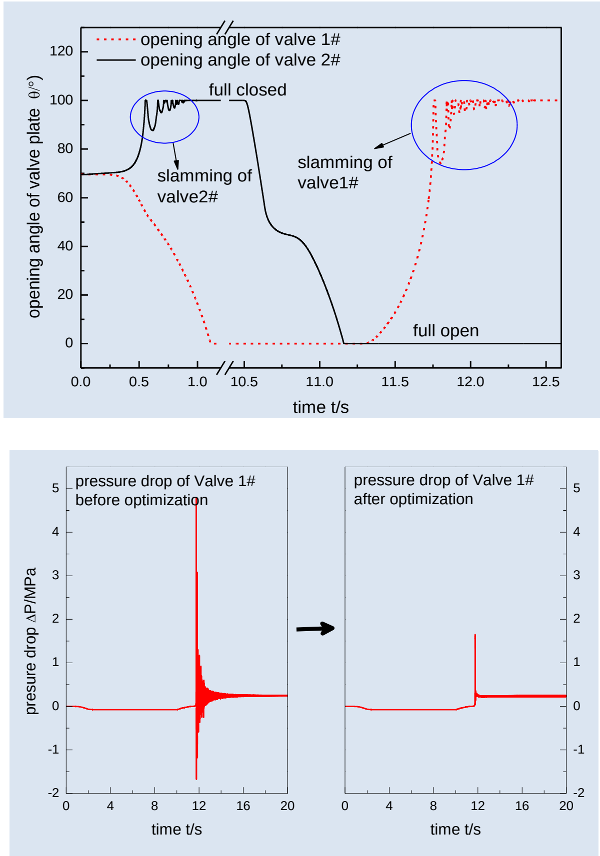


Fig. 5 Calculation results of water hammer phenomena using MOC

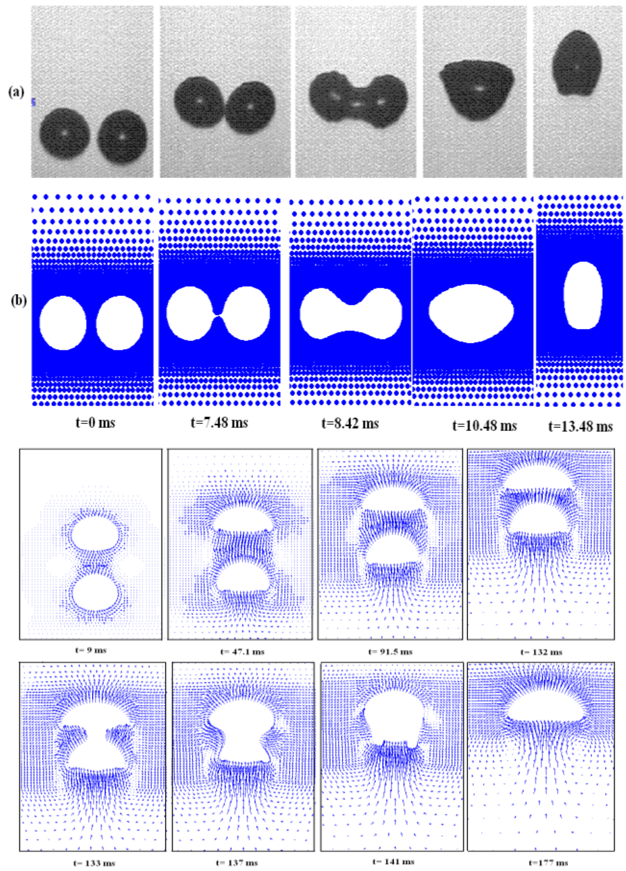
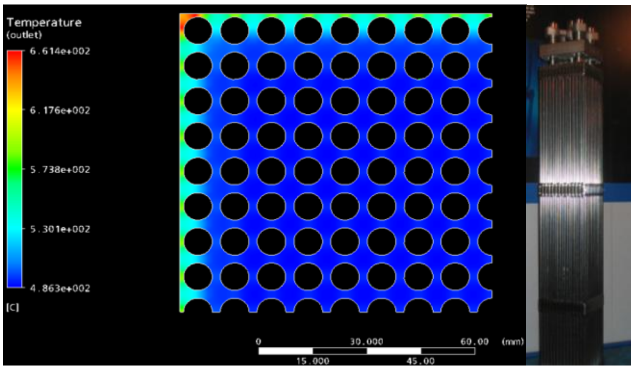


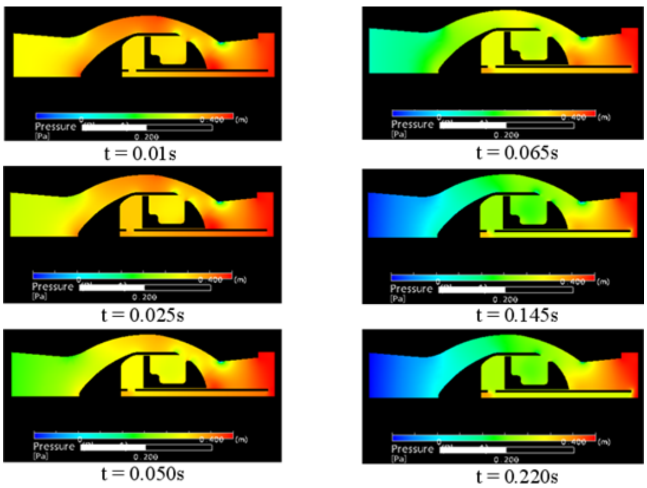
Fig. 6 Coalescence of two bubbles

MPS method, which was first proposed by Koshizuka and Oka (1996), is one of modern particle methods. Tian and Chen et al. have investigated the transient coalescence characteristics of two bubbles (horizontal and vertical direction) using MPS method, and some fundamental results on bubble dynamics have been obtained. Some results are shown in Fig.6.

Commercial CFD tools such as CFX have been adopted by XJTU-NTRL to evaluate the flow and heat transfer characteristics of new-designed fuel assembly(Fig. 7a), internal flow of dual-crashed check valve(Fig. 7b) etc. A very comprehensive thermal-hydraulic performance of nuclear systems can be obtained by using the 3-D CFD method coupling with sub-channel analysis code i.e. COBRA IV.



(a) Temperature distribution in FA cross section



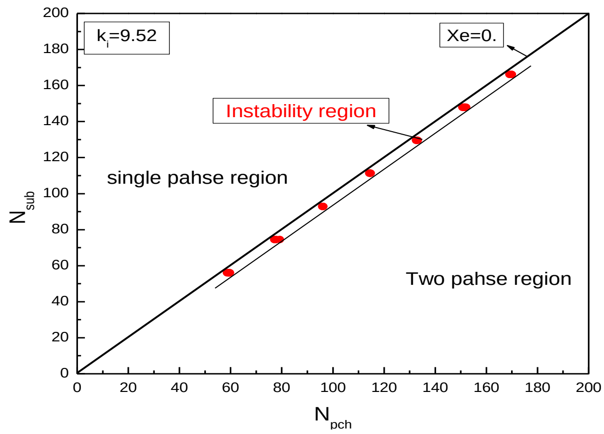
(b) Internal pressure distribution in dual-crashed valve

Fig. 7 Application of CFD in nuclear thermal-hydraulic

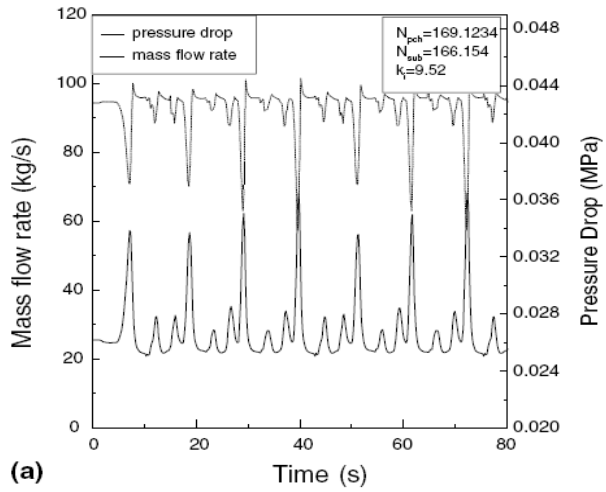
3. Instability Analysis in Complex Nuclear Systems

3.1 Density wave oscillation (DWO) of CARR

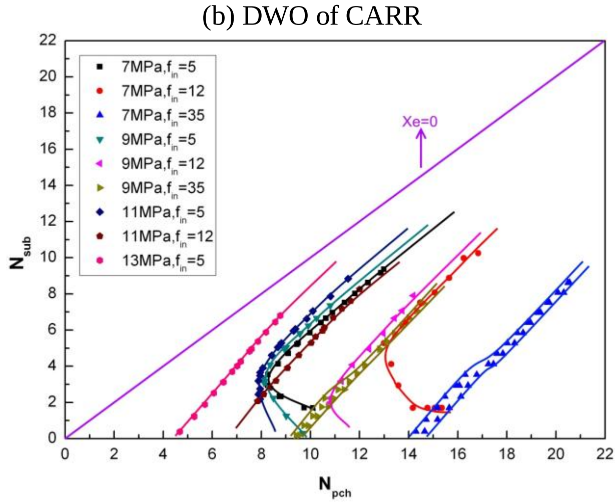
The two-phase flow instability at natural circulation condition of China Advanced Research Reactor (CARR) has been numerically investigated using independently developed code by XJTU-NTRL. As shown in Fig. 8a, the results indicate the presence of an instability region under the conditions of low equilibrium quality in the outlet at low pressure. It is a special kind of DWO(Fig. 8b) that occurs in very low equilibrium quality region with the characteristics of geysering and 'Type-I' DWO. The variations of the mass flow rate, the pressure drop and the boiling boundary are analyzed separately.



(a) FI instability boundary for CARR



(b) DWO of CARR



(c) FI boundary of twin-channel system

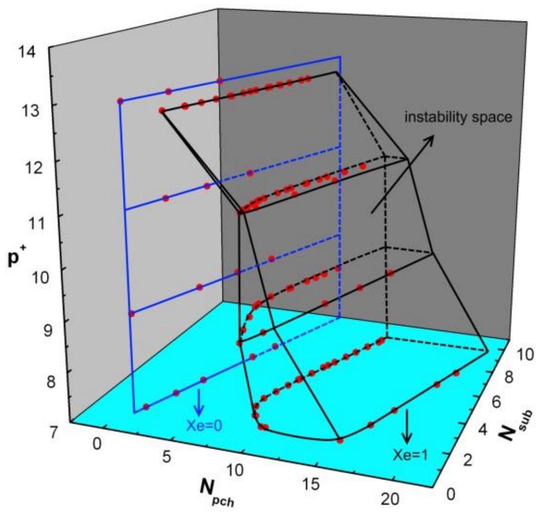
Fig. 8 Flow instability boundary

3.2 Flow instability in Parallel multi-channel system

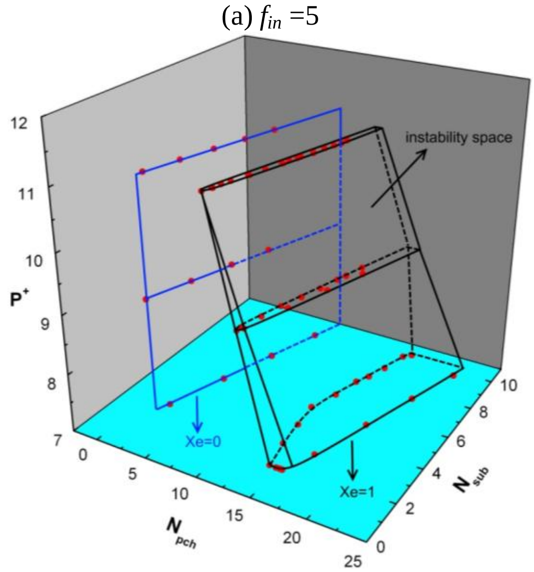
The two-phase flow instability between multi-channels (FIBM) was investigated and instability boundaries were obtained on the parameter plane of subcooling number and phase change number(Fig.8c). FIBM was comprehensively studied at different system pressures, different inlet resistance coefficient and asymmetric heating. A 3-D instability space was obtained in the parametric plane of the subcooling, phase change numbers and dimensionless pressure. A concept of instability space(or instability reef) is firstly proposed by XJTU-NTRL and it is greatly affected by inlet resistance coefficient and pressure as shown in Fig. 9.

3.3 Flow instability under rolling condition

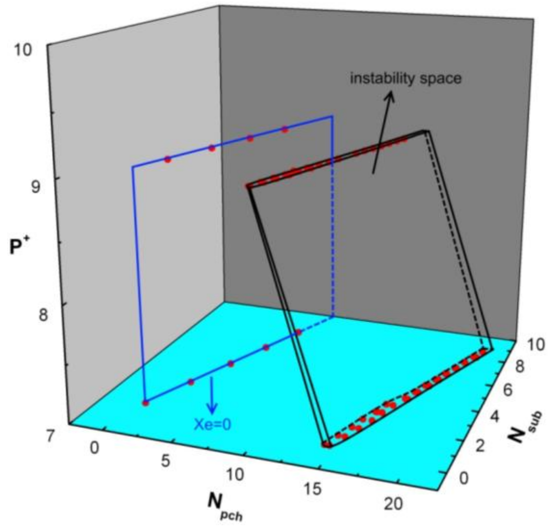
The two-phase FIBM is studied under ocean conditions theoretically. The influence of ocean condition(i.e. rolling etc.) on the FIBM is analyzed and the additional force(inertia force, centrifugal force etc.) was considered in solving momentum equations. The instabilities boundaries and stability spaces of the multi-channel system under rolling condition are obtained. Compared with Fig. 8c, the FI boundary at rolling condition is quite different from that of stational condition. The results also show that the influence of pressure on FI instability reef is quite small at rolling condition.



(a) f_{in} =5



(b) f_{in} =12



(c) f_{in} = 35

Fig.9 Influence of inlet resistance coefficient on 3-d instability space (instability reef)

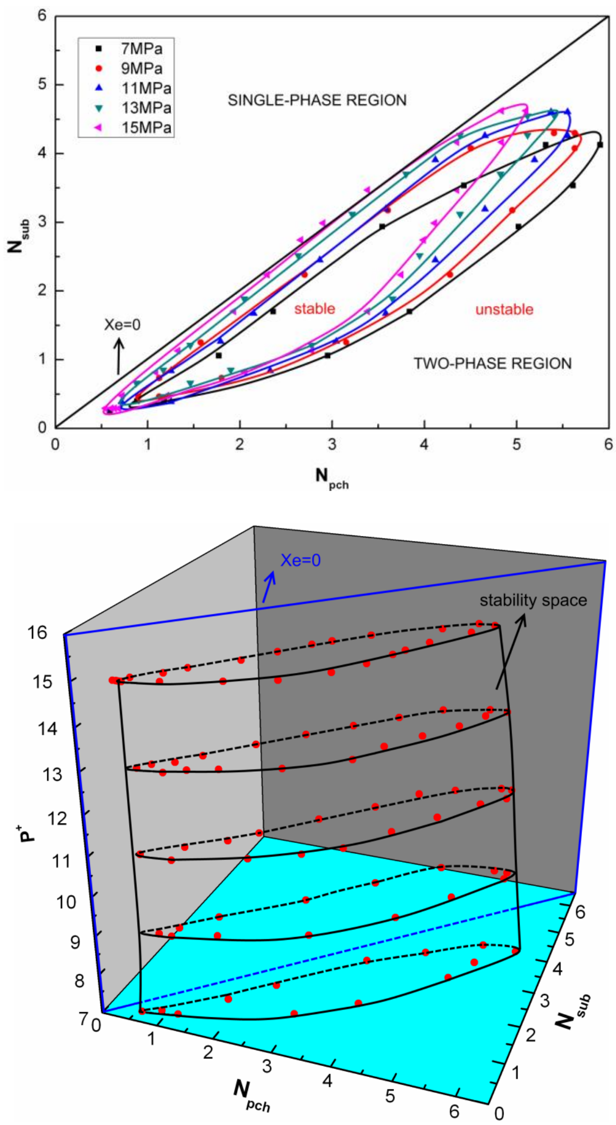


Fig. 10 Flow instability boundary and instability reef at rolling condition

4. Severe Accident Analysis of Advanced Light Water Reactors

Severe accident analysis is one important research issue in XJTU-NTRL. Both experimental investigation and numerical computation have been conducted. The related research includes IVR-ERVC analysis, coolability of molten debris bed, SBLOCA analysis and so on.

4.1 IVR-ERVC analysis

In-Vessel Retention (IVR) of core melt is a key severe accident management strategy adopted by advanced light water reactors (ALWRs), AP600, AP1000 etc. External Reactor Vessel Cooling (ERVC) is a novel severe accident management for IVR analysis. IVR analysis code in severe accident (IVRASA) has been developed to evaluate the safety margin of IVR for advanced PWR with anticipative depressurization and reactor cavity flooding in severe accident. IVRASA, a point estimate procedure has been developed for modeling the steady-state endpoint of two core melt configurations: two-layer melt configuration and three-layer melt configuration, as shown in Fig.11.

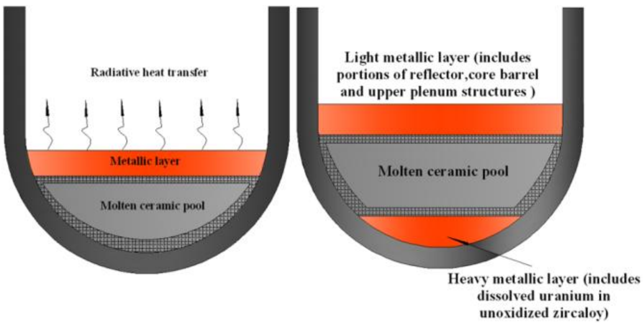


Fig. 11 Two kinds of core melt configurations

Fig. 12 Molten zirconium-4 ally

4.2 Experimental study of coolability of molten debris bed

Experimental study on coolability of particulate core-metal debris bed with Oxidization was conducted by Professor Su. Formation mechanism and migration principle of hot spot, fragmentation and enhanced heat transfer in zircaloy debris bed were investigated experimentally. Molten zirconium-4 ally is shown in Fig. 12.

4.3 SBLOCA analysis

The model of SBLOCA of APWR i.e. AP1000 was developed to investigate the transient characteristics of the main parameters of primary system using SCDAP/RELAP5 MOD3.4. The node map of SBLOCA of AP1000 and calculation results are shown in Fig. 13.

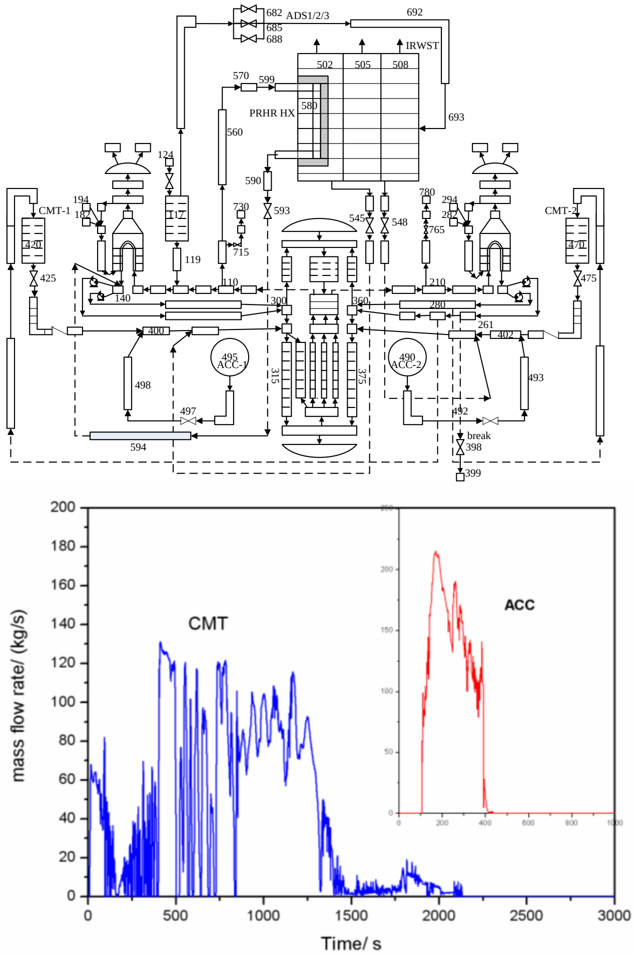
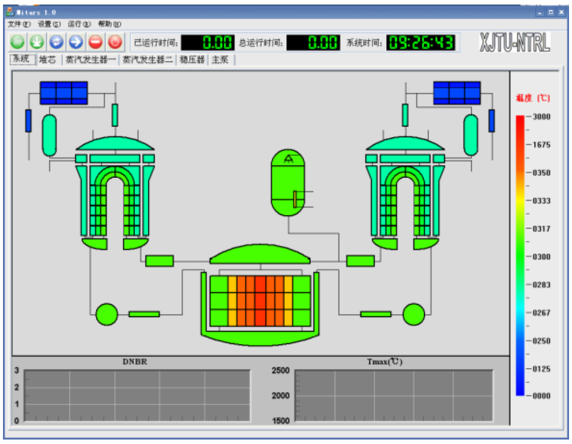


Fig. 13 The node map of SBLOCA and results of mass flow rates of accumulator and CMT

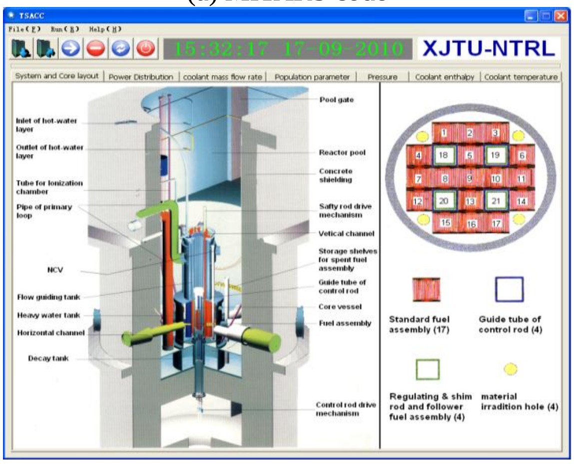
5. Code Development on Thermal-Hydraulics and Safety Analysis of Nuclear Reactors

More than twenty codes have been developed independently by XJTU-NTRL for thermal-hydraulic and safety analysis of various nuclear reactors or systems including Qinshan NPP (PWR), China High Neutron Flux Research Reactor, Xi'an Pulsed Reactor, China Advanced Research Reactor (CARR), Passive residual heat removal systems of advanced PWR, AC600, AP1000, High Temperature gas-cooled Reactor (HTR-10), Molten Salt Reactor (MSR).

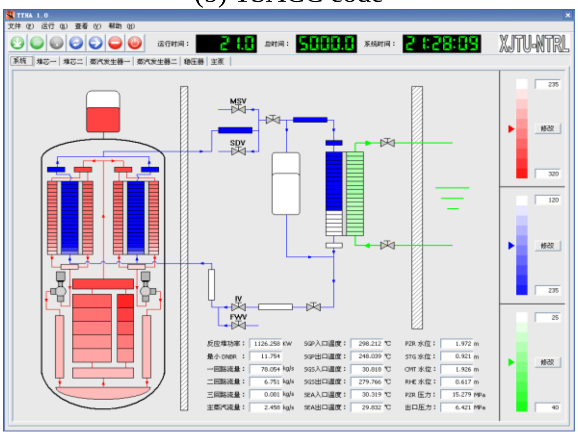
a) The Microcomputers Steady-state Analysis code for Reactor Systems (MISARS) and the Microcomputers Transient Analysis code (MITARS) were developed in 1995 using visualization technology. MISARS and MITARS are used to analyze the transient characteristics of nuclear power system. b) MACORS (Microcomputer Analysis Code of Reactor system) can analyze the steady-state natural circulation, transient thermal-hydraulic behaviors, residual heat removal capacity of advanced PWR such AP1000.



(a) MITARS code



(b) TSACC code



(c) TTHAsoft code

Fig. 14 Visual interface of developed TH and safety analysis code by XJTU-NTRL

C) TSACC was developed to analyze the transient thermal-hydraulic and safety behaviors and of CARR. Modular design, visualization input, real-time processing and dynamic visualization output were achieved using Microsoft Visual Studio.NET which makes TSACC much more convenient for further application. d) TTHAsoft was developed to analyze the transient thermal-hydraulic characteristics of Chinese advanced pressurized water reactor (AC-600) with two loops in the event of the loss of feed-water accident, the double loops loss-of-flow accident, and the reactivity insertion accident.

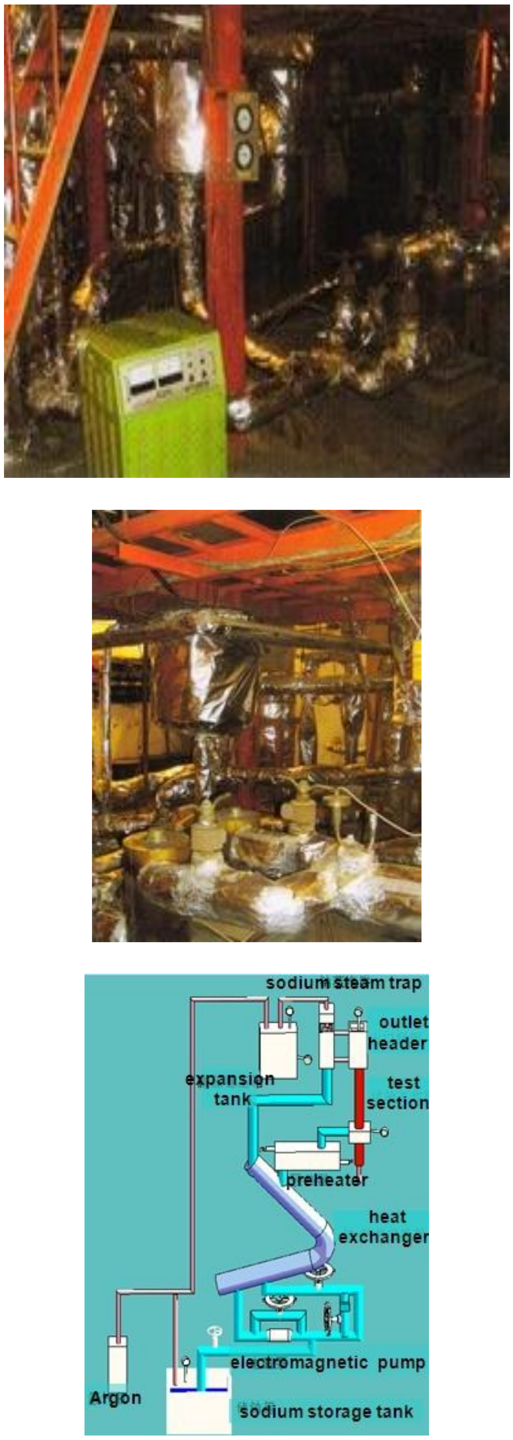


Fig.15 Liquid sodium thermal hydraulic test loop in XJTU-NTRL

6. Research And Design of Generation IV (SFR And MSR)

6.1 Sodium cooled Faster Reactor (SFR)

SFR (Sodium cooled Fast breeder Reactor) is an important fourth-generation reactor, and a lot of research work related SFR has been carried out in last ten years in XJTU-NTRL. Sodium experimental loop was established in 1984, and it is an important platform for experimental study of sodium flow and boiling heat transfer. Sodium loop is shown in Fig.15. Sodium boiling heat transfer experiment of single rod was conducted. Two-phase flow instability of sodium was also investigated by experimentally and analytically. The related experimental researches were included such as high temperature liquid heat transfer, nucleate boiling, two-phase flow pressure drop, critical heat flux (CHF) etc.

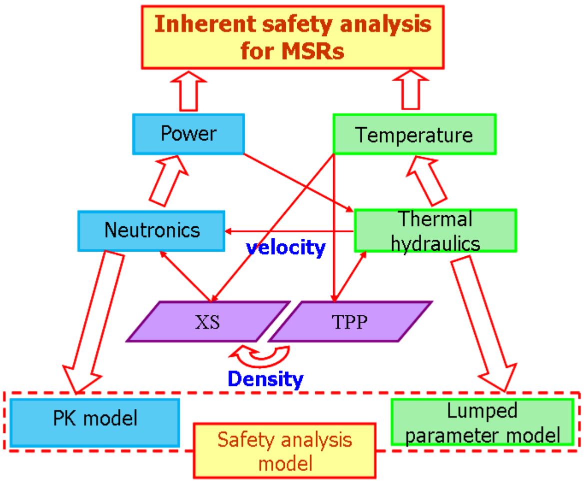


Fig. 16 Logical scheme of MSR research activities

6.2 Molten Salt Reactor (MSR)

The new concept MSR is the only liquid-fuels reactor in the six candidates of Generation IV advanced nuclear reactor systems, and it is top-ranked in sustainability because of its self-closed fuel cycle and excellent performance in waste burndown.

XJTU-NTRL concentrates on the fundamental theories related to the inherent safety of MSRs including thermophysical properties of molten salts, neutronics, thermal hydraulics, neutronics – thermal hydraulics coupling and safety analysis, the logical scheme of which is displayed in Fig. 16. The equation of state (EOS) was founded to estimate the static thermophysical properties of multicomponent molten salt systems in MSRs. With considering the flow effects of the liquid fuels, a general energy-time-space dependent neutron kinetic model for MSRs was derived from the most fundamental particle conservation principle, which was then simplified to a multi-group diffusion model by defining the corresponding group constants for numerical calculation. The advanced implicit method was improved to realize the direct coupling between neutronics and thermal hydraulics. The general exact and approximated point-kinetic (PK) models for safety analysis of MSRs was derived from the energy-time-space dependent neutron kinetic model, during which the formulation for the effective fraction of delayed neutron was obtained which is one of the most important parameters for safety analysis of MSRs.

Based on these proper theoretical models, some correct numerical algorithms were developed to design a neutronics - thermal hydraulics coupling and safety analysis code for MSRs. Collaborated with Karlsruhe Institute of Technology (KIT, original FZK), the established theoretical models and the designed code were applied to the molten salt actinide recycler and transmuter (MOSART, shown in Fig. 17). Some valuable results were obtained, which not only evaluated the safety characteristics of MOSART, but also verified the correctness of the established models and code.

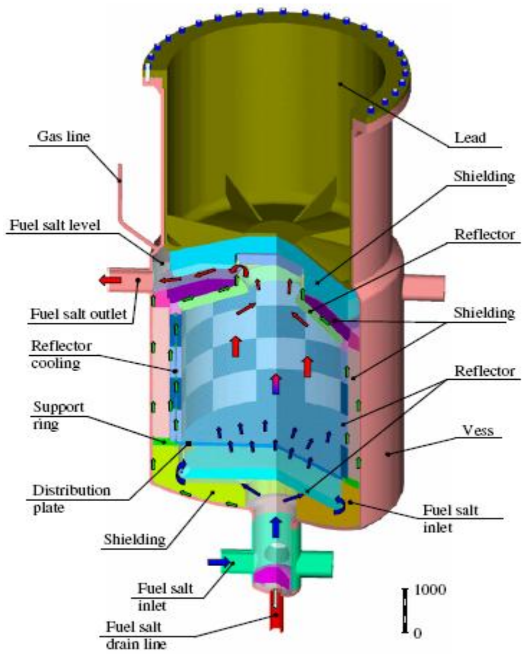


Fig. 17 Core conceptual design of MOSART

運営委員会報告

熱流動部会運営委員会(H22－1) 議事録

(1) 日時：平成 22 年 6 月 22 日(水) 13:00-16:30
(2) 場所：日本原子力学会　会議室
(3) 配布資料：
① 議事次第・平成 22 年度役員リスト
② 総務委員会活動報告
　参考資料②－①　熱流動規約（案）
　参考資料②－②　部会の指定寄付の扱いについて
　参考資料②－③　日本原子力学会熱流動部会 役員任期規定(案)
　参考資料②－④平成 21 年度収支予算及び実績表
　参考資料②－⑤組織別 長期事業・予算規模計画案
　参考資料②－⑥　NURETH13 収支決算書
③ 部会等運営委員会報告
④ 研究委員会活動報告
⑤ 国際委員会の活動概要
⑥ 広報委員会活動報告
⑦ 出版編集委員会の活動概要

議事

1. 部会長挨拶（大塚部会長）

第 1 回運営委員会の開催にあたって大塚部会長より挨拶があり、次に示す 3 テーマ、「交流・連携」、「情報提供」、「人材育成」に重点を置いて運営を行いたいとの所信が述べられた。

2. 自己紹介

平成 22 年度役員の自己紹介が行われた。

3. 研究委員会報告（山本研究委員長）

資料 4 を用いて、以下に示す議事が行われた。「シビアアクシデント時の格納容器内の現実的ソースターム評価」特別専門委員会について、2011 年春の年会で企画セッションを立案する方向で引き続き検討する。「熱水力安全評価基盤技術高度化検討 WG」について、主査の交代を確認する。

4. 総務委員会報告（大川総務委員長）

資料 2 および参考資料を用いて、以下に示す議事が行われた。

4.1. 熱流動部会規約および役員任期規定

規約改定案について、春の年会での総会以降、部会員からのコメントは無かった旨が報告された。役員任期規定に関しては、第 1 条中の「運営小委員会を構成する部会長、副部会長および運営委員（以下、役員と称する）」を「運営小委員会を構成する役員（部会長、副部会長および運営委員）」と修正した上で、秋の大会の部会総会に諮ることとした。また、各部会の設置目的をテーブル化することが議論された。

4.2. 熱流動部会予算

H21, H22 年度は、次年度繰越金の 10%が IT 化基金（H22 は費目が異なる可能性あり）として本部予算に繰り入れられるが、H23 以降は廃止される見込み。また、H22 年度に、NURETH-13 からの寄付金がある見込み。参考資料⑤を用いて、H23～27 年度長期計画について説明があり、年度予算と併せて承認された。参考資料⑥を用い、NURETH-13 の会計報告と大川総務委員長による監査結果が報告された。熱流動部会が主催する会議については、透明性を高めるため、原則として総務委員長による監査を実施することを確認した。

4.3. 部会等運営委員会の報告

資料 3 を用いて、西総務副委員長より、6/22 に開催された部会等運営委員会の概要が説明された。

4.4. 安全工学シンポジウム

安全工学会が主催（原子力学会は共催）する安全工学シンポジウム（7 月 8～9 日）の準備状況が紹介された。

4.5. その他

シンポジウムや学会の主・共催に関して、学会に依頼文の提出が必要であることを確認した。

5. 企画委員会報告（木村企画委員長）

Dr.フォーラムの準備状況が報告された。昨年度実施した技術紹介は、実施する場合、三菱重工殿または東芝殿に依頼すること、また、関係大学・機関に博士号取得者の参加調整を開始することが説明された。

6. 国際委員会活動概要報告（大野国際委員長）

資料 5 を用い、大野委員長より NTHAS7 と日韓学生セミナーの準備状況が説明された。日韓学生セミナーについて、キーノート講演のテーマおよび日本側講演者について意見交換した。日韓学生セミナーに付随する見学会について、見学者の受け入れ条件を韓国側に確認することとした。また、日韓学生セミナー参加支援の応募者は協賛部会の会員に限ることを確認するとともに、応募が想定数を超過した場合の対応などについて協議を行った。

7. 広報委員会活動報告（玉井広報委員長）

資料 6 を用い、玉井広報委員長から、ホームページの更新状況とニュースレター、メーリングリストの発行状況が報告された。ニュースレター69 号は、ドクターフォーラムの案内を掲載する必要などがあるため、発行時期を調整中である。WEB のトップページに、熱水力ロードマップへのリンクを張ることが承認された。本件に関連し、特別専門委員会等の活動報告を確実に行うべきであることを確認した。ニュースレターの内容に関して、海外の研究施設紹介、ANS の状況、部会構成員の推移など掲載について意見交換した。

8. 出版編集委員会活動報告（宋出版編集委員長）

資料 7 を用いて、宋出版編集委員長より、H19～H21 年度の投稿論文数の推移、NTHAS6 特集号の編集状況が報告された。NTHAS6 特集号には 9 編が掲載予定。また、論文誌の運営について、大手出版社への委託が検討されていることが紹介された。

以上

平成 22 年度　熱流動部会役員

部会長	大塚　雅哉	（日立）
副部会長	片岡　勲	（大阪大学）
総務委員長**	大川　富雄	（大阪大学）
総務副委員長**	西　義久	（電力中央研究所）
広報委員長**	玉井　秀定	（JAEA）
同副委員長*	染矢　聡	（産業技術総合研究所）
研究委員長**	山本　泰	（東芝）
国際委員長**	大野　修司	（JAEA）
同副委員長*	池田　秀晃	（三菱重工）
企画委員長*	木村　暢之	（JAEA）
出版編集委員長**	宋　明良	（神戸大学）
同副委員長*	波津久　達也	（東京海洋大学）
表彰委員長	秋本　肇	（JAEA）
海外担当役員	二ノ方　壽	（東京工業大学）

*:任期2年の1年目、**:任期2年の2年目

国際会議カレンダー（Web のみに掲載）

熱流動部会のホームページ http://wwwsoc.nii.ac.jp/aesj/division/thd/ **より最新の情報を入手して下さい。**

＜編集後記＞

ニュースレターへの原稿は，随時受付を行っております．研究室紹介，会議案内，エッセイ等寄稿お願い致します。70号では初めての試みとして国外の研究室を紹介致しました．またニュースレターに関するご質問，ご意見，ご要望等ありましたら，ぜひ下記宛にe-mailをいただければ幸いです．熱流動部会に入会したい方，入会しているがメールが届かない方が身近におられましたらご相談ください.

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熱流動部会のホームページ：
http://wwwsoc.nii.ac.jp/aesj/division/thd/
からニュースレターの PDF ファイルは入手可能です。