Status of Ongoing Research on SCWR Thermal-hydraulics and Safety

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I. INTRODUCTION

SCWR (**S**uper**C**ritical pressure **W**ater-cooled **R**eactor) is an advanced reactor concept which has advantages in improving economics while utilizing most of the existing PWR and BWR technologies as well as fossil fuel power plant technologies. The Generation IV International Forum selected the SCWR as one of the viable candidates of a nuclear power plant to be deployed by 2030, especially for economic electricity generation. In order to organize and support the multi-lateral collaboration among the member countries, the SCWR Steering Committee has been established in March, 2007, and several Project Management Boards (PMBs) such as Thermal-Hydraulics and Safety, Material and Chemistry, System Integration and Assessment, and Fuel Qualification have followed.

Since the formation of the provisional SCWR Thermal-Hydraulics and Safety Project Management Board (TH&S PMB) in November 2004, the participating countries, Canada, EU (as a consortium), Japan (as a consortium), and Korea have been working hard on the preparation of the Project Plan (PP). This plan is a kernel part of the Project Arrangement (PA) and will serve as a basic document for the forthcoming multi-lateral collaboration. The endorsement of the PA by the participating countries is expected to be completed no later than the end of 2009. The PP describes the coordinated research activities in the technology development area (TDA) of SCWR TH&S. In the System Research Plan (SRP), the required research items have been identified and these include heat transfer, hydraulic characteristics, critical flow, identification of safety requirements and evaluation, stability, development of system codes and relevant methodologies, subchannel analysis, and simulation of system performance and behavior during transient and accident. Most of these items can be performed on an individual basis, but others may require the integrated efforts of all or some participants. The PP describes the framework of the collaboration scheme, required resources, estimated schedule and deliverables. It will be reviewed annually by the PMB members and may be modified on the suggestion of the Signatories and the approval of the System Steering Committee.

In the meantime, a considerable amount of work has been accomplished by the member countries and arrangements will be made for those identified as shared items as soon as the PA is signed. In this paper, the accomplishment of the member countries in the SCWR thermal-hydraulics and safety area is described briefly. Besides the member countries China is actively working on the SCWR thermal-hydraulics research but it will not be covered in this paper. Russia is also considering joining this PMB but its decision has not yet been made.

II. CANADIAN THERMAL-HYDRAULICS AND SAFETY PROGRAM

Thermal-hydraulics characteristics at super-critical water-flow conditions are required in support of the design and qualification of the fuel bundle and safety analyses for the SCWR. GIF participants in the SCWR development are preparing a Project Plan for thermal-hydraulics and safety research work. The plan lists tasks required for completing the conceptual design of the SCWR, and covers key areas such as heat transfer, critical flow, instability, development of analytical toolsets for supercritical-water applications, and preliminary safety analyses. Completing these tasks will demand a large coordinated effort between research organizations and the academic community.

The Canadian contribution to various key areas of the GIF SCWR Thermal-hydraulics and Safety Project has been identified in the Project Plan. It consists of projects directly relevant to the CANDU SCWR fuel and core designs at AECL and fundamental research and development (R&D) projects related to the SCW flow and heat transfer at various Canadian universities.[1] In addition, AECL has initiated other collaborative projects with Canadian universities (with proposed support from the Ontario Research Fund) and Chinese universities to develop the future reactor design. Information from these projects is also applicable to the Generation IV SCWR design and will be included as part of the Canadian contribution to GIF. The Thermal-hydraulics and Safety projects in the grant program focus on improving / developing heat-transfer prediction methods for supercritical heat transfer in tubes and bundles, examining the stability and critical-flow characteristics of supercritical flow, and performing simulations of the depressurization phenomena through small breaks at supercritical conditions. The tube-data-based prediction method for supercritical heat transfer is applied in subchannel analyses, while the bundle-data-based prediction method is implemented for safety analyses.

The design criterion for the CANDU-SCWR is based on the cladding temperature limit for normal operation and trip analyses. Experimental data on heat transfer are crucial in establishing this limit accurately.[2] A database on supercritical heat transfer in tubes has recently been assembled.[3] It is being applied to assess various correlations and, if necessary, to improve prediction accuracy.

Figure 1 illustrates variations in the heat-transfer coefficient as a function of temperature for supercritical water flow inside a 10-mm inside diameter (ID) tube.

A project has been initiated at the University of Ottawa to develop a look-up table for heat transfer covering trans-critical conditions (*i.e.*, both the near-critical region and the supercritical region) in tubes. Advantages of the look-up table approach include superior prediction accuracy (representing directly the database), wide-ranging applicability, and a smooth transition in tabulated values between different regions.

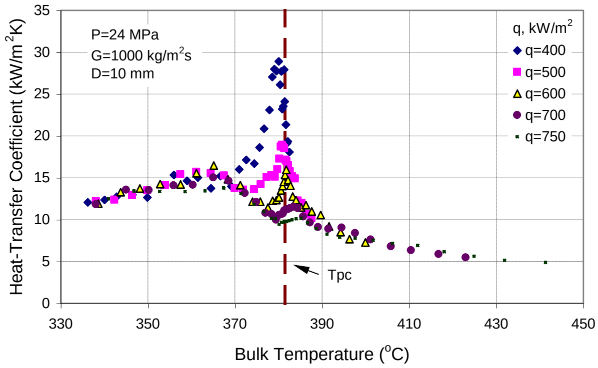

Figure 1: Heat-Transfer Coefficient in Supercritical Water Flow.

Performing heat-transfer experiments with supercritical water flow is complex and expensive due, primarily, to the harsh operating environment and the high level of required heating power. Surrogate fluids (such as carbon dioxide and refrigerants) have been suggested for replacing water in heat transfer studies. These fluids were previously utilized in studies of critical heat flux and film-boiling heat transfer at subcritical conditions. Applying these fluids reduces experimental cost and schedule, reduces test-section design and operation risk, and increases testing flexibility. This arises from the fact that supercritical conditions for surrogate fluids are less severe than those for water.

Figure 2 illustrates the range of reduced pressure and reduced temperature covered in the supercritical heat-transfer database for carbon dioxide flow.[3]

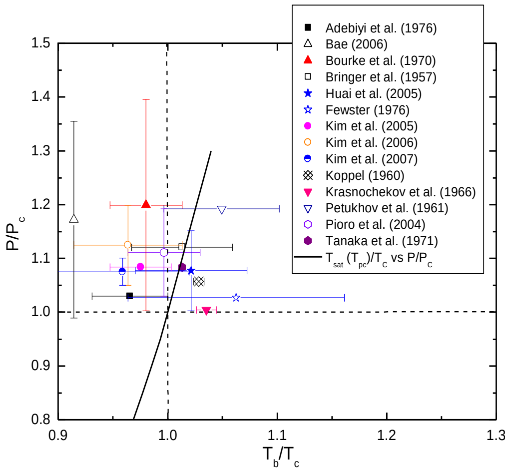

Figure 2: Range of Selected Supercritical Heat-Transfer Data for Carbon Dioxide Flow.

A project has been established to expand a sub-critical refrigerant test facility at Carleton University for supercritical heat transfer experiments. The test facility can accommodate test sections such as tubes, annuli, and small bundle subassemblies to study various separate effects on supercritical heat transfer. A tubular test section is being designed for commissioning the facility using Refrigerant-134a. The first test series examines the effect of spacing devices on supercritical heat transfer in annuli.

Large amounts of supercritical heat transfer data are available for tubes, but there is a lack of data for bundle geometries. A heat-transfer test facility has been designed for construction at the University of Ottawa (UO).[4] It employs carbon dioxide as working fluid and is capable of testing small bundle subassemblies. Key components of this test facility have been procured.

A project has been arranged to complete the construction of the facility and perform commissioning test using a tubular test section. The commissioning data will be compared against experimental data in the AECL database[3]. Another project has been established to perform heat-transfer experiments using a 3-rod bundle string in this test facility. The objective of this experiment is to generate bundle heat-transfer data in carbon dioxide flow. This data is essential for quantifying the impact of flow and enthalpy distributions in subchannels on supercritical heat transfer, and is also applicable for validating subchannel codes and computational fluid dynamic tools.

Computational fluid dynamic (CFD) tools have been widely used in support of fuel design for SCWR. These tools are based on fundamental conservation equations but depend strongly on the turbulence model selected in the calculation. Currently, there is little (or no) information on turbulence measurements in supercritical flow. A project has been awarded to the research team at the University of Ottawa to obtain turbulence measurements in a 22.9 mm tube with super–critical carbon dioxide flow. Subsequently, the measurement technique may be implemented to the 3-rod bundle.

The CANDU-SCWR may be susceptible to dynamic instability due to the sharp variation in fluid properties (such as density) in the vicinity of the critical point. This instability may lead to a high cladding temperature in the fuel, prematurely impacting on the operating and safety margins. Analytical models have been developed for predicting the onset of dynamic instability with in-phase 1D oscillations and out-of-phase 2D and 3D oscillations.[5] A project has been approved for the University of Manitoba to perform flow-stability experiments using carbon dioxide in single and parallel channels. Test data will be applied for the validation of the analytical model.

In support of the design and operation of the reactor safety (or relief) valves and the automatic depressurization system, the critical (or choked) flow characteristic must be established at supercritical conditions since current information has been obtained at subcritical conditions. This established characteristic is also required in the analysis of a postulated large-break loss-of-coolant accident event. A project has been awarded to the École Polytechnique to construct a test facility for critical-flow measurements in water at supercritical conditions. Blow-down experiments from a supercritical pressure tank to a medium pressure reservoir will be performed with discharge nozzles of different shape, size and length. Direct experimental measurements of the temperature and pressure along the discharge nozzles, and of the void fraction and flow rate at the nozzle outlet will be obtained. These data will enable accurate benchmarking of existing critical-flow models and, if needed, the development of new ones.

The basic thermal-hydraulics phenomena during hypothetical accidents involving depressurization of the reactor coolant system at subcritical water conditions (such as critical break discharge and flashing behavior) have been extensively analyzed within the nuclear industry. However, very little analysis is available for reactors near or above the supercritical pressure. A project has been initiated at McMaster University to examine depressurization characteristics for near critical and supercritical systems, taking into consideration the unique properties as the fluid transitions through the critical state. These systems include simple pipes and tank geometries (which have been previously studied at sub-critical conditions) and constricted flow passage through nozzles (simulating the small breaks phenomena). Simulation results will be compared against the experimental data at subcritical and super–critical conditions.

III. EUROPEAN CONTRIBUTION TO THE GIF THERMAL-HYDRAULICS AND SAFETY PROJECT

In Europe the High Performance Light Water Reactor (HPLWR) is currently under development. The High Performance Light Water Reactor is a Light Water Reactor (LWR) with supercritical water at 25 MPa as coolant and moderator.

A consortium of 10 partners from 8 European countries and three so-called active supporters cooperate within the "High Performance Light Water Reactor Phase 2" project which started in 2006 and will end in 2010. This project is co-funded by the European Commission. The objective of this project is to assess the feasibility of this reactor concept and to assess the economical potential of this reactor concept, see Starflinger *et al*.[6]

Most of the European research activities on the SCWR are covered by the HPLWR project in its Phase 2. The outcome of the research activities on safety and thermal hydraulics is contributing to the TH&S program within GIF. The research activities and deliverables are defined in the project plan of the TH&S project and the project plan is part of its project arrangement.

In the following two sub-sections, a brief description of the current status of the research on thermal hydraulics and the safety concept is given.

III.A. Status of research on thermal hydraulics for the HPLWR

Heat transfer from the fuel rods to super critical water is a very important research issue since it determines the temperatures of the fuel cladding. It is well known that this heat transfer is strongly influenced by large changes in the physical properties of super critical water near the pseudo critical point. The heat transfer may be enhanced or strongly decreased (called heat transfer deterioration) depending on local conditions like heat flux and flow rate of SCW. A second important aspect in the heat transfer from the HPLWR fuel rods is the wire which is wrapped around each fuel rod. The wire serves as spacer between the fuel rods and enhances mixing between the sub-channels of the fuel bundle.

As a starting point of the thermal hydraulic research work, a data base of experimental data for supercritical water in smooth heated tubes has been prepared by Loewenberg *et al*.[7] The data cover a broad range of experimental conditions for sub- and supercritical pressures. The data base has been used for the validation of CFD models that have been developed by Zhu and Laurien[8] and by Visser *et al*.[9] Besides supercritical water, also super critical CO_2 has been used for CFD code validation, showing similar physical effects.

Analytical work as well as CFD work has been performed by Palko and Anglart[10] to improve the basic understanding of heat transfer mechanisms for super critical water with a focus on the mechanisms describing the onset of heat transfer deterioration. A heat transfer mechanism identification and ranking table has been made by Anglart.[11] CFD is also used to quantify the effect of the wire wrap in representative sub-channel geometries. Recent results have been published by Himmel *et al*,[12] Chandra *et al*,[13] Laurien *et al*,[14] and Kiss *et al*.[15]

A coupled neutronics-thermal-hydraulics analysis for the HPLWR core has been developed to identify hot spots in the fuel assemblies and to verify that the material limits can be met. With these coupled codes, axial and radial power distribution of the SCWR core will be determined as well as temperature distributions for cladding, moderator and coolant.

III.B. Safety concept for the HPLWR

A safety concept for the HPLWR has been proposed by Bittermann *et al*,[16] which now needs to be worked out by means of analyses of transient and accident scenarios using state-of-the art system codes. Plant models including an accurate representation of the reactor pressure vessel (RPV) and the complex geometry of the core (three-pass core) have been developed for various thermal-hydraulic codes (RELAP5, CATHARE, APROS) as well as for coupled neutronic thermal-hydraulic codes (SMABRE/TRAB-3D, ATHLET-KIKO3D). The initial analyses focused on the hydraulics of the initial core design, which contributed to the development of an improved geometry. The analyses of the current configuration showed that the flow can now be expected to be adequate for all loads of interest. Results concerning these aspects will be published in due course.

A list of transient and accident scenarios has been compiled, which includes the supposedly most severe conditions with respect to preserving the fuel integrity. As a continuous coolant mass flow rate through the reactor is required for the once-through steam cycle implemented in the design of the HPLWR, special attention must be devoted to transients resulting in a loss of flow. Analyses are currently carried out to verify the response of the plant to a Loss of Feed Water (LOFW), including various cases with respect to pump failures, run-down times and scram intervention times. These studies aim, among others, to determine the required features of the feed-water pump-motor system, including the need for fly-wheels. Loss-Of-Coolant Scenarios have been started to be analyzed, and preliminary results have recently been obtained. These analyses are especially important because of the cool ability of the three-pass core following a sudden depressurization and the consequent fast reduction in water inventory in the core is a critical issue for the safety concept, as no natural circulation mechanism is available.

A preliminary safety system design has been proposed by de Marsac *et al*,[17] which has to be specified now in more detail to control the individual accident scenarios. For instance, the low pressure coolant injection (LPCI) system could be similar to the active LPCI system of a boiling water reactor. It could act either after depressurization through the spargers or in case of a loss of coolant accident. Investigations are currently under way for the optimization of the active heat removal systems. Additionally, a passive high pressure coolant injection (HPCI) system is investigated, which has not yet been applied to any pressurized or boiling water reactors. Detailed transient analyses will be required to decide about its feasibility.

Keresztúri *et al*,[18] started to simulate RIAs scenarios, and the results obtained show significant perturbation of the local power. Due to the very heterogeneous moderator density distribution, the results are sensitive to a great extent to the initial position of the control rods and to core loading, especially to the number and position of the assemblies containing burnable poison rods. The acceptance criteria, however, are fulfilled so far. In some cases, the hot channel temperatures are not far from the limits, which points out the necessity of the RIA analyses. In the next period, a broad range of RIAs and ATWS transient will be investigated, and the analyses will show whether modifications of the core neutronic design will be required.

IV. STUDY OF HEAT TRANSFER TO SUPERCRITICAL PRESSURE FLUID IN JAPAN

A conceptual study of the pressure-vessel type SCWR started at the University of Tokyo in 1989. The GIF SCWR concept with pressure-vessel is based on the concept that has been developed in the University of Tokyo.[19]

Two R&D projects on the pressure-vessel type SCWR with fast/thermal options are ongoing in Japan jointly by universities, research institutes and industries.[20]

A R&D project on fast option, entitled "Research and development of the Super Fast Reactor" was entrusted to the University of Tokyo in December 2005 and will be completed in March 2010. Aiming at a highly economical fast reactor, the plant concept is being developed with quantitative characteristics/performances. The databases of the thermal hydraulics and materials (including water chemistry) are being developed by experiments.

Another R&D project on thermal option, entitled "Development of SCWR in GIF Collaboration (Phase-I)", was granted to Toshiba Corporation and The Institute of Applied Energy in August 2008 and will be completed in March 2011. The purpose is to assess the viability of the thermal SCWR concept through the GIF collaboration. The R&D areas include System Integration and Assessment, Thermal-Hydraulics and Safety, and Materials and Chemistry, which correspond to the GIF/SCWR projects.

In this report the typical thermal-hydraulic test results for fast option are explained.

IV.A. Heat Transfer test for Freon[21]

Thermal-hydraulics tests at supercritical pressure conditions with water and Freon have been done to obtain heat transfer data, using tube and bundle.

Experiments are performed with a supercritical pressure HCFC22 forced circulation loop, newly set up at Kyushu University, Japan. HCFC22 is used as a substitute for water because its critical pressure and temperature of 4.99 MPa and 96.2°C are far lower than those of water (22 MPa and 374°C), and therefore the experimental conditions can be flexibly altered. Steady state tests are carried out with a single circular tube test section of 4.4.mm I.D and with a sub-bundle (Bundle-I) test section composed of seven heater rods simulating the actual fuel bundle geometry as shown in Figure 3.

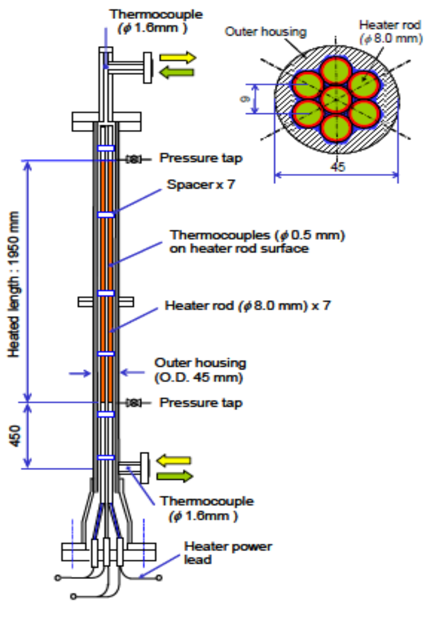

Figure 3: Sub-bundle(Bundle) test section

Figure 4 shows the typical result. In the sub-bundle channel, the occurrence of heat transfer deterioration is generally suppressed even in the upward flow, and the heat transfer is similar to that in the tube flow in the normal heat transfer region of the tube flow.

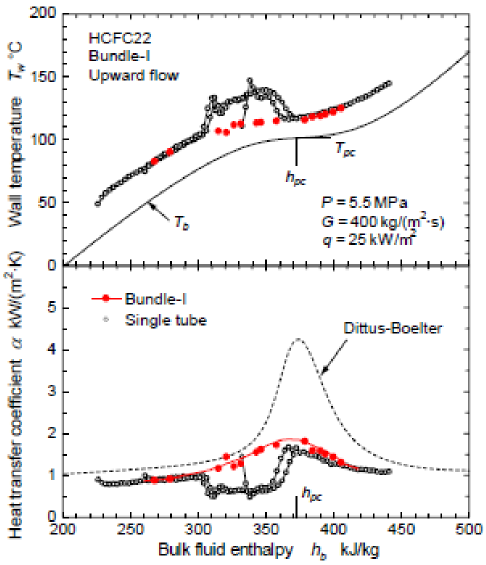

Figure 4: Typical comparison between tube and Bundle

IV.B. Heat Transfer test for supercritical pressure water[22]

Figure 5 shows a schematic drawing of the test section. A test section including a single heater rod simulating fuel rod has been fabricated and installed into a high pressure and high temperature water circulation loop in JAEA. Supercritical pressure water at 25 MPa flows in the test section and the surface temperatures of the heater rod are measured to evaluate the heat transfer coefficient around the single heater rod.

Figure 6 shows the typical data for wall temperatures and heat transfer coefficients. HTC in this Figure is the heat transfer coefficient evaluated by Dittus-Boelter correlation based on the hydraulic diameter of the test section. Run930 and Run698 are the experimental results obtained by Yamagata *et al.*,[23] using a straight tube with the ID 10 mm at the mass flux of 1156-1235 kg/m^2/s and the heat flux of 698 and 930 kW/m^2, respectively. The measured maxi–mum heat transfer coefficient in this Figure is about 29 kW/m^2/K. The maximum value and the bulk fluid enthalpy for which the maximum value occurred are lower than the predicted values. This trend is almost the same as the Yamagata's Run 930.

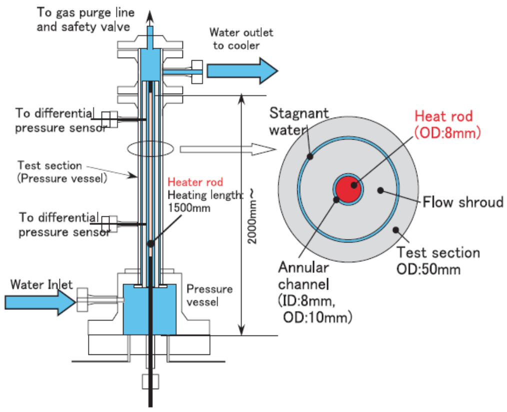

Figure 5: Schematics of test section for supercritical water around a single heater rod

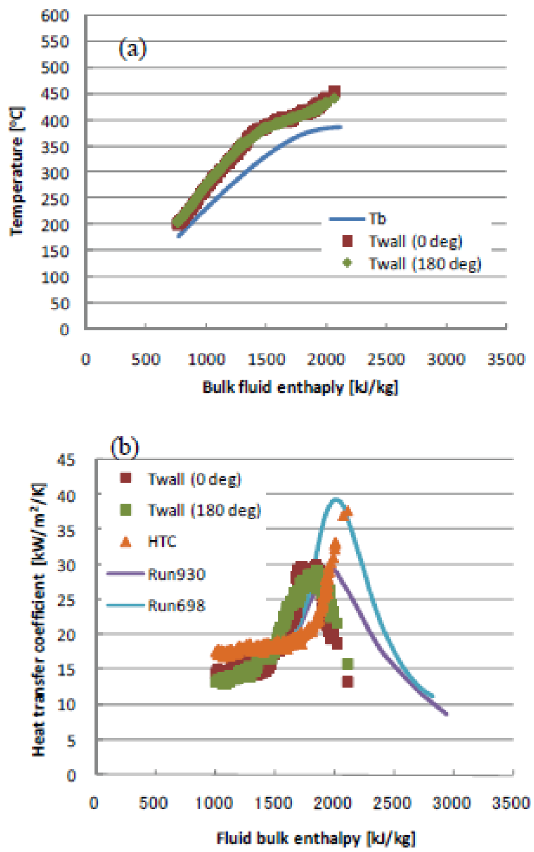

Figure 6: Typical test results for power=40 kW, G=2060-2480 kg/m^2/s P=25.5 MPa

V. KOREAN CONTRIBUTION TO THE GIF THERMAL-HYDRAULICS AND SAFETY PROJECT

V.A. Heat Transfer test for CO_2

Figure 7 shows a schematic diagram of the test facility. The design pressure and temperature of the main loop are 12.0 MPa and 80°C, respectively. Figure 8 shows the test sections and the locations of the measuring points. The test section at the left is a circular tube with an inside diameter of 4.4 mm and heated by a direct current power supply to impose a uniform heat flux on the tube internal surface. The middle one is a 9 mm tube test section.

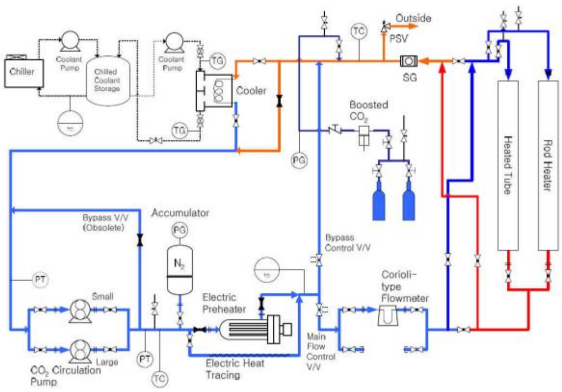

Figure 7: Schematic of test facility SPHINX

The details are the same as those for the 4.4 mm tube and 6.32 mm tube, except for the heated length. The tube of 6.32 mm ID corresponds to the subchannel hydraulic diameter of the core design by KAERI.[24] The right one is the test section for an annular channel. A heater rod with an outside diameter (OD) of 8 mm is centered in the 10 mm ID tube. The hydraulic diameter of this annulus channel is the same as the 4.4 mm tube.

The supercritical CO_2 flows upward. The fluid temperatures are measured in the mixing chambers at the inlet and the outlet of the test section as well as along the tube surface. An eccentric annular subchannel of 9.5 x 12.5 mm (1 and 2 mm gaps for the narrow and wide side, respectively) was tested also.[25]

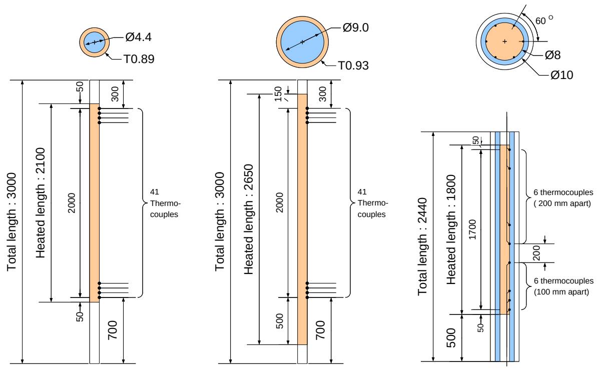

Figure 8: Test sections (tube and annulus channel)

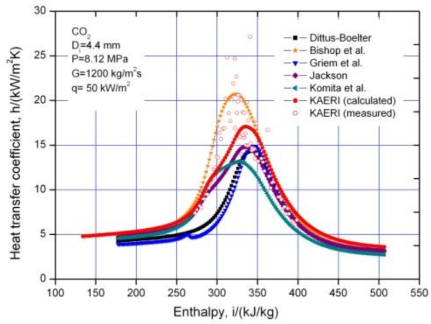

Figure 9: Comparison of the estimated heat transfer coefficient by various correlations against the experimental data.

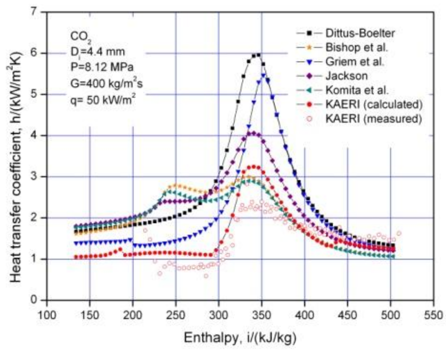

Figure 10: Comparison of the estimated heat transfer coefficient by various correlations against the experimental data.

Figure 9 and Figure **10** show typical results of the research on a heat transfer at supercritical pressure at KAERI. In the Figures the predictions by various correlations including the correlation, which was proposed based on the test data obtained at KAERI, are shown. At a condition of p=8.12 MPa, q=50 kW/m^2 in the tubes of 4.4, mm ID the proposed correlation predicts the experimental data reasonably well when compared with the existing correlations for both the normal and impaired conditions.

Table **1** summarizes the geometries and dimensions of the test sections used in the experiments at SPHINX in KAERI. The test results have been published elsewhere.[24-31]

In 2009, downward flow tests for the annulus channels will be performed.

Geometry and dimension			Up flow	Down flow
Tube	4.4 mm		●	●
	6.32 mm	Plain	●	●
		Wire	●	●
	9.0 mm		●	●
Concentric annulus (8 x 10 mm)			●	2009
Eccentric annulus (9.5 x 12.5 mm)			●	2009
Pressure transient			2009	2009

Table 1: Tested or planned geometries and dimensions at SPHINX in KAERI

V. B. Adaptation of Safety Analysis Code

Based on TASS/SMR[32], a computer code, TASS/SCWR has been developed for the safety analysis of a SCWR. For the modelling of a reactor coolant system, five one-dimensional conservation equations of a two-phase flow are formulated,[33] where the thermodynamic properties are calculated by using the IAPWS-IF97[34] formulation. The fission power input to the fuel is obtained from the reactor kinetics equations with six delayed neutron groups. An ANS73 decay heat curve has been incorporated into the database. Heat transfer correlations and conduction equations are modelled for the calculation of a heat generation in a core and the heat removal in a passive residual heat removal system (PRHRS).

A natural circulation condition has been simulated by using the TASS/SCWR and MARS[35] codes, and the results were compared with each other. Initial pressure and temperature were 25 MPa and 300°C. 10 MW was supplied to node 2 and removed from node 12. The fluid density and temperature calculated by TASS/SCWR and MARS, as shown in Figure 11 were in good agreement with each other.

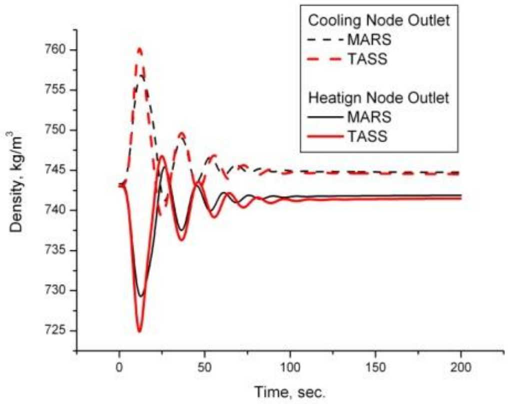

Figure 11: Comparison of the density between TASS/SCWR and MARS.

IV. CONCLUSION

In this paper the ongoing research activities as well as recent accomplishment in the TH&S PMB member countries are provided. Although the official endorsement of PA is still under processing, relatively active research is being performed in the member countries as well as in China (not introduced here since its participation status is currently an observer). The degree of research depth in the member countries is different from each other, and it will be relieved by continuing negotiation and further collaboration process.

A number of research and development (R&D) projects have been established to provide thermal-hydraulics and safety-related informa–tion in support of the development of the CANDU® supercritical water-cooled reactor (SCWR) in Canada. Thermal-hydraulics-related projects in these programs focus on the development of heat transfer prediction methods for CANDU-type bundles. These projects cover analytical and experimental studies in supercritical water and surrogate fluids in tubes and bundles.

Two R&D projects on the pressure-vessel type SCWR with fast/thermal options are ongoing in Japan jointly by universities, research industries and industries.

In Europe, the High Performance Light Water Reactor (HPLWR) is currently under development. The High Performance Light Water Reactor is a Light Water Reactor (LWR) with supercritical water at 25 MPa as coolant and moderator. Computational fluid dynamics, analytical work, and coupled neutronic-thermal hydraulic analyses have been executed to assess the heat transfer rate and fuel temperature in the core of the HPLWR. A safety concept for the HPLWR has been proposed, which now needs to be worked out by means of the analyses of transient and accident scenarios. For this purpose several thermal-hydraulic system codes have been upgraded and tested for super critical water conditions.

Korea's 3-year project comes closer to the end as of February 2010. The work scope has been limited to the supercritical heat transfer to CO_2 due to the lack of resources. The continuing support of the government for the fundamental research on SCWR is expected at least in the fields of thermal-hydraulics and safety, and material and chemistry.

Acknowledgements

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Japans' contribution includes the results of "Research and Development of the Super Fast Reactor" entrusted to the University of Tokyo by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT), and the results of "Development of SCWR in GIF Collaboration (Phase-I)," funded by the Ministry of Economy, Trade and Industry (METI).

Korea's contributions are part of the results of the 3-year project (2008-2010) in the Nuclear R&D program funded by the Ministry of Education, Science and Technology.

Nomenclature

d	diameter
T	temperature
P	pressure, MPa
G	mass flux, kg/m^2s
q	heat flux, kW/m^2

Subscript

b	bulk
w	wall

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