Integral thermal-hydraulic test facilities to support the development of the large passive nuclear power plant in China

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Abstract: The first passive pressurized water reactor AP1000 and the upgraded power version CAP1400 are going to be developed and built in China. Several experimental facilities have been launched at the State Nuclear Power Technology Research and Development Center (SNPTRD) in order to assist with the design of the large passive nuclear power plant and focus on the study and the experimental research on the physical phenomena, characteristics and performance of the safety systems involved. For the Passive Core Cooling System (PXS), an integrated test facility named “ACME” (Advanced Core-cooling Mechanism Experiment) was designed and is currently under construction. With a 1/3 height scale, ACME has a 2-loop arrangement and the test reference pressure was chosen to represent the coolant saturation pressure in a LOCA. For the Passive Containment Cooling System (PCS), several separate-effect test facilities and an integral test facility are being set up: (1) separate experiments were designed for the testing of the water film flow, the water film heating, the steam condensation and heat transfer test, *etc*.; and (2) an integral test was designed for the study of integrated phenomena and physical features of the CAP1400-PCS. Also, a series of facilities specialized on the study of the performance and mechanisms of core-melt debris and features of In-Vessel Retention (IVR) are being designed. They consist of a metal layer demonstration experiment, and an External Reactor Vessel Cooling (ERVC) 3D experiment. All these test facilities are introduced in this paper.

Keyword: passive nuclear power plant; passive safety system; thermal-hydraulic test; severe accident; in-vessel retention

1 Introduction

The State Nuclear Power Technology Research and Development Center (SNPTRD) is a young subsidiary unit of China’s State Nuclear Power Technology Corporation (SNPTC). The main aim of the SNPTC is to construct the first passive pressurized water reactor AP1000 in the world, and to develop an upgraded power version CAP1400 of passive plant with independent intellectual property rights [1, 2]. In accordance with this objective, the purpose of the SNPTRD is to assist with the design and focus on the study and the experimental research on physical phenomena, and the characteristics and performance of safety systems. Close cooperation with Tsinghua University will provide sufficient educational and scientific support to the SNPTRD.

Several experiment facilities have been launched at the SNPTRD, which are mainly aimed to evaluate the thermal-hydraulic performances of engineered safety systems of the CAP1400. For the testing of the Passive Core Cooling System (PXS), an integral test facility named “ACME” (Advanced Core-cooling Mechanism Experiment) has been designed and is currently under construction.

For the Passive Containment Cooling System (PCS), several separate effect test facilities and an integral test facility are being set up. The PCS experiment program consists of two parts: (1) separate tests including several individual experiments for various independent physical phenomena are designed for the testing of water film flow and distribution, steam condensation and heat transfer test *etc*.; (2) an integral test is designed for the research on integrated phenomena and physical features of the CAP1400-PCS, in order to enhance the data obtained by the integral test facility for the AP600/AP1000.

For the severe accident mitigation test, a series of facilities, which are used to study the performance

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and mechanism of core-melt debris and the features of In-Vessel Retention (IVR), are being designed too. The IVR experimental program consists of (1) a metal layer demonstration experiment, and (2) an external reactor vessel cooling experiment.

In this paper, the above test facilities are introduced in detail.

2 ACME test facility

ACME is a newly designed passive core-cooling integral test facility for the CAP1400. Based on the passive safety AP1000 technology transferred from Westinghouse, the CAP1400 development plan, which was proposed by the SNPTC, is currently being designed. The CAP1400 has a 1400MWe electric power output with a 2-loop arrangement of reactor coolant system, and an enlarged passive safety system capacity to accommodate the higher core power. In order to predict the performance of the CAP1400 passive safety system and develop a database for the benchmark of safety analysis codes, the ACME test program will be conducted by the SNPTC with the incorporation of the Institute of Nuclear and New Energy Technology (INET) belonging to Tsinghua University. ACME will be built in the INET and will become the first integral thermal-hydraulic test facility for the PWR passive safety system in China.

Similar test facilities around the world include ROSA in Japan[3], SPES in Italy[4], and APEX in USA[5], which were used for the AP600 and AP1000 certifications. Based on the NRC verified scaling method, the Hierarchical, Two-Tiered Scaling Analysis (H2TS) method, a 1/3 height scale ACME has been designed through detailed scaling analysis. With the experience learned through APEX and SPES tests, the ACME possesses its own features to try to maximize the benefits of the scale down test. The integral scaling ratios are 1/3 height, 1/94 volume and 1/54.3 core power. Nearly 1000 instruments of various types are going to be installed for the acquisition of thermal-hydraulic data. The overall design philosophy tries to assure that the important thermal-hydraulic phenomena and the behavior of the prototype plant may be captured and preserved.

The ACME has the same 2-loop reactor coolant system and passive safety system configuration as the CAP1400. The ACME includes a two loop Reactor Coolant System (RCS), a PXS, an auxiliary system, and an I&C system. Each RCS loop consists of 1 Steam Generator (SG), 2 Reactor Coolant Pumps (RCPs), 2 cold legs, and 1 hot leg. A pressurizer is connected to one HL through the surge line. The PXS of ACME includes 1 Passive Residual Heat Removal Heat eXchanger (PRHR HX), 2 Core Makeup Tanks (CMTs), 2 ACCumulators (ACCs), 1 In-containment Refueling Water Storage Tank (IRWST), 2 containment sumps and a 4-stage Automatic Depressurization System (ADS). The main auxiliary systems include a normal residual heat removal system (RNS), a chemical and volume control system (CVS), a Main Feedwater System (MFS), a water purification system, a condensate return system, and a draining and venting system. These systems are needed for the establishment of the expected stable test initial condition. To simulate the RCS and PXS in a proper manner, the detailed scaling needs to be done to obtain the ACME RCS and PXS design specifications. The rest auxiliary system is only functionally simulated. Figure 1 shows a conceptual 3D design, and Figure 2 shows a conceptual layout.

The ACME is expected to simulate the integral-system response after the end of the first blowdown phase during SBLOCA (Small Break Loss of Coolant Accident). Consequently the simulation will cover the natural circulation phase, the ADS depressurization phase, the IRWST gravity injection phase, and the long term recirculation cooling phase, in each phase, the specified components in the passive safety systems will be actuated. During the AP600/AP1000 certification, the test data were obtained from three test facilities: ROSA, SPES-2 and APEX. Therefore, the test initial pressure was chosen to be 9.2MPa, which is close to the coolant saturation pressure of the secondary side of the Steam Generator in the natural circulation phase during a LOCA. This is a unique choice different from the SPES-2, APEX and ROSA. This pressure selection enables the ACME facility to simulate almost all the important events and safety system activities with the same pressure as the prototype, including CMT draining, ADS 1-3 opening, ACC injection, ADS 4 opening, IRWST injection, sump injection and long term cooling.

For the SBLOCA simulation, the location of the break can be changed according to the test matrix, which includes:

- Top or Bottom on cold leg or hot leg,
- The Pressure Balance Line (PBL) of the CMT,
- Inadvertent ADS actuations,
- Double Ended Direct Vessel Injection (DEDVI) line breaks.

The test condition can also be customized by adjusting the system configuration, so both Design Basis Accident (DBA) and Beyond DBA (BDBA) can be simulated.

Besides SBLOCA simulation, some specified non-LOCA scenarios could also be simulated, mainly related to a station black out. With careful configuration adjustment, separate effect tests or other innovative study of the performance of passive safety could be performed on the facility.

According to the current ACME plan, the stages of the program are:

- During the period from July 2010 to June 2011, the ACME facility was under conceptual and engineering design.
- During the period from July 2011 to June 2012, the ACME facility will be constructed.
- During the period from July 2012 to June 2013, a shake down and the test matrix will be performed on ACME.

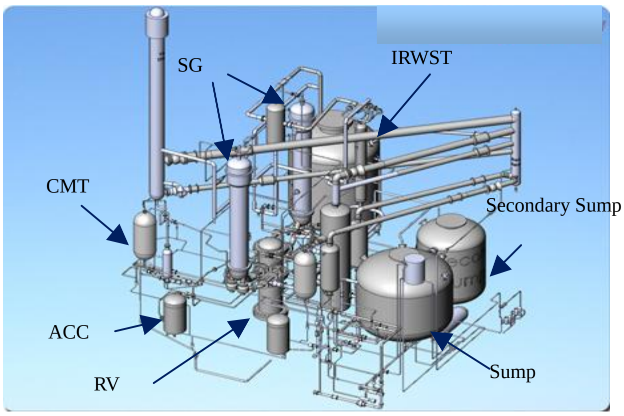

Fig. 1 A conceptual 3D design.

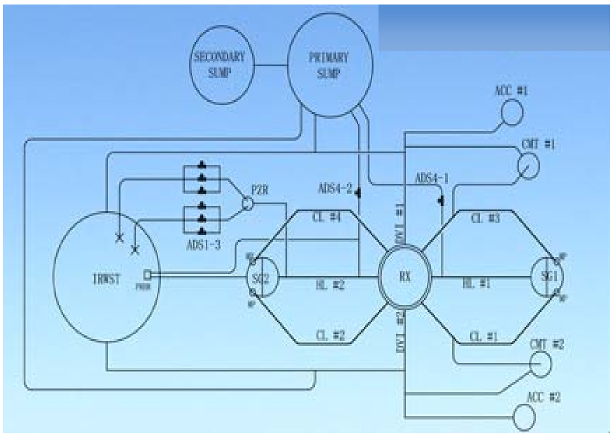

Fig. 2 A conceptual layout.

3 Passive containment cooling system test facilities

The Passive Containment Cooling System (PCS) is an important section and technical feature of the passive safety systems of the AP1000 as an advanced PWR nuclear power plant. However, because of the differences in the CAP1400 design, lots of tests designed for the AP600/AP1000 cannot take into account the operational conditions of the CAP1400. It is necessary to research and develop test facilities to face the new issues.

The tests of the PCS consist of tests of independent physical phenomena (separate effect tests) and integral physical feature tests. Separate effect tests, including a water distribution test, an in-containment steam condensation test and a coupled heating and evaporation test on water film, are designed for a specific physical phenomenon or a feature of the system performance. Integral tests are designed to enhance the understanding and study the overall properties of the system's performance.

3.1 Separate effect tests

Because of the changes in the water film flow rate and the difference of energy released from the reactor, it is necessary to study steam condensation and heat conduction in the containment, as well as water film cooling of the outer containment surface under an accident scenario. The SNPTRD is preparing engineering validation tests on the CAP1400 PCS

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3.1.1 Water distribution tests

The PCS is an important part of the advanced pressurized water reactor. It is important to be able to remove heat through the water film covering the containment surface following a design basis accident. The water flow and containment size of the CAP1400 are both increased compared with the AP600/AP1000 and the structure and position of weirs are changed, which may induce more uncertainties and disturbances. In order to enhance the accuracy of the previous test facility and gain insight on the properties of water film performance specific to the dimension of the CAP1400 containment, a new water distribution test facility is under construction. Thanks to this facility, it will become possible to measure and study the coverage, thickness, and stability of the water film, as well as the influence of the weir on the film.

Similar tests were carried out previously when the AP600 was under design. These tests involved evaluating the effectiveness of water distribution, the influence of the weir on the flow, and the delay for the water film flowing from the distribution bucket to reach the spring line of the simulated containment.

New tests have three purposes. One of them is to study the relation between the percentage of water film coverage and the flow of water, and these data will be used as input for computer code. Another goal is to measure the time it takes for the water film to stabilize. Ultimately, the tests will also be useful to validate and optimize the design of weirs.

Water distribution tests will be conducted using a 1/8th sector model of the full scale CAP1400 containment dome. The model will be designed and constructed to represent major design features, dimensional characteristics, including appropriate weld and surface variations, and surface treatments which could influence the effective, reproducible distribution of cooling water over the surface of the containment dome. Measurements of the water distribution will be obtained by collecting the flow over defined areas and by selected measurements of the film thickness using capacitance probes. Tests will be conducted over a range of water flow rates that may lead to the flow branch being anticipated. Figure 3 shows the 3D framework structure of the facility. The width of the facility is nearly 23m, while height is 14m.

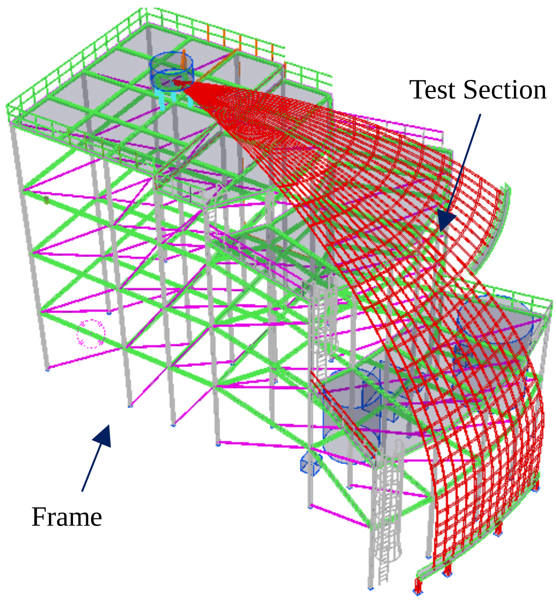

Fig. 3 Framework structure of the water distribution test facility.

3.1.2 Steam condensation tests

Steam condensation on the cold containment surface is one of the dominant heat transfer mechanisms for the PCS. The facility aims to study the steam condensation phenomenon in the containment vessel and obtain experimental data at various thermal-hydraulic conditions. A previous study has already been done on this topic[6].

The steam condensation test program includes the plate model tests and two-dimensional model tests. The purpose of the plate model test is to provide data on condensation mass transfer in the presence of a non-condensable gas at various inclination angles, velocities and steam/air/He concentrations. The test section of the plate model is a 2-meter long rectangular box with an inner channel approximately 25cm wide. Gas mixture passes through the channel, steam condensates on part of the upper surface below the maximum pressure 0.6MPa. When steady state conditions are achieved, measured data are compared with the current correlations of limited pressure and temperature.

The two-dimensional model tests are conducted to provide data on condensation mass transfer in conditions under which free convection dominates. The test section of the two-dimensional model featuring a 1:15 geometric scaling of the containment is designed to withstand the prototypical pressure. Concentration of helium in the non-condensable gases is set prior to the injection of steam and is kept constant. Steam is injected from the bottom of the test section and condensates on the top and lateral surfaces. When steady state conditions are achieved, data are measured and compared with the correlations calculating results.

Besides, a one-dimensional model is also under design and construction, in order to supply research capabilities on detailed phenomena. Dimensionless numbers, heat and mass transfer coefficients are calculated by both measurement data and correlations. By comparing the measurement data with the heat and mass transfer correlations, the use of the correlations can be validated and bias factors can be defined. Figure 4 shows the conceptual design of the test section of the steam condensation facility.

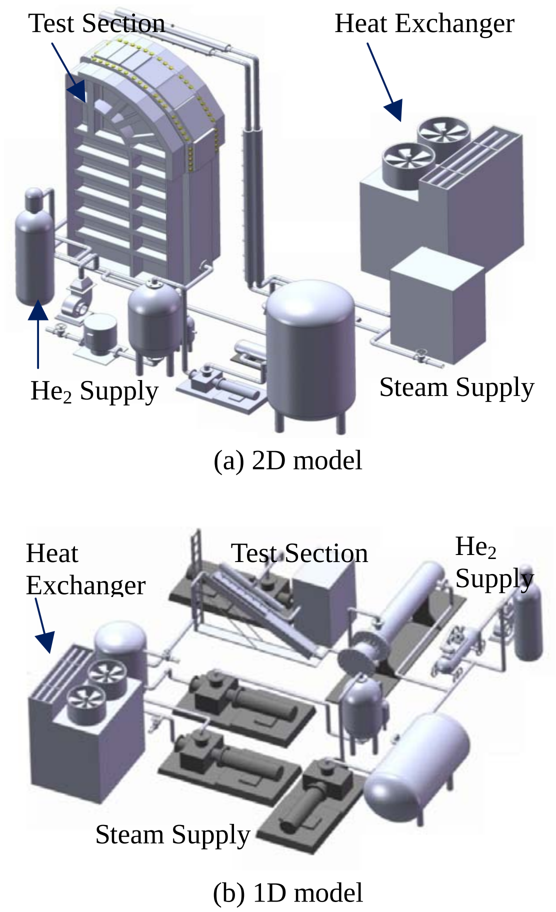

Fig. 4 Conceptual design of the test section of the steam condensation test facility.

3.1.3 Coupled heat transfer tests

The goal of the coupled heat transfer tests is to validate the heat removal mechanisms happening through steam condensation inside of the containment and the flow of the water film outside of it. It is meant to study the effects on heat transfer of the inner vessel pressure, the fraction of non-condensable gas, the velocity and temperature of the steam, the temperature and velocity of the water film outside, the temperature, velocity, and humidity of the air in the annulus of the PCS. The tests are similar to the small scale tests (SSTs) performed at the Westinghouse Science & Technology Center. The range of hydraulic parameters will cover those considered in the SSTs. Since the CAP1400 has a higher core power and a larger containment volume than the AP1000, the containment pressure, water film velocity and steam supply in the tests would increase. The test facility consists of a test vessel, pipes, a boiler, and pumps. Figure 5 shows the conceptual design details of the test section of the facility.

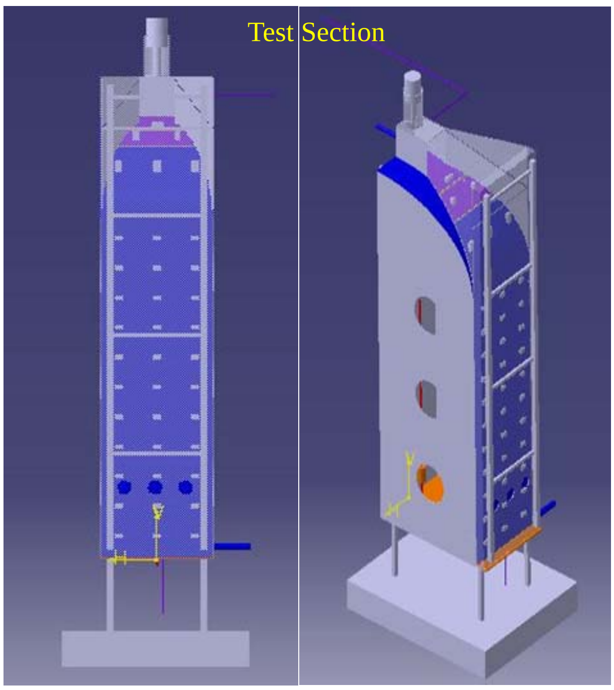

Fig. 5 Conceptual design of the test section of the coupled heat transfer test facility.

3.2 Integral physical feature tests (large-scale tests)

Since the tests above can only examine individual effects, a comprehensive large scale integral test facility is necessary for the study of integrated phenomena or physical features of the CAP1400-PCS in order to revise the data gathered by the integral test facility for the AP600/AP1000, as well as to guide the design of the CAP1400 PCS.

The PCS large scale test (LST) facility for the CAP1400 is designed to characterize the heat removal capabilities of the CAP1400 containment design. This experiment is designed to produce similar containment dome heat transfer processes and circulation/stratification patterns to what would happen inside the containment of the CAP1400. The test facility's height is 9.7m with an inner diameter of 5.37m. The operating pressure is under the maximum pressure of 1MPa. The flow structure is similar to the one used for the AP600, which was designed and constructed by Westinghouse Science & Technology Center.

The purpose of the tests is to examine the anticipated thermal-hydraulic phenomena on a large scale: interior natural convection and steam condensation, exterior water film evaporation, heat removal through air cooling, and behavior of the water film.

The test data can be used to validate the computational analysis code of the CAP1400 containment, and even guide the design of the CAP1400-PCS. Figure 6 shows the conceptual design of the test vessel of the LST facility.

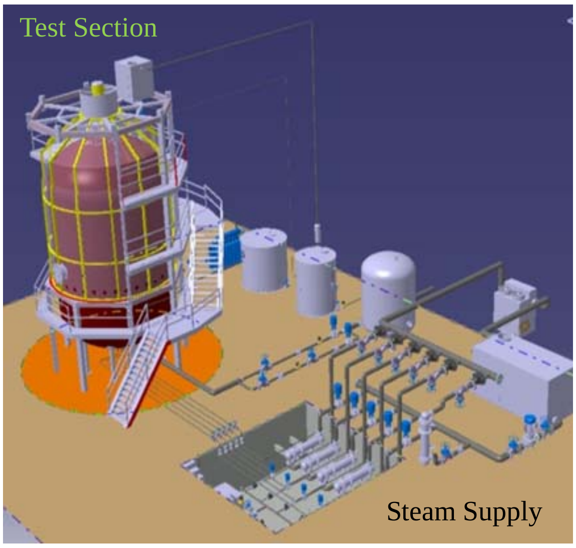

Fig. 6 Conceptual design of the test vessel of the LST facility.

4 IVR test facility

If insufficient cooling of the nuclear reactor core occurs during a postulated severe accident, the core will uncover, overheat, melt, collapse, and eventually relocate to the lower head of the reactor vessel. IVR is one of the severe accident management strategies that has been adopted by operating nuclear power plants and advanced light water reactors. One valid means to achieve IVR is through the implementation of external reactor vessel cooling (ERVC). By flooding the reactor cavity, the reactor vessel would be submerged in water to induce nucleate boiling on the outer surface of the vessel as the wall temperature rises above the boiling point of water. With nucleate boiling taking place on the outer surface of the vessel, the decay heat transferred from the core melt to the vessel wall can be removed effectively to ensure that the vessel head will retain its integrity. It was known that the vessel would fail if critical heat flux of the external surface was less than the local heat flux at the vessel wall. Therefore, quantifying the heat flux of the vessel wall and the critical heat flux (CHF) limit for the boiling process on the wall is the most important issue for the analysis of the IVR strategy. To achieve this goal, two experiments will be implemented. One of them is the metal layer demonstration experiment, and the other is the ERVC 3D experiment.

4.1 Metal layer demonstration experiment

The metal layer demonstration experiment facility is built to verify the correlations used to calculate the heat transfer characteristics of the metal layer in the CAP1400. The Globe-Dropkin correlation is widely used to calculate the top heat flux in the metal layer. A previous study has been done [7], but the condition in the CAP1400 exceeds the valid range of this correlation. The parameter range of the Globe-Dropkin correlation is more than two orders of magnitude smaller than the actual condition in the CAP1400. Therefore, the validity of the correlation needs to be verified. The purpose of the Metal Layer Demonstration Experiment is to validate the correlation under the conditions present in the case of the CAP1400, namely a high Ra number.

The physical phenomenon taking place in the metal layer is a natural convection happening when a fluid layer is heated from below and cooled on the top. The Metal layer demonstration experiment will simulate this natural convection in the metal layer of the CAP1400. In the experiment, the Ra number is the most important parameter and how to get a high value for it is the most difficult problem.

The experimental facility also includes among others a test section, a recirculating cooling water system, an electricity supply, measuring instruments, a control console. The test section consists of a column vessel with a diameter of 1m and three different heights. The bottom of the test section is heated by an electrical heater strip; the top is cooled to a desired temperature and the side is thermally insulated. The heating plate is made from copper and the heater strip is set in the copper plate. The cooling plate is made from copper with internal channels; the recirculating cooling water flows through the channels to cool the plate to the desired temperature. Platinum resistors

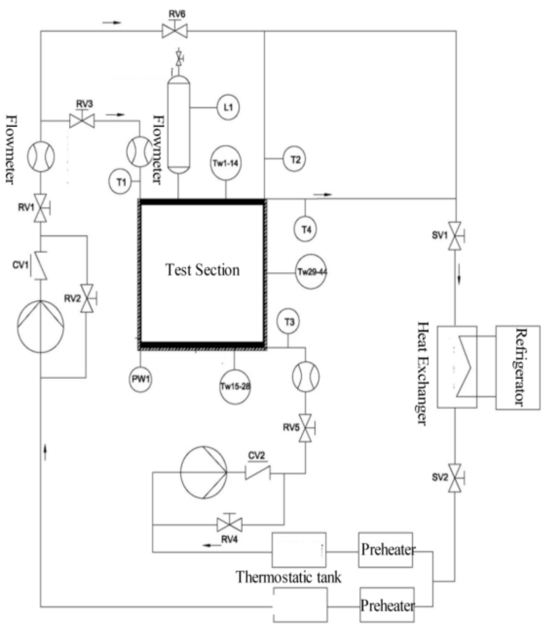

Fig. 7 The flow diagram of the metal layer test facility.

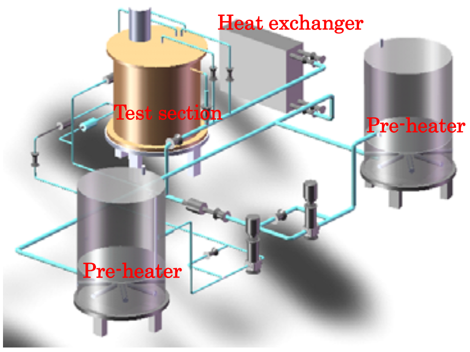

Fig. 8 3D Schematic diagram of the metal layer experiment facility.

are used to measure the surface temperature of the heating and cooling plates, and thirteen platinum resistors are present on the surface of each plate. The highest Ra number reachable in the test is 10^{12}. Therefore, the experiment can satisfy the requirement dictated by the actual conditions of the CAP1400. The height of the facility is 2.5m and the facility's heater power is about 40kW.

A series of experiments will run with about twenty different experimental conditions. With these runs, the performance of the Globe-Dropkin correlation with high Ra numbers will be validated and a more appropriate correlation for the CAP1400 will be obtained. Figures 7 and 8 show the flow diagram and the 3D schematic diagram of the test facility.

4.2 ERVC 3D experiment

The ERVC 3D experiment was developed to investigate the phenomena of natural convection boiling and critical heat flux on the outer surface of a scaled reactor vessel. This research has been done previously on the AP600 [7].

4.2.1 Scaling analysis

Scaling analysis is performed to support the design and construction of a scaled test vessel including the thermal insulation structure to simulate natural convection boiling and critical heat flux on the outer surface of a reactor vessel under severe accident conditions. The analysis considers all the key transfer processes taking place in the system. There are five transfer processes that take place simultaneously in the reactor's vessel/thermal insulation system. These are (i) the nucleate boiling process on the outer surface of the vessel, (ii) the heat conduction process in the vessel wall, (iii) the momentum transfer process in the two-phase boundary layer, (iv) the steam venting process through the gap, and (v) the water ingression process through the bottom. The characteristic time for each of these transfer processes is determined by performing a scaling analysis. By comparing the characteristic time for a given transfer process with the residence time for the two-phase boundary layer flow in the annular channel, the relative importance of the transfer process can be assessed. The dimensionless parameters controlling the key transfer processes that need to be preserved in the experimental simulation can then be determined from the results of the scaling analysis, and a number of criteria that need to be satisfied in order to preserve the time scale ratios for the key transfer processes can be derived eventually. At the end, a 1:4 scaled model will be developed utilizing the results of the scaling analysis.

4.2.2 ERVC 3D test apparatus

The ERVC 3D test facility was developed to investigate the phenomena of natural convection boiling and critical heat flux on the outer surface of a simulated CAP1400-like reactor vessel surrounded by a thermal insulation structure. The objectives are: (i) to identify the influence of the induced flow on the natural convection boiling process and of the local critical heat flux on the vessel's outer surface, (ii) to determine the effects of the inhomogeneous circumferential heat flux on the local CHF, and (iii) to study the influence of adding tri-sodium phosphate (Na_3PO_4) and boric-acid (H_3BO_3) on the CHF of the vessel's outer surface.

The test facility consists of a water tank with a condenser assembly, a heated hemispherical vessel surrounded by a scaled streamline insulator structure, a data acquisition system, a photographic system, and a power control system. Figures 9 and 10 show the flow diagram and schematic conceptual design of the ERVC-3D test facility. The total power of the facility is about 3MW.

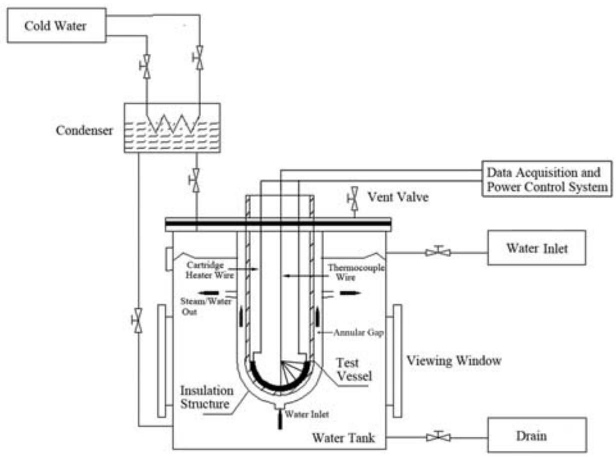

Fig. 9 Flow diagram of the ERVC-3D test facility.

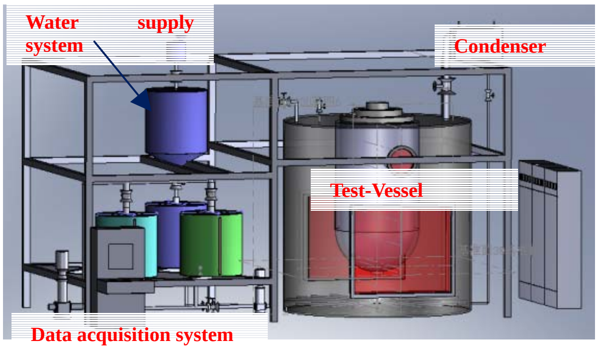

Fig. 10 3D schematic diagram of the ERVC-3D experiment facility.

5 Conclusions

These experiments or tests are mainly focused on the research around the CAP1400, but they are also aimed at advanced PWRs with much higher power-rating. The facilities launched in the SNPTRD are summarized in Table 1.

Once the tests proceed successfully as planned, the research and development ability of SNPTRD will be improved to a brand new level.

Table 1 Tests facilities constructed under the SNPTRD

Tests	Aiming Sys.	Functions
ACME	PXS	To simulate PXS performance under LOCA or MSLB accident scenarios.
Steam condensation tests	PCS	To verify the correlations of steam condensed in the containment.
Water distribution tests		To verify the performance of the water film flowing on the containment dome.
Coupled heat transfer tests		To simulate the synthetic function of the steam condensed inside of steel containment vessel and the water film cooling outside of it.

Large-scale Tests		To simulate the integral performance of the PCS for specific steam injection flow rate.
Metal Layer tests	IVR	To verify the correlations used to calculate the heat transfer characteristic of the metal layer in core-melt debris.
ERVC 3D tests		To simulate the phenomena of natural convection and CHF of the outer reactor vessel.

Nomenclature

ACC	Accumulator
ACME	Advanced Core-cooling Mechanism Experiment
ADS	Automatic Depressurization System
AP1000	Advanced Passive 1000 PWR
BDBA	Beyond Design Basis Accident
CAP1400	Chinese AP1000
CHF	Critical Heat Flux
CMT	Core Makeup Tank
CVS	Chemical and Volume control System
DBA	Design Basis Accident
DEDVI	Double Ended Direct Vessel Injection
ERVC	External Reactor Vessel Cooling
H2TS	Hierarchical, Two-Tiered Scaling Analysis
INET	Institute of Nuclear and New Energy Technology
IRWST	In-containment Refueling Water Storage Tank
IVR	In-Vessel Retention
MFS	Main Feedwater System
PBL	Pressure Balance Line
PCS	Passive Containment Cooling System
PRHR HX	Passive Residual Heat Removal Heat Exchanger
PXS	Passive Core Cooling System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RNS	Residual heat removal System
SBLOCA	Small Break Loss of Coolant Accident
SG	Steam Generator
SNPTRD	State Nuclear Power Technology Research and Development Center
SNPTC	State Nuclear Power Technology Corporation

References

[1] NDRC: China Medium and Long Term Energy Development Plan Outlines, State Council, 2004:2004-2020.

[2] LIN, C. G.: Advanced Passive Pressurized Water Reactor Technology, Beijing, Atomic Energy Press, 2010.

[3] NAKAMURA, N., WATANABE, T., TAKEDA, T., MARUYAMA, Y., and SUZUKI, M.: Overview of Recent Efforts Through ROSA/LSTF Experiments, Nuclear Eng. and Tech., 2009, 41(6):753-764.

[4] FRIEND, M. T., WRIGHT, R. F., HUNDAL, R., HOCHREITER, L., and OGRINS, M.: Simulated AP600 Response to Small-Break Loss-of-coolant-accident and Non-loss-of-coolant Accident Events: Analysis of SPES-2 Integral Test Results, Nuclear Technology, 1998:19-42.

[5] REYES, J. N. Jr, and HOCHREITER, L.: Scaling Analysis for the OSU AP600 Test Facility (APEX), Nuclear Eng. and Design, 1998, 186:53-109.

[6] ANDERSON, M. H.: Steam Condensation on Cold Walls of Advanced PWR Containments. Steam Condensation on Cold Walls of Advanced PWR Containments, 1998.

[7] THEOFANOUS, T. G., LIU, C., S., ANGELINI, S., KYMÄLÄINEN, O., and SALMASSI, T.: In-vessel Coolability and etention of a Core Melt, DOE 10460.