R&D on Nuclear Safety and Severe Accident Mitigation in China

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Abstract

Recently China issued an ambitious program of mid-term nuclear power development. It is expected that the total nuclear power installation will be around 200GW by 2030 and around 400GW by the middle of this century. Nuclear safety has been well recognized having the top priority in the nuclear power development, especially after Fukushima accident.

Nuclear safety and severe accident research activities have been growing rapidly in recent years. Numerous projects financed by the Chinese central government, local government and nuclear industries were launched. In the frame of the "National Large Scale Project", many projects were initiated and research infrastructure was constructed. More recently the National Energy Administration approved research projects in direct connection with the Fukushima accident. According to the newly issued Nuclear Safety Plan a much larger scale of infrastructure and community will be established for nuclear safety research.

The nuclear safety research community is expanding strongly in recent years. It covers nearly nuclear industries, research centers and universities. To some degree the R&D activities in China are well coordinated. There is tight collaboration and interaction among this community. This paper gives an overview about Chinese nuclear power technology development and the R&D activities in nuclear safety and severe accident mitigation. Two R&D projects ongoing at Shanghai Jiao Tong University are selected as examples, to outline some features of nuclear safety research in China.

Keywords: R&D, Nuclear Ssafety, Severe Accident, China

1. Introduction

Nuclear power has been accepted worldwide as one of the key energy sources due to its advantages with respect to environment protection, economic competitiveness and power supply stability. Since the start of the economic reform in 1970s, the Chinese economics has been undergoing rapid development. One of the bottlenecking issues in the Chinese economics development is sustainable and environment friendly energy supply. By 2020 the electricity demand in China is expected to become double as compared to 2010 (WNA 2012). For the time being, more than 80% of electricity production comes from fossil fuel. China recently overtook the USA as the world's largest contributor to carbon dioxide emissions (U.S. DOE 2012). It is predicted that China's share in global coal-related emissions will grow by 2.7% per year and reach 9.3 billion tones in 2030. Economic lost due to pollution is estimated up to 6% of GDP (World bank 2007).

Development of environment friendly energy supplies becomes thus a crucial issue in the future Chinese economy. Due to the well known limitation in renewable energy and hydro-power, nuclear power is

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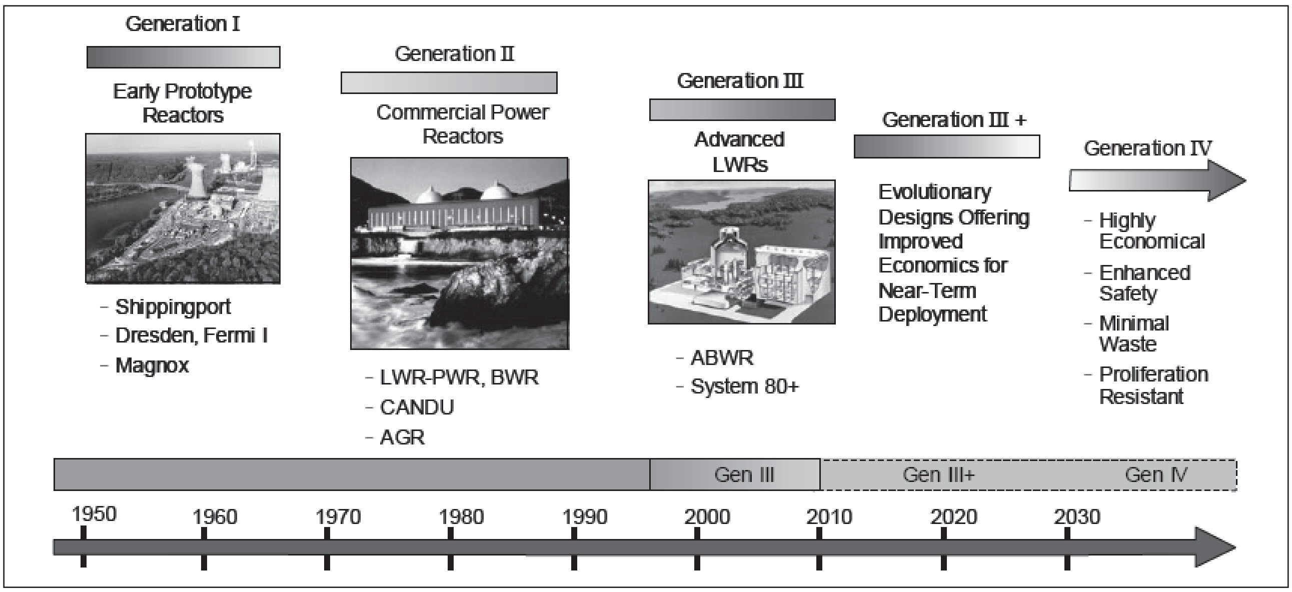
considered as a safe, clean, sustainable and economic energy source. In November 2007, China issued an ambitious program of mid-term nuclear power development (NDRC, 2007). A new target was issued most recently (Xinhua News Agency, 2012a). It is expected that the total nuclear power installation will reach 88 GW with 58 GW in operation and 30 GW under construction by 2020 (China Energy News, 2012). According to the estimation of the Chinese nuclear experts, the nuclear power installation will be around 200GW by 2030 and around 400GW by the middle of this century (WNA, 2012). That will make about 25% of the total electricity production at that time.

From the technology point of view, four generations of nuclear systems are identified by the international nuclear community (U.S. DOE, 2002), as illustrated in Figure 1.

The first generation (GEN-I) was the products of 1950's and early 1960's. They consist mainly of demonstration plants of small power. Based on the experience gathered in the first generation and applying standardization the second generation (GEN-II) was established. The most nuclear power plants operating nowadays worldwide, or all nuclear power plants operating in China, belong to the second generation. After the accident of Three Mile Island (TMI) and Tschernobyl intensive efforts were made to improve the safety features of the second generation nuclear power plants. Compared to the second generation, the third generation (GEN-III) of nuclear power plants owns a much higher safety level. The core melt frequency is lower than 10-5 per reactor and per year, and severe accident mitigation measures and related guidelines are integrated into the GEN-III systems. Various types of GEN-III light water reactors (LWR) are now available. Two of the most representative types of pressurized water reactors (PWR) are AP1000 of Westinghouse (Cummins et al., 2003) and EPR of AVERA (Czech et al., 2004), both are now under construction in China. One of the common features of the GEN-III reactors is their enhanced safety performance. This is achieved using different approaches, from the improvement of human reliability to the introduction of completely new subsystems. Passive safety systems are widely applied, especially in the AP1000 concept of Westinghouse. In China, passive safety systems are recommended to be applied wherever it is feasible. In the last three decades, extensive R&D activities were carried out worldwide to enhance safety of NPPs and



Source : U.S. DOE (2002)

Figure 1 Nuclear power technology development to develop severe accident mitigation systems. Since the beginning of this century, the priority of safety and severe accident research has been set back in favor of other features such as sustainability and economics, and in favor of the development of innovative nuclear systems of the fourth generation (GEN-IV).

The Fukushíma accident rekindles the interest of international nuclear community in safety and severe accident R&D. The importance of and the needs for strongly enhanced synergizing international research activities are well recognized to further improve nuclear safety culture and to stimulate safety research (Yamada, 2012). The ambitious program of nuclear power development in China requires high safety level, motivates R&D in safety and severe accident mitigation and attracts interests and attentions of international community for stronger interaction and collaboration with Chinese nuclear community.

2. Chinese Nuclear Power Technology

Based on the experience gathered worldwide in the nuclear power development of the past 50 years, attention has be paid in China to the selection of reference technology lines and to the realization of self-reliance technology, to ensure a safe, economic and sustainable development of nuclear power.

2.1 Selection of Technology Lines

For the time being, 15 units have been put into operation with a total installed capacity of 12.3 GW and 26 units are under construction with an installed capacity of 31.0 GW (Wang, 2012). Although all these units consist of water-cooled reactors, they are from different technology lines of GEN-II and GEN-III, e.g. Chinese small power PWR, Canadian CANDU, French 900 MW class PWR and its modified CPR1000, Russian WWER, AP1000 of Westinghouse and EPR of AREVA. The existing experience emphasizes the necessity to reduce the number of technology lines and to define a few major technology lines for the future Chinese nuclear power plants. Considering the Chinese specific situation and the experience gathered in the national and international nuclear community, it is well agreed and decided that pressurized water reactor of GEN-III will be the main reactor type for the future Chinese nuclear power generation, at least for the mid-term. Passive safety systems should be a key feature of the Chinese GEN-III PWR and need to be implemented wherever it is feasible. In addition, it should fulfill the following requirements (Ouyang, 2008):

a) Economical competitiveness;
b) Operating reliability and easy maintainability;
c) Compliment with the latest safety codes for severe accident prevention and mitigation measures issued by China National Nuclear Safety Administration (NNSA) and IAEA;
d) Digital instrumentation and control system;
e) Advanced human factor engineering technique and advanced main control room.

The above technology requirements justify the choice of AP1000 technology of Westinghouse as one of the reference technology line for the Chinese GEN-III PWR.

2.2 Process to Self-reliance Technology

As soon as the future technology lines are defined, extensive efforts should be made to develop self-reliance technology. To achieve the mid-term target, China issues twofold strategy (Cheng et al., 2009). In one side construction of NPPs based on existing GEN-II PWR technology will be continued. Modification of the GEN-II PWR power plants will be undertaken, especially with respect to safety performance. In the other side large efforts are made to accelerate the self-reliance process of the GEN-III PWR technology. The Chinese government has issued a National Large-Scale Pogram to develop technology of advanced large-scale pressurized water reactors (The State Council, 2006) and to accelerate the self-reliance of the Chinese nuclear technology. The nuclear power self-reliance program has been launched with Sanmen project in Zhejiang Province and Haiyang project in Shandong

Province as supporting projects (Ouyang, 2008). By 2017, a prototype reactor of the Chinese self-reliance GEN-III PWR with 1400 MW electric power (CAP1400) will be constructed and put into operation (Zheng, 2012).

Realization of self-reliance nuclear technology requires high quality coordination, including various institutions for design, research, manufacture and education. For this purpose a new organization, the State Nuclear Power Technology Corp. LTD (SNPTC), was founded in 2007. SNPTC is responsible for the self-reliance of the Chinese GEN-III PWR technology and has established sub-companies for research, design and manufacture, respectively (www.snptc.com.cn).

In the frame of the "National Large-Scale Program", comprehensive R&D activities on severe accident prevention and mitigation are proposed. They include both fundamental and engineering aspects. Some R&D studies focus directly on the design and licensing requirements of CAP1400, whereas some other activities deal with basic phenomena involved in severe accident procedures, e.g. phenomenological studies on fuel-coolant interaction, hydrogen distribution and combustion.

2.3 Actions after Fukushima Accident

The Fukushima accident gives fatal impact on and consequence for nuclear power. On the other hand, it forces the international nuclear community to learn lessons for safer and more reliable nuclear power systems. The importance was well recognized of further enhanced research in safety culture, severe accident prevention and mitigation and international collaboration (Yamada, 2012). Most of the countries with nuclear power plants undertook actions and responded to Fukushima accident immediately. Directly after Fukushma accident, the Chinese State Council made four decisions (Wang, 2012):

a) To immediately organize comprehensive safety examination (CSE) to nuclear power plants to identify weakness and take corresponding measures if necessary;
b) To improve management and to ensure safe operation by utilities and to strengthen regulation procedure by regulatory body;
c) To apply the most advanced standards to review and to examine all NPPs under construction;
d) To establish Nuclear Safety Plan (NSP). No new NPP will be approved until NSP is issued.

The main purposes of CSE are to evaluate the conformity of the exiting NPPs with the current Chinese regulations and standards, to identify the potential safety weakness of the existing NPP and to propose requirements to enhance their safety. The main issues of CSE cover the NPP capability against flooding, earthquake, fire, state blackout, emergency power losing, multiple external nature hazards and severe accidents mitigation capability. Furthermore, public information and communication, environmental monitoring system and emergency response system are also subjects of CSE. The main outcomes of CSE are:

a) The safety of the most Chinese NPPs meets the Chinese nuclear safety regulations and the safety requirements defined by IAEA nuclear safety standards;
b) Prevention and mitigation of severe accident of NPPs have been taken into consideration and managed effectively during design, manufacturing, construction, commissioning and operation in new Chinese NPPs;
c) Based on the lessons learned from the Fukushima accident, additional research works are recommended in order to further enhance the nuclear safety, especially in the fields of extreme external events, severe accident and emergency Management.

Shortly after Fukushima accident, the China National Energy Administration initiated several projects dealing with safety and severe accident mitigation aspects, also directly in connection with Fukushima accidents (NEA, 2012; WNA, 2012), such as:

- Passive emergency power supply;

- Passive cooling of refueling water pool;
- Passive containment cooling;
- Hydrogen control systems;
- Enhanced resistance against earthquake and other external disasters;
- Numerical simulation platform for severe accident;
- Robot for emergency application under severe accidents;
- Emergency treatment of radioactive water.

In October 2012, the Nuclear Safety Plan (NSP) was approved by The State Council (XinHua News Agency, 2012b; WNA, 2012), which defines tasks and targets by 2015 and 2020, respectively. One of the main goals by 2020 is to practically eliminate the possibility of large release of radioactive materials. During the first period, five key projects are proposed. One of them is related to R&D in nuclear safety and severe accident prevention and mitigation. This project consists of two parts. The first part is devoted to the development of research capability and infrastructure, whereas the second part to scientific tasks covering more than 100 individual subprojects. International collaboration is identified as one of the key measures to successfully carry out these subprojects.

3. R&D Activities

According to the ‘defense in depth’ safety concept in nuclear power plants, the main safety measures can be divided into three categories;

- Normal operating conditions. Here the focus is on the reliability of operating systems, operating personals, safety culture and management. It covers the procedure of design, construction and operation of nuclear power plants. With these measures the occurrence of abnormal and accidental events are reduced as far as possible.
- Mitigation of design based accidents and prevention of severe accident. In case of accident occurrence, safety systems are put into operation, to guarantee the shutdown of reactor and safe removal of decay heat. To achieve these goals, nuclear power plants have reactor shutdown system and core safety injection systems (CSIS).
- Mitigation of severe accident. The objectives of severe accident mitigation systems are to minimize the consequence of severe accidents, to guarantee the integrity of containment (the last barrier to confine radioactive materials) and to avoid large release of radioactive materials to environment.

In China R&D activities in all three categories are ongoing. Very strong activities are focused on core safety injection system and severe accident mitigation. Some key procedures involved in severe accident are summarized in Table 1. It starts from core melt process. The core melt collected in the lower plenum of the reactor pressure vessel (RPV) and forms a melt pool. The formation and behavior of melt pool in the lower plenum is a key process to be considered in the mitigation measures. In GEN-III LWR, the applied severe accident mitigation concepts can be divided into two classes. In one class, the mitigation system focuses on the integrity of RPV and confines 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 PWR. The AP1000 and the further developed CAP-series apply IVR concept (Zheng, 2012; WNA, 2012), whereas the VVER and EPR constructed in China use ex-vessel core catcher concept (Czech et al., 2004;, WNA, 2012). Independent of severe accident mitigation concepts, there are several common and important processes, such as hydrogen safety, steam explosion and containment cooling and integrity.

Table 1 shows the author’s personal evaluation of the processes according to the amount of previous R&D studies worldwide, knowledge at present stage and the importance for future Chinese R&D. The evaluation score ranges from 1 to 3 and stands for ‘low’ to ‘high’. The score ‘3’ means a large number of previous studies, or sufficient existing knowledge or high importance for Chinese future R&D. For all phenomena, moderate to extensive R&D activities

Table 1 Main SA processes and their status

Phenomena	Previous R&D	Existing knowledge	Importance for China
Core melt process	2	2	1
Melt pool behavior	2	1	2
In-vessel retention, IVR	2	2	3
Ex-vessel core catcher	3	2	2
Fuel coolant interaction	3	1	1
Hydrogen safety	3	2	3
Containment cooling	2	2	3

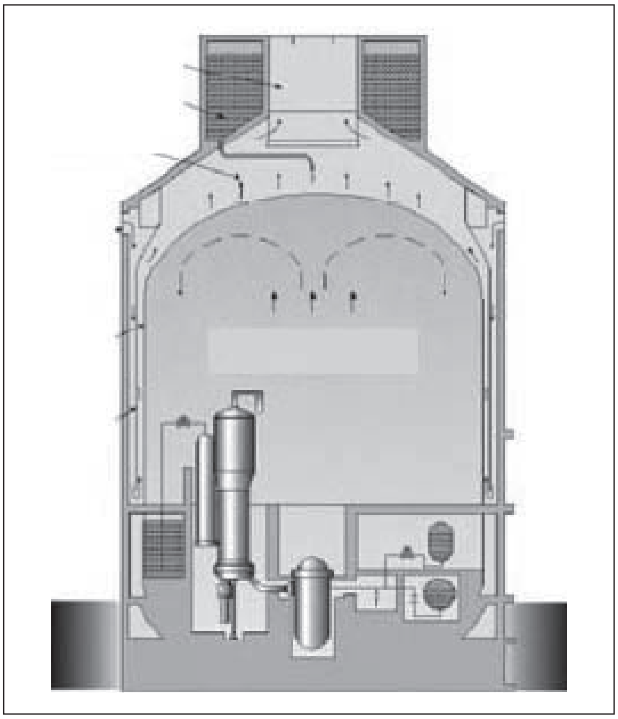
were carried out in the past worldwide. However, there exists deficiency in knowledge related to all phenomena. Considering the present situation in China, three phenomena are considered by the author as having the highest priority, i.e. IVR related phenomena, hydrogen safety and containment cooling and integrity.

Research activities on nuclear safety and severe accident mitigation are being carried out at various Chinese institutions, e.g. large scale integral test facilities for core safety injection system and containment cooling are under construction at the State Nuclear Power Research Institute of SNPTC (Chen, 2012; SNPRI, 2012), a large R&D program on IVR related phenomena has been established at China Nuclear Power Research Institute (CNPRI, 2012) of China Guangdong Nuclear Power Corporation, and some other research works are launched at Nuclear Power Institute of China (NPIC, 2012; Lu et al., 2008) of China National Nuclear Corporation related to safety performance improvement in accident prevention and mitigation. In the following, two R&D projects on severe accident mitigation at Shanghai Jiao Tong University are selected and presented in this paper.

3.1 Passive Containment Cooling

As the last safety barrier containment integrity has achieved strong attention of the Chinese nuclear community. Passive containment safety systems have been widely applied to advanced water-cooled reactors. As a long-term passive decay heat removal AP1000 of Westinghouse or Chinese CAP-series use the passive water injection flow, natural convection of air combined with thermal radiation to remove the decay heat from the containment to outside. For the short-term (the first 72 hours) water-film evaporation heat transfer plays the dominant role.

Figure 2 shows schematically the passive containment cooling system of AP1000 and the Chinese CAP series. It consists of two containment shells, i.e. the inner steel containment shell and the outer concrete containment shell. Between both containment shells there exists a baffle plate, which divided the gap into two flow channels. A water tank locates at the top of concrete containment shell. Under accident conditions, water injection from the water tank is initiated to the top of the inner steel containment shell. It flows downward along the inner containment shell, forms water film, evaporates and transfers heat. In addition, natural convection of air is produced due to buoyancy effect. In the inner flow channel with higher temperature, air mixed with evaporated steam flows upwards, whereas in the outer flow channel air enters into the channel at the top opening, flows



Source : Cummins et al. (2003)

Figure 2 AP1000 containment and its passive cooling system downwards and turns over at the channel bottom. This natural convection of air contributes also to the heat removal from the containment to environment. Water tank is sufficient for water injection of 72 hours. During these 72 hours heat transfer through water film evaporation plays dominant role. After the termination of water injection, natural air convection or other measures have to guarantee sufficient containment cooling.

Heat removal via water film evaporation is strongly affected by the water film behavior such as water film thickness, coverage rate and partial pressure of steam. Up to now water film behavior under prototypical conditions is still insufficiently understood, e.g. under count-current air flow conditions.

One of the key features of heat transfer via air convection is the strong effect of buoyancy and interaction with thermal radiation (Cheng & Mueller, 1998; Cheng et al., 2001; Cheng et al., 2001). Convective heat transfer capability could be significantly reduced under buoyancy effect and needs detailed investigation. Therefore, investigations on fundamental heat transfer mechanism and heat transfer capability are carried out with both experimental and numerical approach at Shanghai Jiao Tong University. Several test facilities have been constructed to achieve different purposes, as summarized in Table 2.

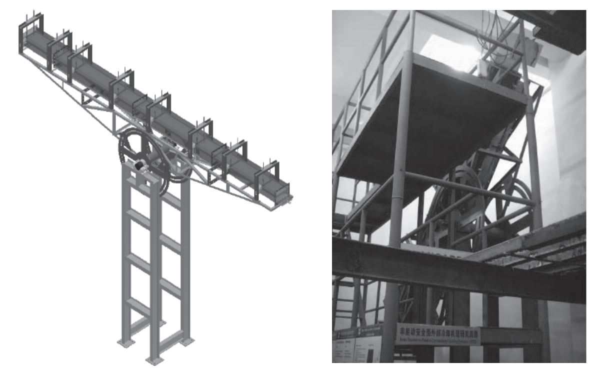
(a) Natural Air Convection

The experimental work with MICARE (MIx Convection of Air in Rectangle Channel) test facility belongs to fundamental features with the main objectives to study natural air convection behavior and heat transfer. Sophisticated measurements provide test data of three-dimensional velocity profile for improved understanding of phenomena and validation of models and simulation codes. Figure 3 shows the test section MICARE, which is a square flow channel with the cross-section up to 400 mm x 250 mm. One side of the channel is electrically heated. The heated plate is designed to achieve well uniform distribution of heated wall temperature. The orientation of the flow channel can be changed arbitrarily. The test section has a total height of 8 m. The test section can be connected to an auxiliary equipment to realize a forced flow of air into the test channel. The wall temperature can be varied up to 200°C. The test facility is equipped among others with a large number of thermocouples to measure the distribution of wall temperatures. Hot-wire anemometer and thermocouples are applied to measure the air velocity and air temperature distribution in the flow channel (Yu, 2012a; Yu et al., 2012b).

Figure 4 shows examples of experimental and numerical results related to flow behavior of natural air convection and heat transfer from the heated plate. For numerical simulation CFD program with six different turbulence models were applied. It is seen that selection of turbulence models affects strongly numerical results. Further studies are required on turbulence modeling of flows under strong buoyancy effect. The test data achieved are well suitable for validation of CFD programs.

Table 2 Test facilities at SJTU for containment cooling

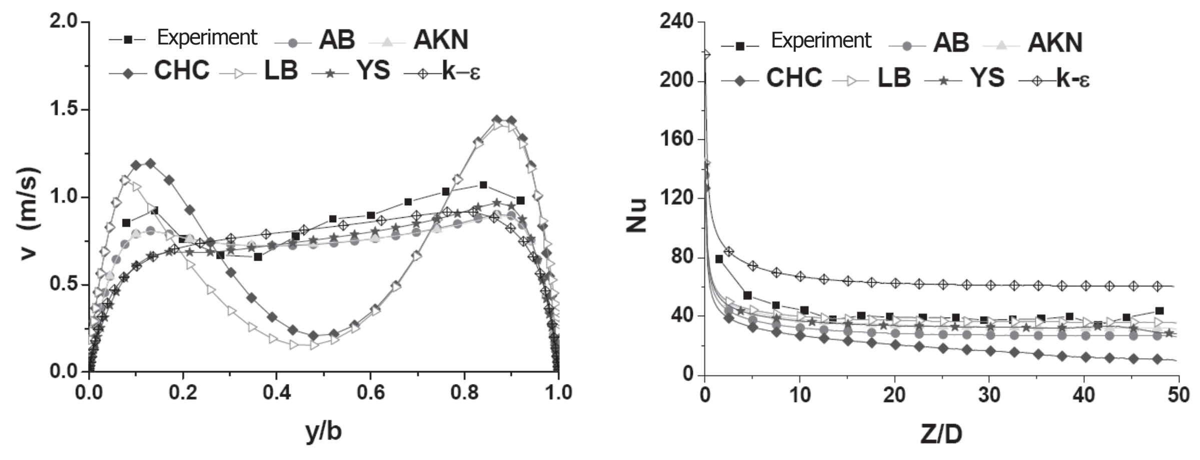
Test facility	Main purposes
MICARE	Natural air convection heat removal capability and effect of thermal radiation heat transfer
WABREC	Water film behavior under isothermal conditions (film thickness, film velocity, coverage rate); Effect of various parameters, such as count current air flow, inclination, surface properties;
WAFCORE	Water film behavior and heat transfer under heat input: Boiling inside film; Effect of air flow; Engineering confirmation tests



(A) Scheme of MICARE (b) Picture of MICARE

Source : Yu (2012a)

Figure 3 Test facility MICARE



(a) velocity (b) heat transfer

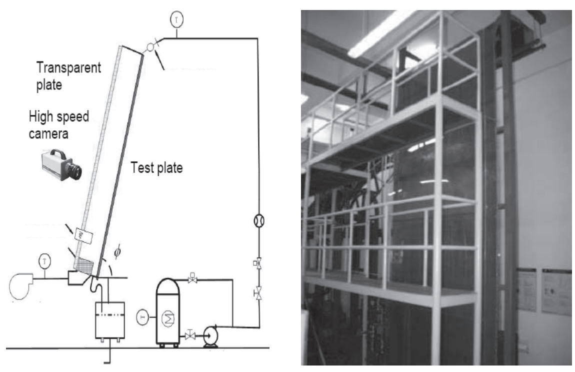
Source : Yu (2012a)

Figure 4 Comparison of experimental data with numerical results using various turbulence models. AB, AKN, CHC, LB, YS and k-ε indicate six different turbulence models:

(b) Water Film Behavior

The test facility WABREC (Water Behavior in Rectangle Channel), as schematically illustrated in Figure 5, is devoted to study water film behavior, such as water film thickness, break and dynamics. It consists of two parts. In the top it is a one fourth cylindrical surface with 2m diameter, whereas the lower part is a straight plate of 2m width and 5 m length. The upper part is required to achieve an expected water film distribution at entering into the straight plate. Plate surface is pained with Carbozinc 11 HSN, the same material as used in AP1000. Capacitance probes are used to measure local film thickness. High speed camera is applied to visualize water film surface dynamics and to measure surface wave propagation velocity and water film coverage rate. An outer channel with transparent channel wall can be added to the test section to enable count current air flow conditions.

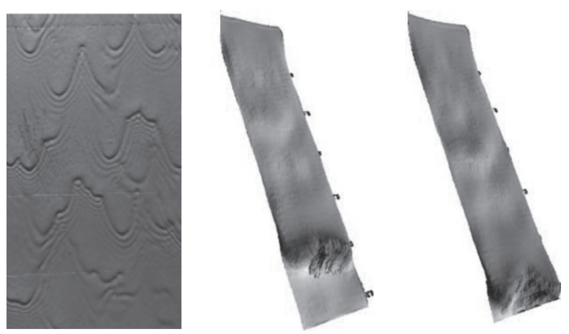
Figure 6 shows the water film surface and surface wave characteristics from both experimental observation (Figure 6a) and numerical simulation of CFD (Figure 6b & 6c). As seen, water film surface wave behavior can be reasonably simulated with CFD approach.



(a) Scheme of WAFREC (b) Photo of WAFREC

Source : Yu (2012b)

Figure 5 Test facility WABREC



(a) Experimental observation (b) CFD results at t0 (c) CFD results at t0+0.05s

Source : Yu (2012a)

Figure 6 Experimental and numerical results of film surface wave

Based on the present studies, a new correlation is proposed for the average film thickness:

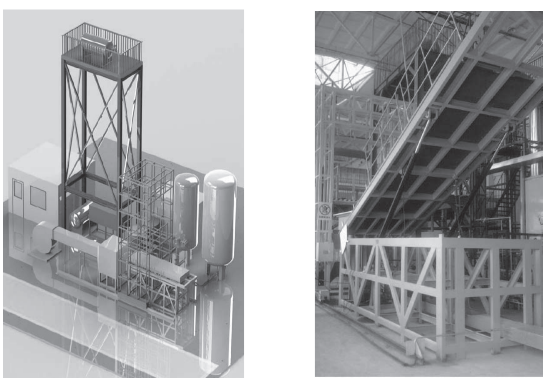
$$\bar{\delta} = 0.462\,\mathrm{Re}^{0.422}\left(\frac{\nu^2}{g\sin\phi}\right)^{1/3} \qquad (1)$$

This equation is valid for water film Reynolds number range (80 < Re < 600).

(c) Engineering Confirmation

In addition to the two fundamental research facilities MICARE and WABREC, A large scale test facility WAFCORE (Water Film Cooling in Rectangular channel) was recently constructed, to investigate water film performance under prototypical conditions.

The WAFCORE facility, as shown in Figure 7, consists of a plate made of stainless steel with a dimension of 5 m x 1.2 m. The plate surface is painted with organic zinc to maintain good wettability. The plate is heated by high temperature oil from back side. A visualization window is mounted parallel to the stainless steel plate, about 5~30 cm apart from it, to form a rectangular channel simulating the air gap existing between the containment steel vessel and the baffle in the reference design. A blower located at the bottom of the channel, circulates air flowing into the rectangular channel with velocity up to 15 m/s. Heat flux at the plate surface could be varied up to 100 kW/m2. In addition to previous fundamental studies, two additional phenomena will be investigated at the WAFCORE facility, i.e. water film behavior under heat input and boiling phenomena and heat transfer inside water film.



(a) Scheme of WAFCORE (b) Photo of WAFCORE

Source : Hu (2012)

Figure 7 Test facility WAFCORE

3.2 IVR Related Phenomena

During the transient phase of severe accident (SA) progression, integrity of reactor pressure vessel (RPV) lower head is threatened by a wide spectrum of phenomena, e.g. various melt relocation scenarios, potential steam explosion, jet impingement, etc. A limiting case and strategy in late phase of severe accident is maintenance of lower head integrity through external cooling of the RPV head to reach the in-vessel retention of molten pool (IVR-ERVC). Figure 8 shows schematically the principle of IVR and some key phenomena involved.

Under core melt accident, molten fuel and molten structural materials are collected in the lower head of RPV. Decay heat is released in the melt pool and needs to be removed via thermal radiation at the upper pool surface and conduction/convection to water at the outer surface of RPV. Inside core melt with high Rayleigh number, flow has features of strong turbulent mixed convection (Rempe et al., 2004; Rempe et al., 2005). The thermal-hydraulic behavior of the melt pool, and subsequently the heat flux at the RPV outer surface is dependent on melt pool structure, which is

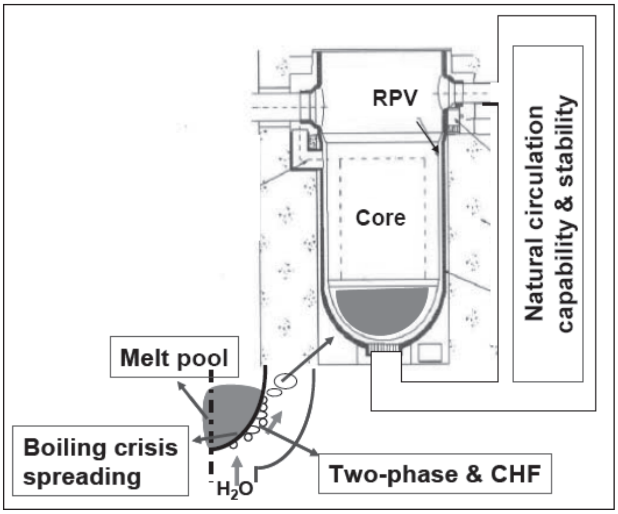


Figure 8 Scheme of IVR principle

in spite of extensive studies still not well understood. Due to high heat flux, boiling occurs at the RPV outer surface and keeps the surface temperature slightly higher than the saturating temperature of water. The upper limit of heat transfer capability at the RPV outer surface is determined by the phenomenon “boiling crisis”. In case no boiling crisis occurs, a sufficient heat removal and RPV cooling is assumed. As soon as boiling crisis occurs, the RPV outer surface temperature could sharply increase. In the nuclear community it is generally agreed that the occurrence of boiling crisis would directly lead to RPV failure. Therefore, efforts were mainly restricted to investigate critical heat flux and to measures enhancing critical heat flux. However, there is significant deficiency in understanding boiling crisis propagation and post CHF heat transfer.

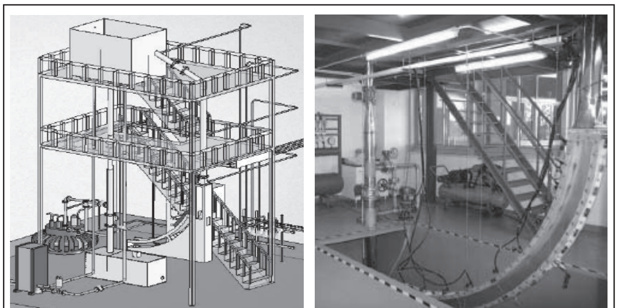
The IVR-ERVC concept was first investigated and explored for the Loviisa pressurized water reactor (PWR) in Finland. It was accepted as the major accident management measure by the Finnish regulatory agency. In the USA, the design of the advanced passive reactor AP-1000 employs ex-vessel flooding as an accident management scheme. The safety strategy of AP1000 is to keep RPV intact at any conditions, including severe accident core melt conditions. There is no core catch outside RPV. Late on, IVR-ERVC was also proposed for other PWRs and BWRs such as Korean APR-1400 (Rempe et al., 2005) and German SWR1000 (Stosic, 2008).

In China, IVR-ERVC concept has been adopted in the design of several advanced reactors, e.g. CPR1000 (Li et al., 2010). For extending the reactor power of AP1000 to higher level, the feasibility of the passive IVR-ERVC concept becomes one of the bottlenecking factors and attracts extremely strong attention of the Chinese nuclear community (Wang et al., 2012). Both experimental and theoretical studies were initiated at SJTU as early as 2008, in collaboration with Shanghai Nuclear Research and Design Institute and CGNPC. The IVR-ERVC project at SJTU consists of two parts. The first one is devoted to fundamental studies on phenomena involved in IVR-ERVC, whereas the second part concentrates on integral heat removal performance.

(a) Fundamental Phenomena

For fundamental studies, the test facility REPEC was built at SJTU and illustrated schematically in Figure 9. The REPEC test facility simulates ERVC with a full height loop and a slice geometry test section. Vapors are generated on the heating surface, mixed with water in the test section, vented through the upper tanker and released to the atmosphere. The configuration of the test section is a slice with the same radius of RPV lower head, and the width of which is 0.15m. To get a direct observation of the flow boiling phenomena, transparence glass windows are designed in the test section. The test section is heated by 27 individually controlled heating zones. The designed maximum heat flux approximates to 1.2MW/m2. The experimental study consists of two phases:

- Phase I. Cold tests: Air is injected from the vessel wall simulating vapor generation. The main purpose of the cold phase is to investigate two-phase flow characteristics such as flow regime, void fraction distribution and natural circulation capability.
- Phase II. Warm tests: Vessel wall is electrically heated and vapor is produced on the wall surface. The main purpose of the warm tests is to study heat transfer & CHF in the gap, thermal stratification and natural circulation instability.



(a) Scheme of REPEC (b) Test section of REPEC

Source : Li (2010)

Figure 9 Test facility REPEC

(b) Engineering Confirmation

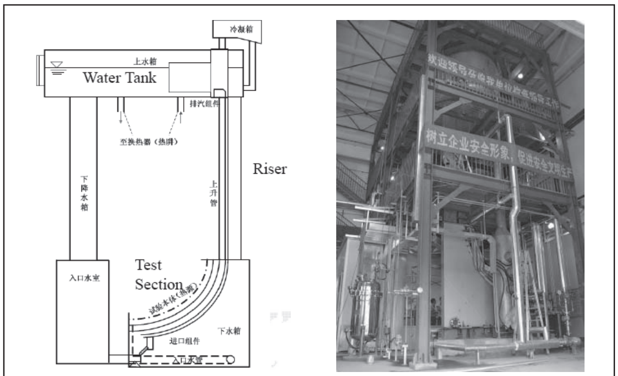
The second part of the IVR-ERVC program at SJTU is oriented on integral performance of IVR-ERVC. For this purpose the REPEC-II facility was constructed, as illustrated schematically in Figure 10.

The main technical features of the REPEC-II test facility are summarized as below:

- Full height; total height about 12 m, same as in the design of CAP1400;
- Surface scale: 1/100;
- Various heat flux distribution simulating different accident scenarios;
- Comprehensive two-phase measurement, including visualization;
- Both forced and natural circulation possible;
- Maximum electric power supply: 1.1 MW, which gives a maximum surface heat flux up to 2.6 MW/m2.

The main objectives of this project are to investigate

- Two-phase flow and heat transfer performance in ERVC;
- Integral ERVC system performance, including natural circulation capability, flow stability;
- Critical heat flux under prototypical conditions, to provide engineering data supporting the license procedure of CAP1400;
- Effect of various parameters on ERVC capability and their optimization.



(a) Scheme of REPEC-II (b) Picture of REPEC-II

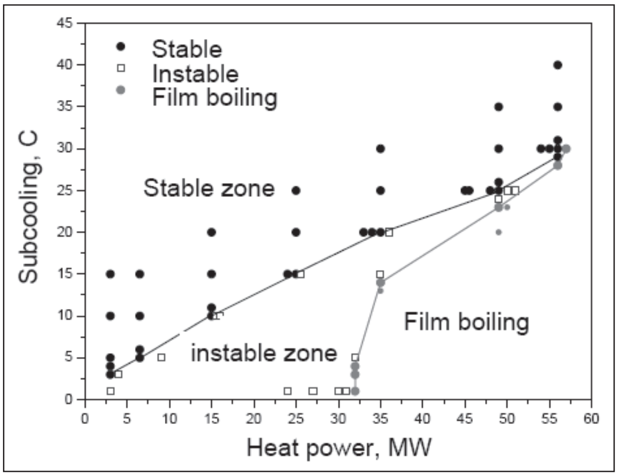
Source : Kuang (2012)

Figure 10 Test facility REPEC-II

In addition to experimental work, numerical investigation using 1-dimensional system analysis program and 3-D CFD program was carried out to study the ERVC system performance and to support the test facility design. Figure 11 shows one example of the system analysis simulation presenting the boundary of flow stability of the CAP1400 ERVC system. As recognized, lower sub-cooling of water entering the RPV cooling channel or higher heat power reduces the stability margin. For a total decay heat of 25 MW, the present ERVC system shows instable behavior, as soon as the subcooling is lower than 15°C. The numerical results require experimental validation and if necessary, the ERVC system needs modification based on both experimental and numerical investigations, to enhance its flow stability.

4. Summary

Development of environment friendly energy supplies becomes a crucial issue in the future Chinese economy. Due to the well-known limitation in renewable energy and hydro-power, nuclear power is considered as a safe, clean, sustainable, and economic energy source. The Chinese government issued an ambitious program of midterm and long-



Source : Li (2012)

Figure 11 Calculated flow stability map of ERVC system term nuclear power development. Nuclear safety was well recognized having the top priority in the nuclear power development. Considering the Chinese specific situation, experience gathered in the national and international nuclear community and the importance of passive safety systems, AP technology of Westinghouse is accepted as one of the main technology line for the Chinese GEN-III PWR.

The Fukushima accident has strong impact on the nuclear power development in China, especially related to nuclear safety and severe accident mitigation. Directly after Fukushima accident, the State Council suspended approvals for new nuclear power plants and conduct comprehensive safety checks of all nuclear projects, including those under construction. The construction work of four approved units was also stopped and many projects underwent delays. Until October 2012, after issuing the Nuclear Safety Plan, approval of new plants was restarted. According to the "12th 5-year Plan for Nuclear Safety and Radioactive Pollution Prevention and Vision for 2020", compiled by the Ministry of Environment and approved by the State Council, China will spend RMB 80 billion to improve nuclear safety of nuclear power plants at operating and under construction over the next three years.

A large number research and development (R&D) projects was launched and are proposed in recent years related to nuclear safety and severe accident mitigation. In the frame of the "National Large Scale Project", which is coordinated by the State Nuclear Power Technology Corporation, research activities were initiated, which cover various aspects of severe accident prevention and mitigation. Numerous test facilities and research infrastructure were established or under construction.

Directly after Fukushima accident, the National Energy Administration initiated and financed research projects, in direct connection with Fukushima accident. In the newly issued Nuclear Safety Plan, R&D in nuclear safety and severe accident prevention and mitigation is a main topic with two different targets. The first target is devoted to the construction of research capability and infrastructure, whereas the second target to scientific tasks covering more than 100 individual subprojects.

In addition to the projects financed by Chinese central government, many research activities are also ongoing at nuclear industries and research institutions financed by nuclear industries or local governments. All these form a large community of nuclear safety research. At the same time, this community is making big efforts in enhancing international exchange and collaboration. In April 2008, the International Workshop on Passive Safety System (IPASS08) was organized in Shanghai, and in September 2012, International Workshop on Nuclear Safety and Severe Accident (NUSSA) was held in Beijing with more than 150 participants from more than 10 countries.

In the last few years the nuclear safety research community has been growing rapidly. It consists of nuclear industries, research centers and universities. The Shanghai Jiao Tong University has been actively working on nuclear safety research since more than 10 years, engaged in various aspects such as IVR-ERVC, PCCS, fuel-coolant interaction and hydrogen safety, and becomes now one of the key institutions in this community. Many test facilities for both fundamental research and engineering confirmation established. The main features of the R&D works at SJTU are:

- In tight collaboration with nuclear industries;
- Fundamental research combined with engineering applications;
- Experimental investigation accompanied with extensive theoretical and numerical studies;
- Open for international collaboration.

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