ISSUES AND CHALLENGES ON ADVANCED THERMAL-HYDRAULIC SAFETY RESEARCH

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ABSTRACT

This paper introduces major research topics on nuclear thermal-hydraulic safety in Korea. These topics were motivated directly by the developmental program of both new nuclear reactor systems and advanced safety analysis tools, and by the lessons learned from the Fukushima accident. The importance of a serious consideration for the multi-dimensional thermal-hydraulic phenomena appearing in nuclear reactor systems is then discussed by taking some research examples and emphasizing their relevance to nuclear reactor safety. Finally some challenging issues for the advancement of nuclear thermal-hydraulic safety technologies are proposed and briefly discussed in close conjunction with recent research efforts. This work will contribute to nuclear safety enhancements, leading eventually to the prevention of any possibility of core melting accidents, which is very essential to the sustainable utilization of nuclear energy worldwide.

KEYWORDS
Thermal-hydraulics, Nuclear safety, Multi-dimensional phenomena, Issues and challenges

1. INTRODUCTION

Recent R&D activities in the area of nuclear thermal-hydraulic (T-H) safety are introduced in this paper within the research framework of preventing severe accidents. In addition, some issues and challenges are discussed from the viewpoint of strengthening the defense-in-depth (DID) concept and a more realistic simulation of nuclear system behaviors for nuclear safety enhancements. Then, the perspectives on advanced T-H safety research are discussed.

Recent research activities for enhancing nuclear safety in Korea have been directly motivated by the development programs of new nuclear reactor systems and advanced safety analysis codes, and by the lessons learned from the Fukushima accident. Recent efforts in the T-H safety research area have focused mainly on two categories. The first is to secure ultimate heat sinks for preventing a severe accident, leading to the development, performance verification, and safety assessment of passive design features for new reactor systems, and an evaluation of the coolability of fuels under accidental situations through sophisticated testing and analysis. Research activities on developing new reactor systems include advanced light water reactors, new research reactors, sodium-cooled fast reactor and gas-cooled reactor. Other efforts are being made on the development of advanced simulation methods of T-H behavior, focusing on the relevance of a multi-dimensional flow to nuclear safety and the development of advanced physical models based on high-precision experiments. These include the development and application of a component-scale T-H analysis code, CUPID, and the coupling of neutronics, and component-scale and system-scale T-H codes for both a multi-scale and multi-physics based safety analysis.

These research works have been conducted not only in collaboration with domestic organizations, but also effectively through international cooperation multi- or bi-laterally.

2. MAJOR TOPICS ON THERMAL-HYDRAULIC SAFETY RESEARCH IN KOREA

Ongoing T-H research program and the current issues for developing new nuclear reactors, such as the APR+ and SMART for Gen-III+ LWRs, new research reactors, and SFR and GCR for Gen-IV reactors, are introduced in close conjunction with developing advanced safety analysis tools for these reactors. New experimental works also have been performed to fill the lack of knowledge or understanding of a physical phenomenon or process, to address new phenomena, processes or system behaviors and to provide the experimental data base for a code development, validation and improvements.

2.1. Advanced Reactors Development

2.1.1. Advanced pressurized water reactors (PWR)

Main research motivation for the advanced PWRs is related directly to recent efforts to introduce new safety concepts, and also to develop new methodologies of safety analysis relevant to these concepts [1]. Some concepts of passive safety system (PSS) implemented in the APR+ and SMART as a measure of ultimate heat sink, and their developmental efforts are introduced here in terms of performance verification and safety evaluation.

Passive auxiliary feedwater system (PAFS) for APR+

Major research efforts associated with the APR+ include: (1) the development of an advanced safety injection (SI) system, DVI+, (2) the development of an improved fluidic device, FD+, a passive SI flow controller, and (3) the development of a PAFS [1,2]. The PAFS, as schematically shown in Fig. 1(a), is an advanced PSS, which is intended to completely replace conventional active system [3]. The PAFS cools down the SG's secondary side, and eventually removes the decay heat in reactor core by introducing a natural driving force: condensing steam in nearly-horizontal tubes of heat exchanger (PCHX) submerged inside the PCCT pool. The important T-H phenomena of investigation for the PAFS include a condensing two-phase flow and their gradual structural developments inside the PCHX tubes, boil-off phenomena outside the PCHX tubes, and the overall natural convection in the PCCT pool [4,5].
The separate effect test, PASCAL, has been performed to verify the heat removal capability of a PCHX design. The integral effect test, ATLAS-PAFS, has shown that the natural circulation in the PAFS system is stably established for DBA and bDBA scenarios, and the water level in intact SG is maintained steadily, as shown in Fig. 1(b),(c). These test data have been used to evaluate the prediction capability of safety analysis codes and identify any code deficiency for transients including a prolonged SBO.

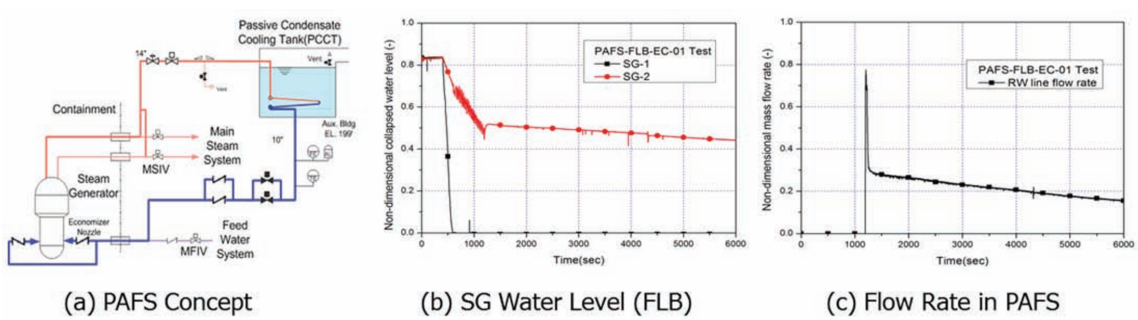


Fig. 1 PAFS Concept and its Actuation during a FLB Accident

Combined emergency cooling concept for a prolonged SBO

Since conventional active safety systems, in general, are not available during a SBO accident, a high pressure core makeup and long-term cooling features, operating passively, are needed to protect core melting even under a prolonged SBO condition. In these passive safety features, T-H safety concerns are newly issued, such as the head loss of natural circulation, three-dimensional (3-D) mixing in a pool, film-wise cooling and the shape of spray water in a containment cooling system, multiple tube bundle effects, air-cooled H/X, etc. In order to extend the operation time of these PSS concepts over 72 hours, a new air-water combined cooling system has recently been proposed [6]. In this concept, the current pool design capacity does not need to be changed, but the cooling capability is extended just by adopting air coolers additionally, as in Fig. 2.

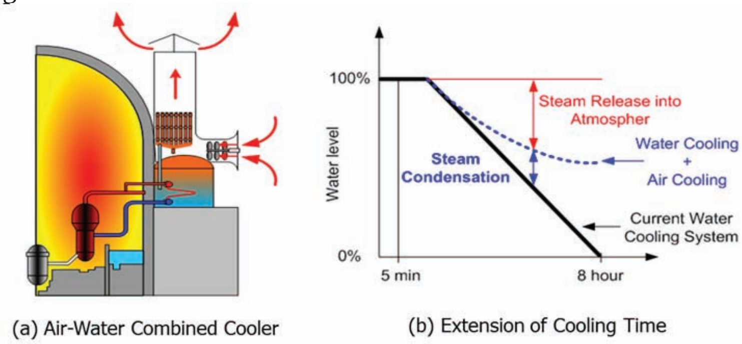


Fig. 2 A Combined Cooling Concept for Coping with Prolonged SBO Condition

__Passive safety concepts for SMART__

A diversity of PSSs were adopted in the SMART, and they include the PRHRS, PSIS, ADS and PCCS as shown in Fig. 3(b), and these have been tested, as shown in Fig. 3(c), on the FESTA facility [7,8]. The PRHRS is composed of 4 trains of H/X, emergency cooldown tank and makeup tank. The passive SI system (PSIS) is composed of 4 core makeup tanks (CMT), 4 safety injection tanks (SIT), pressure-balanced lines to RCP discharge and injection lines to safety injection line. This system is actuated passively only by the gravity force caused by the height difference because all of the tanks are located higher than the injection nozzle around the RCPs. The phenomena of flashing, condensation, and thermal stratification are expected to occur in the CMT, SITs, and piping during the early stage of actuation.

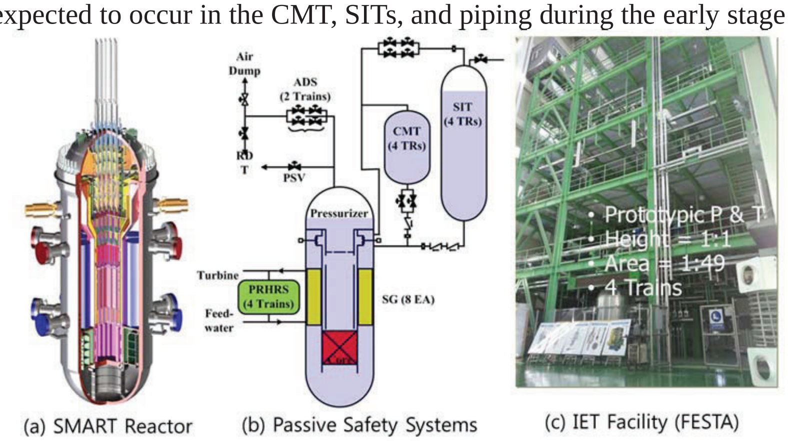


Fig. 3 Passive Safety Systems in SMART and its Testing on the FESTA Facility

2.1.2. Research reactors (RR)

KAERI has newly designed two open-pool type RRs which use plate type fuels: JRTR and KJRR. For both RRs, downward convective flow under low pressure and high subcooling is expected to occur during normal operation condition. From a T-H point of view, understanding of flow boiling phenomena, including onset of nucleate boiling (ONB), onset of flow instabilities (OFI) and CHF, are very crucial since RRs should be designed with sufficient safety margins. Due to the limitation in available experimental data and the prediction methods for downward flow boiling in a narrow rectangular channel under low pressure conditions, new experiments have been conducted [9,10].

Careful investigation on the effect of flow instability (FI) on triggering premature CHF was conducted for downward flow in a narrow rectangular channel as shown in Fig. 4. In-depth observation on local boiling phenomena has also been conducted to improve the understanding of subcooled flow boiling phenomena in a narrow rectangular channel. Finally, the validation of exiting codes for reactor design is in progress, and the improvement of physical models in the codes will also be made.

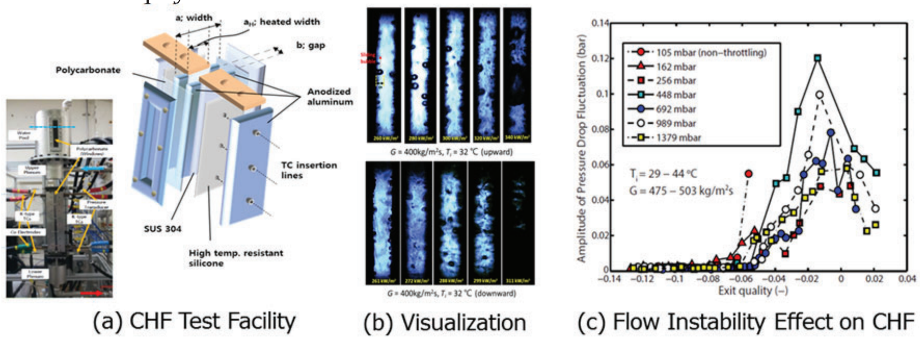


Fig. 4 Boiling Heat Transfer for Downward Flow in a Narrow Rectangular Channel

2.1.3. Sodium-cooled fast reactor (SFR)

The main T-H issues relevant to the SFR design include decay heat removal via natural circulation, sodium-water reaction, and heat transfer in the reactor core and core catcher among others [1]. Here recent T-H research works are introduced focusing on the flow characteristics in subchannels of wire-wrapped fuels and also over the whole reactor vessel containing reactor core and heat exchangers.

<u>**Mixing in subchannels of SFR core**</u>

KAERI is performing the R&D program for developing a SFR reactor, PGSFR. Most influencing parameters on the uncertainty in subchannel analysis are the friction factor and the mixing coefficient. Recently the characteristics of subchannel flow have been experimentally investigated in detail for a wire wrapped 37-pin and 61-pin fuel assemblies [11,12]. The pressure loss in subchannels was measured at the interior and the edge subchannels, and the subchannel flow was measured by the iso-kinetic method. The mixing characteristic was also measured by the wire mesh sensors at the exit of rod bundles, and characterized by the LIF method. The measured flow distribution, typically shown in Fig. 5, is utilized to evaluate the uncertainty in the correlations for pressure drop, friction and mixing factors, which are used in the core thermal margin analysis.

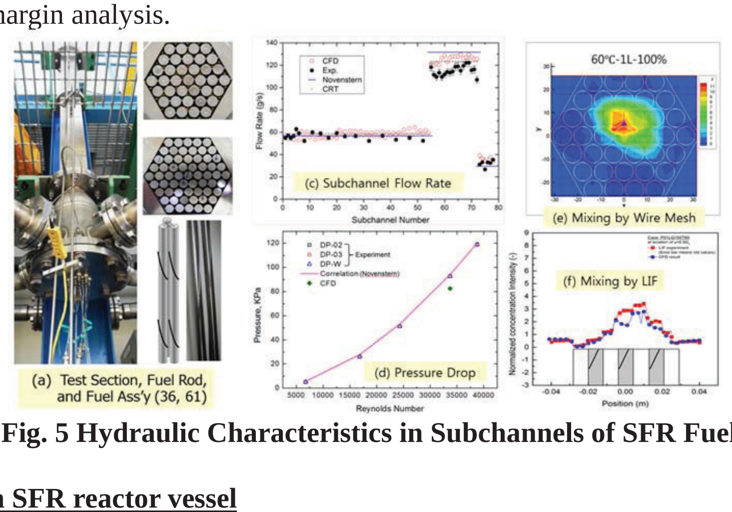
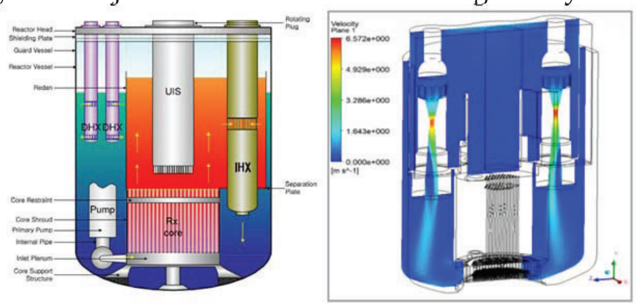


Fig. 5 Hydraulic Characteristics in Subchannels of SFR Fuel

<u>**Flow distribution in SFR reactor vessel**</u>

Since the reactor vessel (RV) in SFR is expected to reveal multi-dimensional flow features strongly, which are sensitive to the internal structure geometry, an experimental program has recently been launched [11]. The major concerns in the pool side are the unbalanced flow characteristics through the IHX, DHX and PHTS pumps, as shown in Fig. 6. For the core side, core inlet flow and outlet pressure distributions are accentual for the thermal margin analysis. The important measuring parameters are the inlet plenum flow distribution, core outlet pressure distribution, IHX and PHTS flow rate, and the pressure loss between major flow paths. The geometric and hydrodynamic similarities are being carefully investigated. The experimental database to be acquired through the current program will be used for the design performance analysis of major devices and the thermal margin analysis.



(a) PGSFR Design (b) Flow in Pool Side

Fig. 6 PGSFR Design and CFD Analysis of Flow Behavior in a Pool

2.1.4. Gas-cooled reactor (GCR)

A nuclear hydrogen development program using GCR has been launched on 2006 in Korea [13]. KAERI is leading the system design and related thermo-fluid experimental research, whereas SNU and KAIST are involved in performing SETs to develop and validate the models in GAMMA+ code, which has been developed to analyze GCR thermo-fluid transients [14]. Main R&D issues in the GCR include the investigations on new T-H phenomena to occur due to specific working fluid such as He and air, the T-H performance of newly adopting components specific to each reactor system, unique heat transfer in major components including the core, heat exchangers, and some accident mitigation measures.

KAERI has developed the lab-scale process heat exchanger (PHE) for the SO_3 decomposition and the bench-scale Alloy617 PCHE for an IHX. Several experimental facilities have been constructed in KAERI: the small-scale gas loop in Fig. 7(a) to verify the integrity and feasibility test of a lab-scale PHE for the SO_3 decomposition, the Helium Experimental LooP (HELP) in Fig. 7(b) for the performance test at the high temperature and high pressure condition, and NAtural Cooling Experimental Facility (NACEF) in Fig. 7(c) to evaluate the performance of RCCS. SNU performed a bypass flow test to quantify the flow rate of the fuel block columns in a prismatic core, whereas KAIST performed the graphite oxidation tests to develop the graphite oxidation kinetics model in the various flow conditions. All the data from these SETs at SNU and KAIST were used to develop and validate the GAMMA+ code.

(a) Small-Scale Gas Loop (b) HELP (c) NACEF

Fig. 7 Test Facilities for Gas-cooled Reactor in KAERI

2.2. Advanced Safety Analysis Codes Development

System-scale T-H code: SPACE

The SPACE code, a best-estimate safety analysis code, is under development to be used for PWR design. This code adopts advanced physical models based on the 2-fluid 3-field approach [15]. It also has the capability of simulating 3-D effects by use of structured and/or non-structured meshes. The code V&V works have been performed at the end of each development stage, as shown in Fig. 8(a), at KAERI. The code assessment has been performed covering several conceptual problems and SET problems generally used in conventional safety analysis code assessment. And the IET's and plant data have also been used to validate overall prediction capability.

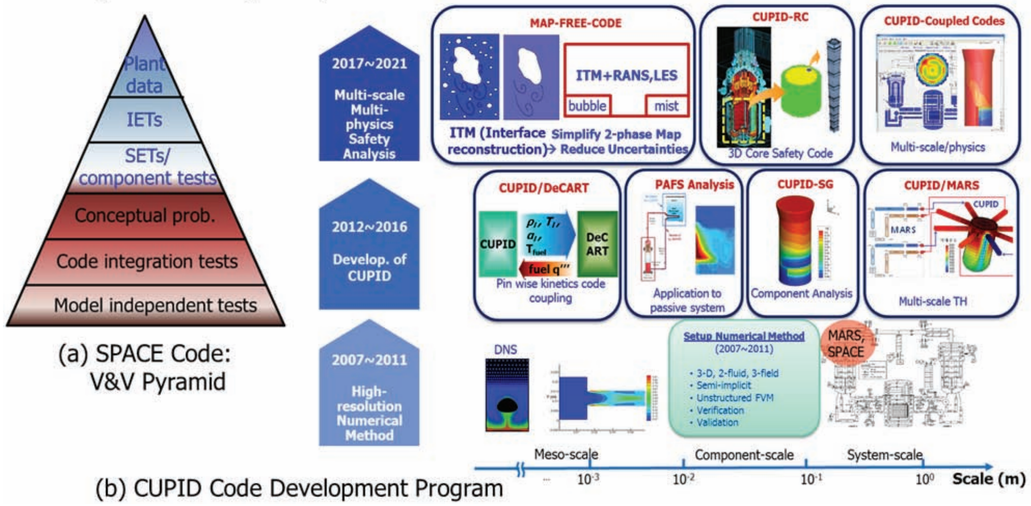


Fig. 8 SPACE and CUPID Codes Development

Component-scale T-H code: CUPID

A realistic simulation of two-phase flow is essential for the advanced design and safe operation of a nuclear reactor system. The need for a multi-dimensional T-H analysis in nuclear reactor components is further increasing with advanced design features in the PSSs. Since the lumped parameter models adopted in system-scale T-H are no longer valid to assess these complicated multi-dimensional phenomena, so KAERI has developed a 3-D T-H code, CUPID, for the analysis of transient, multi-dimensional, two-phase flows in nuclear reactors [16,17]. The code is designed for use as a component- or CFD-scale code, adopting a 2-fluid 3-field model for two-phase flows. The governing equations are solved on unstructured grids, which are very useful for the analysis of flows in complicated geometries. This code has been validated against a set of test problems consisting of conceptual problems and experimental data [5,17,18]. The CUPID code is now being applied to resolving nuclear T-H safety issues which reveal multi-dimensional two-phase flow behaviors, as typically shown in Fig. 8(b). Some of application results, including the analysis of subcooled boiling (DOBO) and pool mixing (PASCAL) tests, will be discussed later. The CUPID is also being used to provide multi-scale and multi-physics simulation of nuclear reactors through its coupling with system-scale T-H codes, and neutron kinetics codes and others, which will also be introduced later.

Containment analysis code: CAP

The CAP code has been developed, based on the 2-fluid and 3-field model, aiming at analyzing the T-H phenomena in nuclear containment of PWR since 2006 in Korea [19]. The evaluation of containment integrity, ECC system performance, and EQ envelop analysis are main concerns among others. This code incorporates most advanced physical models of two-phase flows, together with some unique models, such as water level-oriented upwind scheme and local head model for specific containment modeling. The CAP will be extended to severe accident models in near future.

The V&V works of the CAP is continuously being carried out: conceptual problems, fundamental phenomena, component and principal phenomena, experiment validations, and finally comparison with other codes calculations.

2.3. Basic Research for Nuclear Safety

Coolability of deformed fuels under the reflood conditions

Fuel clad ballooning and the resulting partial flow blockage is one of the major T-H concerns associated with the coolability of partially blocked regions during a LOCA. The fuel relocations initiated at the time of the cladding burst causes a local power accumulation and a high thermal coupling between the clad and fuel debris in the ballooned regions. The previous experiments did not consider the fuel relocation phenomena and the resulting local power increase in the ballooned regions, which is one of the important T-H safety issues in the revision of current LOCA acceptance criteria.

KAERI has performed systematic investigations on the coolability in rod bundles with flow blockages due to clad ballooning in various test geometries. There were no significant effects of the clad ballooning on the reflood PCT. In cases of high power of rod bundles, however, the local power increase due to fuel relocation causes an increase in the PCT compared with intact and ballooned rod bundles. Figure 9 shows the effect of ballooning and fuel relocation on the Reflood PCT [20].

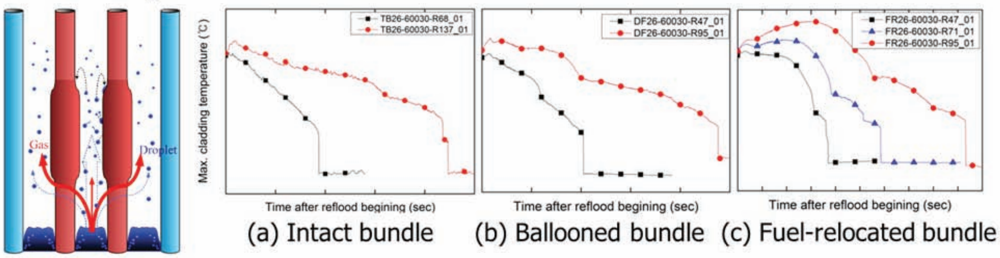


Fig. 9 Effect of Ballooning and Fuel Relocation on the Reflood PCT

Nucleation mechanism

One of the goals of the CMFD tools to simulate a two-phase flow occurring in nuclear reactors is to predict the CHF. For this, the local profiles of various bubble parameters and their axial propagation should be quantified for a subcooled boiling flow at elevated pressure conditions, and the CHF mechanism as well as the local boiling structure near the heating surface at high heat flux condition should be well understood and identified through high-precision experiments. Two examples of our recent efforts for the quantitative visualization of nucleation are briefly introduced here.

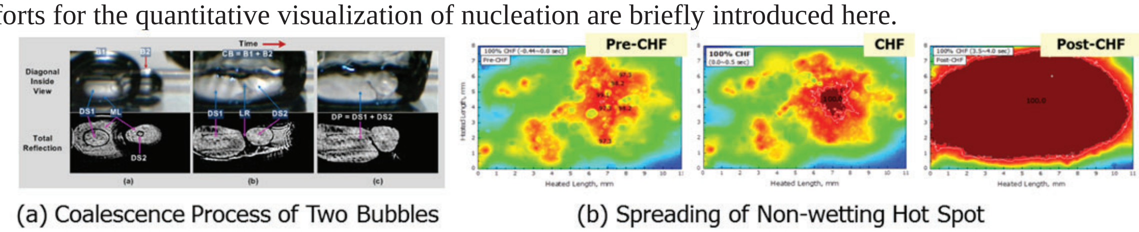


Fig. 10 Coalescence Process of Bubbles and the Spreading of Non-wetting Hot Spot

Regarding the local bubble parameters in a subcooled boiling flow, an experimental program was recently launched to quantify the local bubble parameters for the subcooled boiling flow of a refrigerant R-134a in a pressurized vertical annulus channel [21]. The pressure condition covered the normal operating pressure of PWRs according to the similarity criteria. The radial profiles of the bubble parameters such as void fraction, bubble velocity, interfacial area concentration, and Sauter mean diameter were obtained for seven elevations using optical fiber probes. The local boiling parameters at DNB condition will be one of the targets of future research.

The near-surface boiling structure and CHF triggering mechanism were also investigated for a pool boiling of saturated water by applying total reflection, diagonal inside view, and shadowgraph techniques to transparent ITO boiling surfaces [22,23]. The experimental observations reveal well the near-surface boiling structure and CHF triggering mechanism, as typically shown in Fig. 10.

Condensing stratified flow in a horizontal channel

The direct contact condensation (DCC) occurring in a steam-water stratified flow is an important T-H phenomenon relevant to nuclear safety. The modeling of condensing interfaces in a stratified flow is particularly important in the analysis of pressurized thermal shock and condensation-induced water hammer. The HOCO experiments with a rectangular cross section have been performed to investigate the structures of turbulent thermal mixing and heat transfer near the phasic interfaces of co- and counter-current steam-water stratified flows at atmospheric pressure condition [24]. The PIV and LIF measurement methods have been applied to get CFD-grade data on the temperature and velocity distributions of the water layer in a co-current stratified two-phase flow, especially near the condensing steam-water interfaces, as typically shown in Fig. 11. Interfacial heat transfer coefficient and turbulent heat flux were evaluated from the measurements.

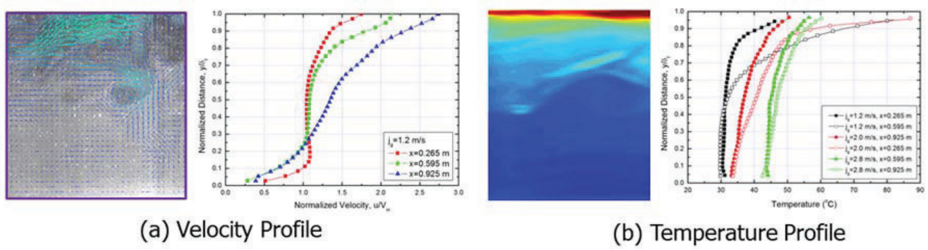


Fig. 11 Velocity and Temperature Distributions Measured near the Condensing Interfaces

Multi-dimensional distribution of two-phase flow

Multi-dimensional two-phase flow phenomena often occur in many industrial applications, particularly in nuclear reactors during a transient period. Proper modeling of complicated behavior induced by a multi-dimensional flow is important for the reactor safety analysis. To validate the performance of SPACE, the DYNAS test, as shown in Fig. 12(a), was performed in a slab geometry of the test section having a scale of 1.43x1.43x0.11 m [25]. Various kinds of 2-D air-water flow configurations could be simulated by selecting different combinations of the inlet and outlet nozzles. The 2-D void profiles were quantified by measuring the local gap impedance at 225 points. Figure 12(b),(c) show a comparison of visualization and impedance contour for a typical case.

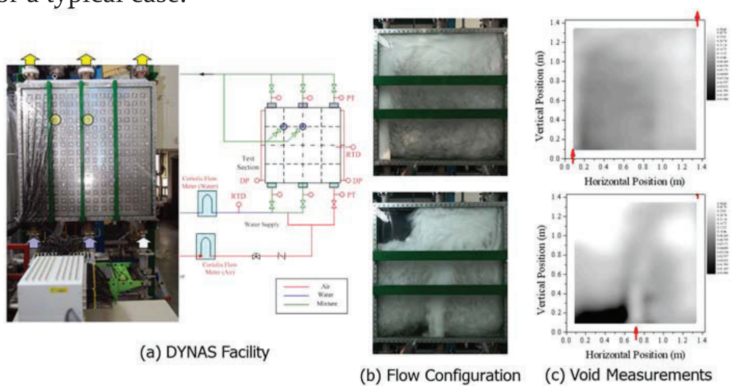


Fig. 12 DYNAS Test for a Two-dimensional Two-Phase Flow

3. MULTI-DIMENSIONAL THERMAL-HYDRAULIC PHENOMENA

3.1. Relevance of Multi-dimensional Phenomena to Nuclear Safety

Most of the currently available system-scale analysis codes in the nuclear T-H area are formulated based on the 1-D approach, and this feature poses practical limitations on their application to reducing the excessive conservatism contained in the existing codes by a realistic simulation or to evaluating the safety margin or performance of advanced reactor systems with a higher reliability. Due to the multi-dimensional characteristics of the T-H phenomena revealed in existing or new reactor systems, it is very important to take into account the following technical concerns:

- To validate the predictability of the existing safety analysis codes, which had been validated mostly against 1-D phenomena in simple geometries, for newly concerned T-H phenomena, and
- To check the applicability of the existing scaling methods, which are usually applicable to a simple flow situation with 1-D characteristics, to analyzing newly emerging complicated phenomena.

For these purposes, it is necessary to evaluate, mostly based on appropriate experimental data, whether the conventional system-scale codes have the capabilities of properly dealing with these multi-dimensional phenomena, and also to check on the appropriateness of using certain experimental data for the code validation from the viewpoint of their scalability to a prototype.

Most of currently available code V&V matrices were mainly issued by OECD/NEA/CSNI [26,27,28]. However, the multi-dimensional phenomena were little concerned at that time. And most of T-H phenomena considered in these test matrices were 1-D and system-wise. In addition, due to a restriction of availability of the SET data base, these matrices treated mainly IET data rather than SET data. Therefore in order to evaluate the code capabilities of predicting multi-dimensional T-H phenomena, new code validation matrix are required. Having in mind that a realistic simulation of multi-dimensional two-phase flow is essential for enhancing nuclear reactor safety, some of multi-dimensional T-H phenomena relevant to the nuclear reactor safety have been chosen here for discussion, including the followings:

- Mixing in a RV: Borated water mixing and bypass, Downcomer boiling, Asymmetric ECC mixing
- Mixing in a SG: Mixing in a hot leg, and SG plena and tubes
- Mixing in a large volume: Pool, Containment, and Spent fuel pool (SFP).

3.2. Mixing in a Reactor Vessel

The DVI nozzles located in the upper part of a RV makes the T-H phenomena in the RV downcomer very complicated and different from those of the cold leg (CL) injection case, and this is believed to govern the behavior of a SI flow in the RV downcomer especially in the case of a LOCA and a MSLB in ALWRs [29]. These phenomena often reveal multi-dimensional characteristics especially in the upper and middle parts of the RV downcomer, and they are closely related to the effectiveness of a SI system. However, most of existing B.E. system-scale codes does not handle these complicated T-H phenomena properly due to their limitations of applicability, which comes inevitably from their mathematical and phenomenological treatments based on the 1-D approach.

Borated water mixing and core bypass

For a CL injcetion mode of SI system, highly borated ECC water is mixed with relatively low borated coolant in the CLs and then the mixed borated water flows into the RV downcomer. For a DVI case, however, delayed boron worth insertion can occur during a MSLB under RCP running condition [29,30]. As shown in Fig. 13(a), a part of ECC water flows directly into the UH when RCPs are running, and mixed with low borated coolant flowing through the UP, HL, SG tubes, CL and RV downcomer and finally returns to the core lately. Therefore, this long circulation through the whole RCS loop will result in delayed boron worth insertion.

Two safety issues exist on the mixing phenomena of borated water in a RV. The one is a *core bypass* phenomenon of borated water in a DVI system when a 1-D system-scale code is applied for a MSLB analysis. The other issue is a quantification of 3-D mixing factor in RV downcomer and core regions. When a system-scale code is used, this core bypass phenomenon is observed due to a high RCP flow during the HPSI injection phase with the DVI system, as shown in Fig. 13(b), where the red colored nodes represent a highly borated zone and core bypass flow. From the CFD analysis and visualization tests, however, a ECC water flowing well into the lower downcomer are observed as in Fig. 13(c),(d).

Currently, a mixing factor for an asymmetric core inlet condition is a newly emerging issue for a proper modeling of core mixing.

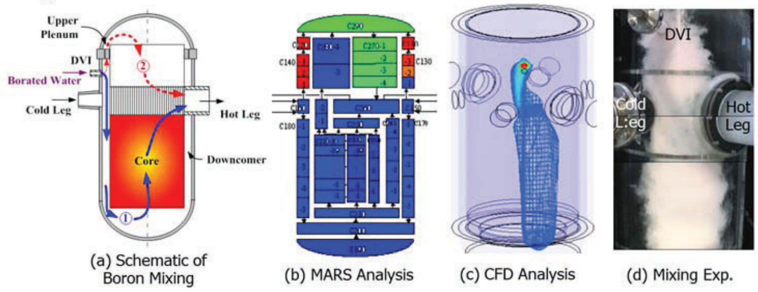


Fig. 13 Mixing of Borated Water in a RV Downcomer

Downcomer boiling

The so-called 'downcomer boiling' phenomenon is a classical safety issue to be addressed in most of PWR since this phenomenon can cause the reduction in hydrostatic head in a RV downcomer and degrade the core cooling capability uring the reflood phase of a postulated LOCA. The DOBO test has been performed to identify this phenomenon, which can be characterized by a strong multi-dimensional behavior as observed in the DOBO test [31]. Due to this features in reality, relatively cool ECC water can penetrate into a lower downcomer region, which leads to enhancing the core cooling. It was found, however, that all the system-scale codes have some limitations in simulating the multi-dimensional behavior regardless of the nodalization method chosen [32,33]. The appropriate treatment for a highly thermal non-equilibrium is also very important for the accurate prediction of ECC performance.

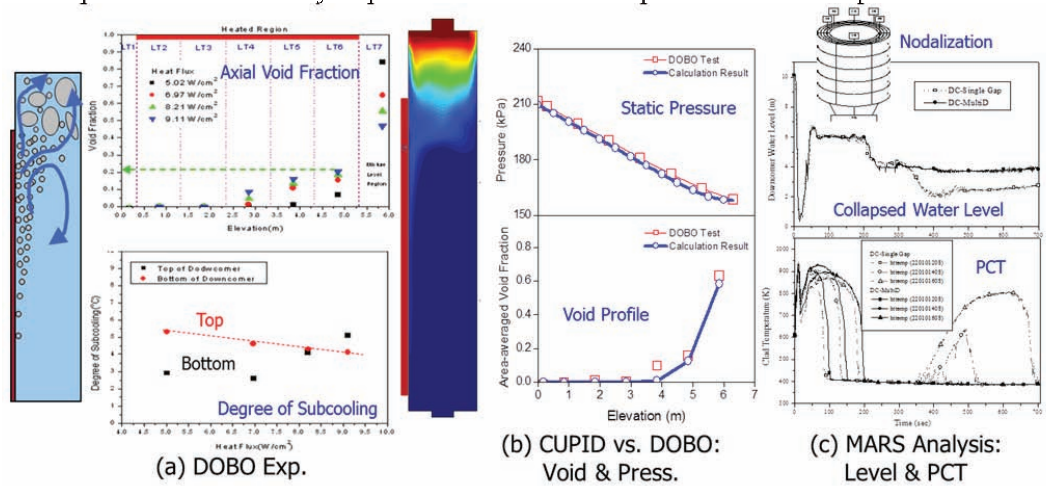


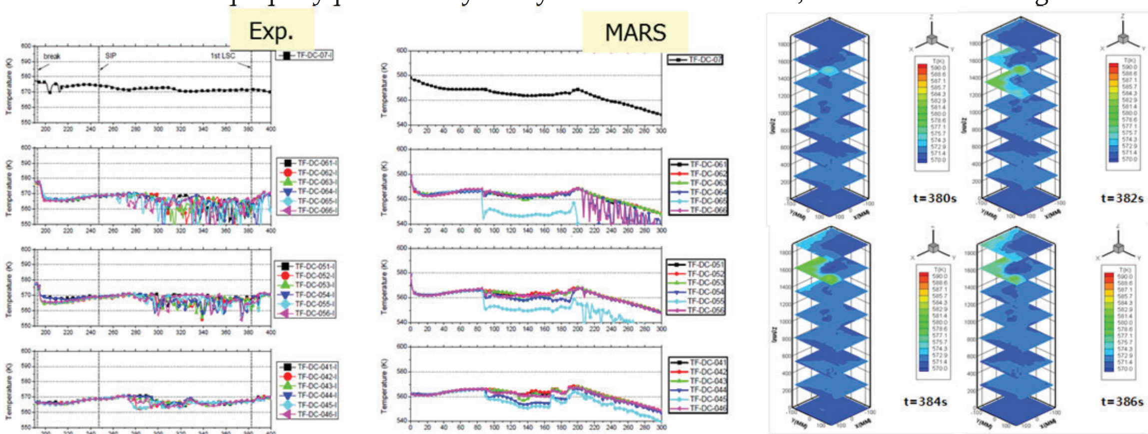
Fig. 14 Subcooled Boiling Revealing Multi-dimensional Behavior in a RV Downcomer

The CUPID code has also been applied to analyze this multi-dimensional two-phase flow in a RV downcomer [18]. From both the DOBO test and CUPID analysis, as shown in Fig. 14(a),(b), the accurate prediction of channel-averaged properties play a critical role in estimating the downcomer boiling phenomena, which affects the emergency core cooling performance directly.

Asymmetric ECC water mixing in a RV during SBLOCA

The multidimensional behaviors in a RV during SBLOCA have been investigated in the ATLAS program [34,35]. A set of new data produced with a help of a fine temperature sensors inside the RV downcomer region indicates clearly that most of system-scale T-H codes fails to predict the flow mixing, regardless of the nodalization method. In fact, a thermal stratification was always predicted by the codes since the predicted cross flow is not so much efficient as the real tests.

Due to such safety significances, an international standard problem (ISP-50) exercise with the ATLAS facility to simulate 50% DVI nozzle break equivalent to 6-inch break was conducted [36]. From the quantitative comparisons with the prediction, it turned out that most of system-scale codes could not correctly predict the thermal stratification in the RV UH and the mixing phenomena at the RV downcomer. Moreover, due to a limitation of the multi-dimensional prediction capability, asymmetric PCT behavior was not properly predicted by the system-scale T-H codes, as can be seen in Fig. 15.



(a) Downcomer Temperature Profile (ATLAS vs. MARS) (b) Cladding Temperature (ATLAS)

Fig. 15 OECD ISP-50: DVI Line 50% Break Simulation Using the ATLAS Facility

To enhance the understanding of multi-dimensional features in the RV, the ATLAS is planned to be revamped to reinforce the instrumentation capability in near future, including installing thermocouple rakes both inside the UH region and near the core exit. And the heater rod surface temperature distribution, representing the PCT, will also be investigated in a multi-dimensional manner [34,35].

3.3. Mixing in a Steam Generator

The reliability of SG tubes and their performance are highly dependent on the mixing phenomena in HLs, SG plena, and tube bundles during the progression of accident. And the SBO-induced SGTR might presumably lead to a containment bypass since, in fact, understanding of the mixing at SG plena is very limited due to the geometry itself. By the way, the current system-scale codes do not predict the exact failure time for the HL, Pzr surge line and SG tubes. Thus, high-resolution temporal and spatial data is needed to validate the codes to understand the temperature distribution in these components of the reactor system. The PSI in collaboration with KAERI plans to launch an international collaborative program for the mixing in HL, SG plena, and SG tubes [37]. This project aims at producing high-resolution data with finer spatial and temporal scales by utilizing facilities of the PSI and the ATLAS in KAERI. In this regard, two tests are planned at the ATLAS facility: (1) mixing under natural circulation, and (2) effect of light gas on natural circulation.

Another example of non-uniform flow behavior in SG tubes has been observed in the LSTF tests showing that a concurrent condensing flow appeared in some U-tubes, while flow stagnation was observed in the others [38]. This causes the reduction of effective heat transfer area in the SG. The inability to predict the U-tube non-uniform flow behavior by a system-scale code results in the overprediction of the RCS depressurization rate, and also in the exaggerated oscillation of the natural circulation flow. It is noted that even though the LSTF is equipped with a very slim SG inlet plenum that may reveal only 1-D flow behavior in it, this kind of non-uniform flow behavior among tubes has been observed. This kind of non-uniform flow among SG tubes has also been predicted mathematically using a 1-D integrated flow model, showing that flow excursion instability can occur under certain conditions [39].

3.4. Mixing in a Large Volume

Mixing in a large volume, such as a pool and containment, is one of the important multi-dimensional phenomena relevant to nuclear reactor safety. There are many examples showing this phenomenon, which include the PCCT pool of the PAFS in APR+, IRWST pool in ALWRs, and SFP of all LWRs.

Mixing in a PCCT pool: Bubble plumes-induced mixing

During an actuation of PAFS in the APR+, evaporative heat transfer from the PCHX makes a decrease of the water level in the PCCT. Boil-off phenomenon outside the PCHX and natural convection flow in the PCCT pool are important phenomena to affect the thermal performance of PAFS during long transients [4]. The PASCAL test showed two kinds of multi-dimensional phenomena, as shown in Fig. 16(a), over the transients: Global circulation induced by bubble plumes above the PCHX tubes and a thermal stratification below the PCHX tubes [3]. Contrary to an active mixing of the water above the tubes, the temperature below the PCHX tubes shows a slow transient until breaking the thermal stratification in the pool, as in Fig. 16(a),(b), which could not be predicted properly by the system-scale codes.

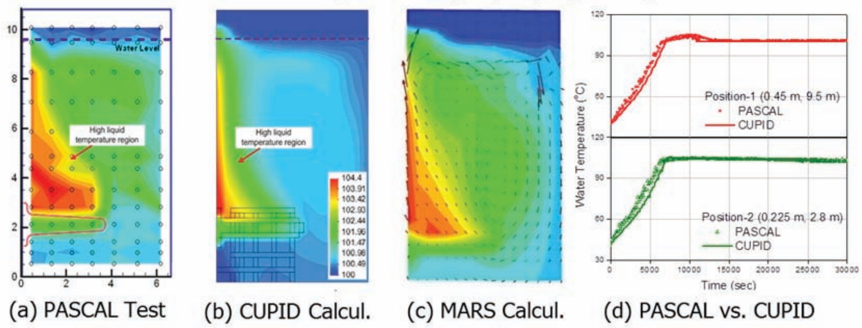


Fig. 16 Thermal Mixing in a PCCT Pool: PASCAL Experiment vs. CUPID Calculation

The PASCAL test data has been used to validate the prediction capability of safety analysis codes and CFD codes with respect to the natural convection in a large water pool. A multi-D component of the MARS code [33] was used to analyze natural convection inside the pool indicating that the calculation underestimates a heat transfer rate at the PCHX in the pool, which are caused by the inaccurate wall heat transfer coefficient in the tube side and incorrect simulation of the natural convection flow in the pool, as shown in Fig. 16(c). The CUPID code has also been applied to the simulation of thermal mixing in this pool. Successful simulation of 8 hours of long transient indicates that the predicted water level and temperature profile is well comparable, as shown in Fig. 16(d), to the measured data [4].

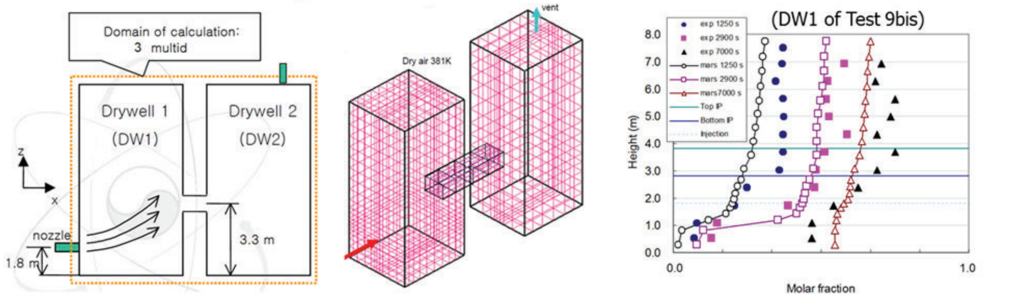
Mixing in a IRWST pool: Condensing jet-induced mixing

Many ALWRs adopt a device of steam discharging into a subcooled water pool for the depressurization or gas venting of relevant nuclear reactor due to the high efficiency of DCC heat transfer and the simple characteristics of engineering application. Practically two kinds of technical concerns exist: The first is the thermal mixing in a water pool, and the other is the thermo-hydraulically induced mechanical loads acting on the structures of relevant systems. The two concerns are interrelated and can be well described only if the local and global behavior of the condensing steam jets and the resultant turbulent jet-induced mixing in a pool are well understood [40].

Predicting the behavior of steam discharge in a pool requires 3-D analysis capabilities. Even though there were some applications of system-scale T-H codes with multi-dimensional analysis capabilities based on the two-fluid model, they might be applicable only for rough estimation of relevant phenomena due to the limitations of at least both the attainable mesh size and the treatable minimal size of physical length scales in the codes. In fact, it is very difficult to properly simulate the two different kinds of steam jet behaviors, namely a forced jet and a buoyant plume, based on the system-scale approach.

Mixing in a containment: Bulk mixing and stratification

Thermal-hydraulics in a containment during an accident include many physical phenomena such as convection, mixing, two-phase flow with phase change and heat transfer. Convective flow is generated in a containment by a jet flow released from a pressure boundary or a break in a case of LOCA. Another important convection occurring in a containment is often induced by a buoyant jet or plume due to a density difference between moving fluid and ambient fluid.



(a) PANDA Test (b) MARS Multi-D Model (c) Steam Concentration: PANDA vs. MARS

Fig. 17 Mixing in a Containment: Comparison of PANDA Test with MARS Analysis

The PANDA test is an investigation on the mixture stratification and mixing phenomena in a large containment [41]. The facility consists of two main large vessels and one inter-connection pipe (IP) as shown in Fig. 17(a). The steam is injected into a vessel and flows into the other through the IP. This test was simulated by the MARS code [33] using multi-dimensional component as shown in Fig. 17(b). From the comparison, it was found that the code could not predict the experimental results as the injection velocity were increasing. This means that the jet stream is not simulated appropriately by a system-scale code. In the low velocity case, however, buoyancy force is dominant due to a low horizontal jet velocity and comparison with experiment shows a good agreement. This is the same as the case of steam-jet induced mixing in a IRWST, which is very different depending on a forced jet and a buoyant plume [40].

Mixing in spent fuel pool (SFP) during LOCA

Accidents in the SFP can be classified into two kinds: a loss of cooling by malfunction of cooling system, and a loss of coolant by a breach in a pool structure. Even though the behavior for completely uncovered SFP associated with a LOCA for both PWR and BWR were conducted at SNL [42], T-H behaviors of SFP with fuel assemblies fully or partly covered by water are relatively unknown [43]. The T-H phenomena in a SFP are characterized by nearly atmospheric pressure, and complex natural convection flows through a large pool.

System-scale analysis codes simulate generally T-H phenomena with 1-D modeling while a 3-D modeling is more adequate for complex T-H phenomena in this large space like SFP. These may pose practical limitations and lack of validation on the applications of the current system-scale codes for a realistic simulation of the T-H phenomena in a SFP. Although CFD codes have capacity to simulate 3-D T-H phenomena, they have several limitations for reliable simulations of SFP accident: simplified modelling due to calculation time and cost, relatively large and coarse meshes, and limited capability and lack of sufficient validations for multi-phase phenomena.

As one of the post-Fukushima actions, IRSN has recently launched the DENOPI project supported by ASN, France [44]. The DENOPI project consists of three different parts: natural convection and boiling in the spent fuel pool scale, T-H at the fuel assembly scale, and oxidation of zirconium in the spent fuel clad scale. KAERI has also recently started a research program on the T-H behavior in SFP accident. The main objective is to characterize T-H behaviors of SFP under a loss-of-cooling and/or -coolant accidents. Several experiments are planned for a better understanding of T-H behavior in partially uncovered SFP. The experimental data will be used to develop related models/correlations, and assess analysis tools such as MARS, SPACE and CUPID codes [16,17,33].

4. SOME CHALLENGES IN THERMAL-HYDRAULIC SAFETY RESEARCH

4.1. Future Direction for Thermal-Hydraulic Safety Analysis

Simulation technologies in science are being continuously developed in parallel with the improvement of computing power. Recently the efforts of applying advanced science level simulation to engineering, e.g., CASL and NEPTUNE programs, have been made by use of high performance computing (HPC) power. The complexities associated with functional requirements for advanced simulations may be categorized according to the target goals of the analysis. In licensing analyses, the goal is to provide a highly confident measure of safety margins and demonstrate the DiD concept. Uncertainties in the licensing analysis must be quantified to a degree that satisfies a level of confidence set by the regulator. For research area, however, the goal is to gain better understanding of physical behaviors and their interaction with system, structures and components (SSC), and materials for nuclear reactor arrangements.
The best strategy for developing next generation safety analysis simulation capabilities may be twofold: a DSA-PSA integrated safety analysis is the one for reducing the uncertainties involved in the safety analysis, and multi-scale/multi-physics analysis is the other for enhancing the prediction accuracy [45].

1) <u>Integrated safety analysis</u>: It requires a significant extension of the phenomenological and geometric capabilities of existing reactor safety analysis codes, enabling detailed simulations that reduce the uncertainties. The use of integrated DSA-PSA tools will help an optimal design of nuclear reactors with risk-informed safety margin quantification [46]. Since the quantification of safety margin for licensing analysis require a huge number of calculations, the integrated analysis tool will benefit surely from modern HPC architectures.
2) <u>Multi-scale, multi-physics analysis</u>: Complex phenomenological models and their dependencies have been simplified in current safety analysis by considering computer hardware limitations. With a HPC, however, these limitations may be removed to allow greater accuracy in representing physical behavior in both DBA and bDBA conditions, and hence more accurate assessment of safety margins will be achievable based on the first-principle methodology. The use of first-principle based methodologies (e.g., molecular dynamics, DNS, micro-behaviors of fuels, etc.) will allow to quantify safety margins accurately, and to widen the possibility of exploring new phenomena that lack the benefit of extensive experimental data.

A full core pin-wise analysis is going to be made in KAERI with the coupling of codes on the fuel (FRAPCON, FRAP-T), neutronics (DeCART, MASTER), reactor core T-H's (MATRA, CUPID or CFD) under the current HPC environments with the following expectation [45]:

- Enhancing the predictability and economy with realistic multi-physic simulation of pin-wise full core analysis which will minimize the interface model assumption.
- Optimizing high-cost large-scale experiments for new reactor and fuel development.
- Reducing developmental cost and time by virtual experiments using the virtual reactor based on high performance computing simulation.

4.2. Multi-scale and Multi-physics Analysis

<u>Coupling of codes</u>
One of the ways to achieve multi-scale and multi-physics analysis is to adopt the coupling of codes handling different scales and physics. Recently in KAERI, a diversity of efforts is planned to set up a multi-physics based code system. The CUPID code has been chosen, as shown in Fig. 18, to play a pivotal role in providing multi-scale and multi-physics simulation of nuclear reactors through the coupling of codes, such as system-scale T-H codes (MARS, SPACE) and reactor kinetics codes (MASTER, DeCART).
The coupling of CUPID-MARS and CUPID-MASTER, and their usefulness is briefly introduced here. In the coupled CUPID-MARS code, two different ways of the coupling can be chosen: "flow field coupling"

and “heat structure coupling”. The heat structure coupling method is applied to the analysis of a PASCAL test, as shown in Fig. 18(b). This shows that the multi-scale T-H analysis using the CUPID-MARS coupling is very useful for the safety assessment of LWRs, which requires different special resolutions [47]. In order to analyze the system transients of strong interactions between neutron kinetics and T-H phenomena, the CUPID code was coupled with a 3-D reactor kinetics code, MASTER [48]. The computational meshes and the coolant temperature at the core exit before and after the CEA drop are shown in Fig. 18(c) indicating that the transient of a single CEA drop leads to an asymmetric core behavior, where the role of a coupled 3-D calculation is crucial.

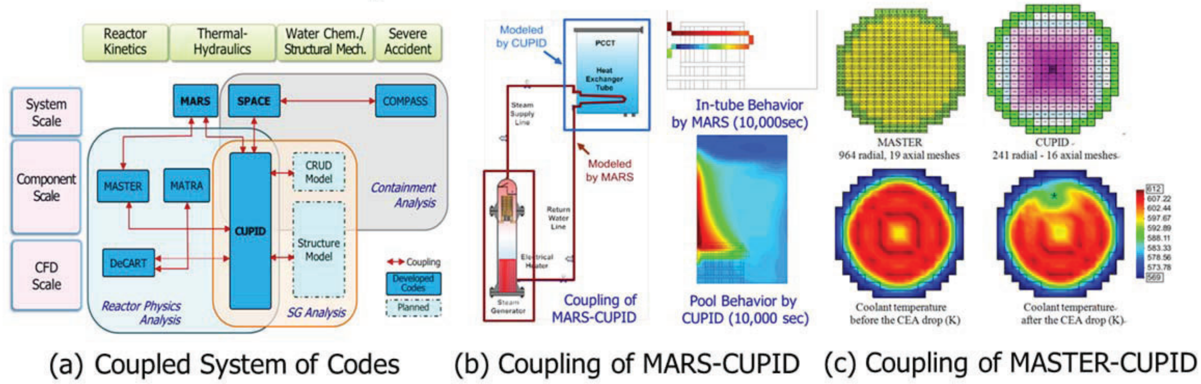


Fig. 18 Coupling of Codes for Multi-scale & Multi-physics Based Safety Analysis in KAERI

Multi-physics analysis : (1) coolability of fuels under accidental conditions

Recent trends for power uprates, plant life extension, and higher burnup of the nuclear power plants (NPPs) face a set of challenging problems that limits the performances of operating NPPs. And basically these problems require multi-physics approach that should resolve interrelations among different physical area, such as T-H’s, chemistry, materials, structures, and neutronics.

One example is a corrosion-related unidentified deposits (CRUD) issue for high burnup fuels during normal operations. The effects of the CRUD on the performance and safety of nuclear fuel should be investigated systematically for transient and accident conditions of the reactor core. The second example is a boron dilution and precipitation which may result in the change of power shape in reactor core. This boron precipitation potentially causes coolant channel blockage or deteriorates the heat removal in the reactor core. Another issue is a debris penetration through the ECCS sump screens, resulting in partial or full blockage of coolant channels in the reactor core that may cause insufficient coolant flow for decay heat removal from the reactor core. The issues on the effect of debris on the heat transfer associated with deposition of debris on the fuel rod surface should be resolved.

Recently KAERI has launched a plan to perform a systematic research on the coolability in rod bundle due to the CRUD on the fuel surface, the flow blockage and surface concentration of LOCA-generated debris, and the boron dilution/precipitation. The experimental data will be used to develop relevant models/correlations, and assess safety analysis codes.

Multi-physics analysis : (2) flow-induced vibration and sludge deposition in SG tubes

Two kinds of T-H concerns that affect the integrity of SG tubes exist among others: flow-induced vibration (FIV) and the outer diameter stress corrosion cracking (ODSCC). Two-phase FIV can cause a mechanical damage such as wear and fatigue at the upper part of tubes, whereas the ODSCC is a major issue that is mainly caused by the deposition of sludge and is very closely related to the T-H conditions in the secondary side of SG. Since an accurate calculation of T-H condition in the SG secondary side is essential, so the T-H analysis of SG by the CUPID code is being done for the prediction of sludge deposition and FIV.

4.3. Multi-purpose Safety Analysis Covering the Design Extension Condition

Right after the Fukushima accident, IAEA published a new safety requirement, SSR-2/1, to replace the previous NS-R-1 in 2012 for strengthening the DiD capability by introducing the DEC concept which include bDBA and severe accident conditions as the design basis [49]. Since the Korea regulatory body is also making similar efforts to strengthen the DiD, a research project is to be launched to extend the SPACE code capability covering both DBA and DEC, and integrate it with nuclear fuel performance and severe accident codes. In this project, new accidental scenarios will be selected to set up a PIRT, and the development requirement and V&V matrix will also be developed. Then, SPACE will extend its simulation capability so that it can properly simulate the DEC scenarios. A severe accident code will also be coupled with the SPACE code so that the transition between DBA to severe accident is smoothly simulated. For the code V&V, experiment data will be gathered to validate the extended SPACE code.

5. SUMMARY AND CONCLUSION

In this paper, current issues and future directions of nuclear thermal-hydraulic (T-H) safety research are discussed, most of which are being performed in close conjunction with new developments of advanced reactor systems and simulation tools, and by incorporating the lessons learned by the Fukushima accident. Major research topics on nuclear T-H safety are introduced, focusing on strengthening the DiD by adopting passive safety measures, and the importance of a serious consideration for the multi-dimensional T-H phenomena appearing in nuclear reactor systems are then discussed by emphasizing their relevance to nuclear reactor safety. Finally, some challenging issues for advanced T-H safety technologies are proposed in conjunction with recent research efforts on applying new approaches such as multi-scale, multi-physics and multi-purpose safety analyses. Then, the perspectives on advanced T-H safety research are discussed.

In conclusion, nuclear T-H safety research should be conducted to obtain *reliable* and *best-achievable* knowledge having in mind the lessons learned from the Fukushima accident including: (1) a combinatory use of proper experiment and analysis technologies, (2) a deepened understanding of the underlying physics, and a continuous development of advanced models and simulation, (3) utilization of best knowledge produced by the best experts and infrastructure, (4) a continued re-evaluation of the safety criteria and methodologies based on the state-of-the art knowledge, (5) active domestic and international collaboration, and (7) efficient communication of R&D outcomes.

Continued efforts on strengthening the DiD concept to practically eliminate a radiological release caused by a severe accident is very important to secure the sustainability of the utilization of nuclear energy worldwide, and this can be achieved based on advancements in nuclear T-H safety research.

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