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On the Scalability of Validation Data for LWR Safety Evaluation

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1. Introduction

System-scale thermal-hydraulics (STH) codes have extensively been used in LWR safety evaluation and rather recently with quantifying prediction uncertainties involved in safety analysis. There still exist some deficiencies, however, originating from our limited knowledge or poor understanding of fundamental physics on key TH phenomena relevant to LWR safety concerns.

In this paper, the scalability of experimental data used for validating STH codes is discussed with their application to safety evaluation accompanied by uncertainty quantification (UQ) in mind. Discussion focuses mainly on our poor understanding of fundamental physics behind widely adopted constitutive relations, mostly developed from unrealistic observation under non-prototypic geometric and TH conditions, and partly on the limited capabilities of STH codes in describing multi-dimensional features of dominant phenomena. Then the perspectives of TH safety research are introduced to enhance our safety evaluation capabilities.

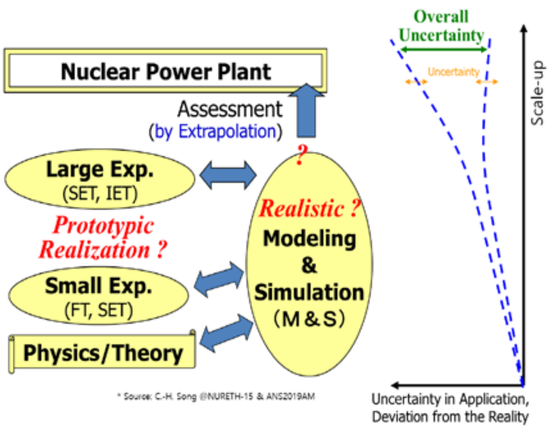

Fig. 1 From Scaled-down Observation to Prototype

2. Some Critical Thermal-Hydraulic Phenomena

The basic concern of 'scalability' often comes from the difference or gap existing between prototypic and down-scaled systems due to their idealization and/or simplification. This presumably causes a dissimilarity of physical phenomena between two systems of different geometric and TH scales from both qualitative and quantitative viewpoints.

Main reasons for checking the scale-up capabilities are associated with *scaling methods* that require the preservation of similarity parameters in a scaled system within formulation capabilities. In some cases, *absolute magnitudes* themselves might seriously affect relevant phenomena.

In this part, some examples are illustrated to show how poor understanding or limited prediction capabilities can affect seriously the safety evaluation of LWRs: (1) some of fundamental two-phase flow phenomena, (2) classical safety issues, and (3) multi-dimensional mixing in a large volume, which is associated with 3-D description of both numerics and physics.

Table 1. Some critical TH phenomena in a typical LWR

			Geometry		Pressure Effect	Multi-physics	Examples	Ref.
			Multi-D	Scale (D,L)				
ECC	LB-LOCA	ECC Bypass	●	●			RV D/C	
	"	Downcomer Boiling	●	●			RV D/C	
	"	Reflood HT & PCT	●	◐		●	Core, Fuel	
	SB-LOCA	Boil-off HT	◐		●	◐	Core, Fuel	
	SBO	Boil-off HT	◐		●	◐	Core, Fuel	
Mixing	"	TI-SGTR	●	◐	◐	●	HL, SG	
	LOCA	Heterogeneous Flow in Singularities	●	●			IL, HL, SG	
	Long-term cooling	Boiling-induced Pool Mixing	●	◐			PAFS, PRHRS	
	"	Condensation-induced Pool Mixing	●	◐			IRWST, CMT	
Two-Phase Flow Fundamental		Interfacial Area Concentration	◐	●	●		Confined, or Large Channel	
		Interfacial Friction	●	◐	●		"	
		Bubble-to-Slug Flow Regime Transition	●	●	●		"	
		Flashing-induced Instability	◐	●	◐		PRHRS, ADS-4	
		Parallel Channel Instabilities	●	◐	◐			

2.1 Basic Two-Phase Flow Relevant to LWR Safety

<u>Hydrodynamics of bubbly flow</u>

There exists a strong dependence of radial void profiles on bubble size even under fixed flow conditions and the turbulence structures of the liquid phase are clearly affected by bubble size. It is also well known that the flow regime transition is very sensitive to the variation of bubble size.

Two distinct modes of structural developments in bubbly flow have been previously observed, which are dependent on the bubble size, and the mechanism of slug formation is also much dependent on the bubble size. The bubble size effect on the behavior of the interactions between the phases could cause variation of the flow structural developments, and eventually influences the flow regime transition.

Most of experimental works and the development of physical models on bubble flow structures based on most of air-water experiments, however, did not properly take into account the bubble size effect, or separate it from other effects.

Pressure effect of nucleate boiling

The profound effects of pressurization were observed on the nucleation and bubble dynamics behaviors. The pressure has a direct influence on the bubble size and this results in the change in various bubble parameters. Significant variations have been observed on the radial profile and axial propagation of the bubble parameters.

The experiments on nucleate boiling under elevated pressure provide new insight on the boiling structure and the influence of pressure on the bubble parameters. The bubble size is significantly affected by a change in pressure, resulting in a decrease in the buoyancy force and relative bubble velocities.

2.2 Classical Safety Issues

Classical issues, relevant to conventional LWR safety over the past several decades, include the following TH phenomena: ECC bypass, downcomer boiling, and reflood heat transfer among others. Most of them are still challenging even in advanced LWRs since they are key attributes to ensuring the reactor core coolability under accidental conditions.

Many of these safety issues are closely related to the multi-dimensional TH phenomena in a reactor coolant system (RCS) of PWR, and they could not be properly dealt with by relying only on classical STH analysis approach.

The STH analysis approach usually adopts the lump parameter (LP) model equipped with one-dimensional (1-D) flow map and the constitutive relations developed based mostly on the 1-D approach. The adoption of parallel pipes model with cross-junction effect, however, may provide improved prediction results even though they are very sensitive to the input values on hydraulic diameter (D_h) and lateral pressure loss coefficient (K) among others. Basically the LP model is satisfied with a mass conserved, but the momentum lost or diffused, and thereby the cross flow and secondary flow are, in principle, hardly to predict.

In this section, the relevance of multi-dimensional phenomena to the coolability of reactor core under accidental conditions is discussed by illustrating some distinct TH phenomena revealing in the reactor vessel downcomer as well as the reactor core.

ECC bypass

The legacy 'Semi-scale' LB-LOCA integral effect tests (IET) revealed about 90% of ECC bypass. This led to hot discussion on the ECCS effectiveness in 1970s and induced public distrust on ECC.

Dowcomer 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 during the reflood phase of a postulated LOCA.

Both the over-estimation of ECC bypass and the unavoidable prediction of downcomer boiling phenomena mainly originate from failing to reveal or predict multi-dimensional phenomena in RV downcomer in the experiments as well as analysis.

Based on conventional STH analysis, which has limitations in simulating multi-dimensional behavior regardless of the nodalization chosen, higher void fraction are estimated leading to smaller hydro-static head in the RV downcomer. Resultantly the prediction of smaller ECC water flowing into the reactor core region and higher fuel temperature inevitably caused some penalty imposed to many of operating PWRs worldwide for decades. The appropriate treatment for a highly thermal non-equilibrium is also very important for the accurate prediction of ECC performance.

Reflood heat transfer

In addition to TH parameters such as the reflood rate and subcooling, fuel clad ballooning and the resulting partial flow blockage are 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.

Multi-disciplinary features of the reflood heat transfer during a LBLOCA also need to be systematically investigated focusing on the effect of spacer grids, liquid droplets, clad ballooning and fuel relocation under single phase convective heat transfer as well as typical reflooding situations.

3. Relevance of Multi-dimensional Phenomena to Nuclear Reactor Safety

Multi-dimensional two-phase flow phenomena often occur in many nuclear reactors during a transient period. Proper modeling of complicated behavior induced by a multi-dimensional flow is important for safety analysis.

Most of the currently available STH analysis codes 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 TH 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 TH 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 STH 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.

Having in mind that a realistic simulation of multi-dimensional two-phase flow is essential for enhancing nuclear reactor safety, some of multi-dimensional TH phenomena are illustrated 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).

4. Perspectives of LWR Thermal-Hydraulic Research

3.1 Challenging Issues on Nuclear Thermal-Hydraulics

<u>Adequacy of Data for V&UQ</u>

The state-of-the art STH codes are equipped with so-called the *multi-D* component model, based mostly on a simple expansion of the 1-D model, or very limited on a full 3-D formulation. However, they suffer from the lack of data for code validation and UQ (V&UQ) against multi-D flow behaviors.

The V&UQ for a BEPU analysis requires the good scalability of validation data: (1) Phase change under prototypic TH conditions including nucleate boiling condition, (2) Local information on momentum and energy transfer for mechanistic modeling to overcome lower spatial and temporal resolution in existing data, (3) Multi-D effect and the scale effect, and (4) Multi-physics related information based on good understanding of TH interaction with structure-material-chemistry: They include data on cladding surface modification under operating conditions such as CRUD, boron precipitation, debris, sludge, and which are prototypic situations of practical interest such as TCD and gap conductance.

The deficiency of experimental data for VUQ includes the phase change under prototypic TH condition including the system pressure effect, and the local information on momentum and energy transfer and their coupled effect for mechanistic modeling. New experimental data should be capable of revealing higher spatial and temporal resolution, and the validity of experimental data will have to be checked from the viewpoints of not only the scaling distortion aspects but also the scale effect itself.

<u>Reliability of Passive System</u>

Increased application of passively operating systems to the reactor cooling during normal operation and accidental situations brings us a strong need to evaluate and verify the reliability of passive systems in addition to investigate the TH performance in order to assure their contributions to nuclear safety comprehensively.

Key viewpoints of investigation are TH phenomena-related uncertainty, instability, scaling, and scale effect. Design performance should be evaluated and demonstrated not only through system-wise analysis and testing in terms of their TH performance and the operational reliability, but also checked in terms of component-wise reliability, failure modes and risks to provide information on PSA approach.

No operational experience showing statistical data for reliability analysis and no consensus on a common approach

3.2 Evolution of Phenomenological Understanding

Continual efforts of improved understanding of relevant physical phenomena should be made in scaling analysis to check the scalability to prototypic situations, and this leads us to revisit the PIRT of our concern.

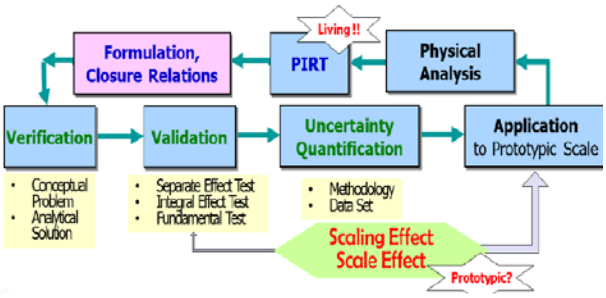

Fig. 2 Revisit of PIRT with Improved Understanding of Key Phenomena

3.3 Physics-based or –informed Modelling

Even with conventional physics-based approach for developing constitutive relations, we will also have to draw our attention to new trends of adopting data-driven approach even though its limitation comes from heavy reliance on the adequacy of data set especially in case of LWR safety issues due to its two-phase flow nature.

Given the role of domain knowledge, physics-informed and machine learning-assisted approach might be a promising alternative based on the sound understanding of relevant domain knowledge. For this purpose, adequacy of data should be evaluated in that they are really rich and applicable to LWR safety issues.

3.4 Future Direction for TH Analysis

The best strategy for developing next generation safety analysis simulation capabilities are twofold: a DSA-PSA integrated safety analysis is the one for reducing the uncertainties involved in the safety analysis, while multi-scale/multi-physics analysis is the other for enhancing the prediction accuracy.

1) Integrated safety analysis: 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. 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) Multi-scale, multi-physics analysis: 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 DEC 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 will allow to quantify safety margins accurately, and to widen the possibility of exploring new phenomena that lack the benefit of extensive experimental data.

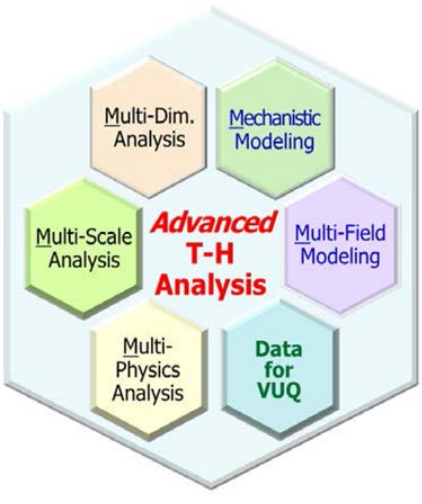

Fig. 3 5 M's featured Thermal-Hydraulic Analysis

4. Conclusions

There still observed some limitations in availability of experimental data, degree of phenomenological understanding of key phenomena, applicability of physical models in terms of model form and model parameters, and the applicability of simulation tools.

Nuclear thermal-hydraulic experiments and analyses for performance evaluation and safety assessment need to be checked in terms of their scalability to determine whether they are prototypic or realistic.

There are some challenging obstacles in the advancement of TH analysis technologies. The first one is to develop innovative experimental techniques which can be applied to physical environments revealing prototypic, local and meso-scales, and anisotropic features of the phenomena of concern. The second one is to suffer from a lack of experimental data that reveals the so-called *5 M's features*, especially from multi-disciplinary aspects covering the DEC with local fuel damages.

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