Report on the 19th International Workshop on Nuclear Safety and Simulation Technology

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Abstract: The 19th International Workshop on Nuclear Safety and Simulation Technology (IMNSST2015) was held on May 24-25, 2015, at Harbin Engineering University, in Harbin, China. The two-day workshop was to promote the key technical communication of numerical nuclear reactor. There were twelve invited presentations at the IMNSST2015, and the subject of the presentations ranges from (i) overview of research progress of numerical virtual reactor in China and CASL in US, (ii) functions introduction of program systems such as MOOSE and NEANS developed for CSAL, (iii) V&V methods for advanced simulation of light water reactor, (iv) details of CFD methods used in nuclear reactor design, such as using structured and unstructured grids without or with adaptive mesh refinement, (v) LWR severe accident analysis codes and simulation of loss of cooling accident in nuclear power station spent-fuel pool, (vi) issues concerning neutron-atypical artifacts introduced by ion irradiation experiments for simulation of neutron irradiation in metals and predicted water chemistry for current and advanced LWR, (vii) transport calculations in reactor physics and radiation shielding, (viii) development of a physical component based nuclear power plant simulator to support nuclear I&C activities, and (ix) fuel performance modeling and simulation. This article provides the overview of the IWNSST2015 with giving condensed summaries of all invited presentations given by international experts.

Keyword: numerical virtual reactor; V&V; CFD; severe accident; irradiation experiments; water chemistry; transport calculations; fuel performance; nuclear I&C activities

1 Introduction

The 19th International Workshop on Nuclear Safety and Simulation Technology (IWNSST2015) was held on May 24-25, 2015, at Meeting Room 370, No. 31 Bldg, Harbin Engineering University. The major topics of this international workshop are related with numerical nuclear reactor. This report will give readers of this journal (IJNS) a comprehensive summary of the two-day workshop.

2 Workshop program and organization of this report

2.1 Workshop program and participants

The 19th International Workshop on Nuclear Safety and Simulation Technology (IWNSST2015) was organized by the College of Nuclear Science and Technology of Harbin Engineering University (HEU). The time table of the two-days is as shown in **Table 1**. There were totally ca. 40 participants which include eleven speakers and HEU organizers, senior master course students, Ph.D. students and young teachers at College of Nuclear Science and Technology (CNST), HEU, and many experts from Chinese nuclear industries. The list of 11 speakers is given in **Table 2**. **Photo 1** shows the group photo of all attendants while **Photo 2** a snap of the workshop room.

Table 1 Time table of the 19th International Workshop on Nuclear Safety and Simulation Technology, May24 and 25, 2015

Date	Time	Presentations
May 24	8:00-8:20	Opening Address by Prof. Zhijian Zhang.
	8:20-9:40	Lecture 1: Introduction and Research Progress of Numerical Virtual Reactor Speaker: Lei Li Chair: Sichao Tan
	9:40-10:05	Lecture 2: Introduction of CASL, MOOSE and NEAMS Speaker: Xiaomeng Dong Chair: Lei Li
	10:05-10:40	Photo taking and Coffee break

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Date	Time	Presentations
	10:40-11:35	Lecture 3: Laser Diagnostic Technique in Verification and Validation of Advanced Simulation of Light Water Reactor Speaker: Sichao Tan Chair: Puzhen Gao
	11:35-14:00	Lunch break
	14:00-15:00	Lunch
	14:00-15:00	Lecture 4: Issues concerning neutron-atypical artifacts introduced by ion irradiation experiments for simulation of neutron irradiation in metals Speaker: Lin Shao Chair: Jing Jiang
	15:00-16:00	Lecture 5: Application of mechanistic code SAMPSON for LWR severe accident analysis to Fukushima Daiichi NPP Speaker: Masanori Naitoh Chair:Hidekazu Yoshikawa
	16:00-16:20	Coffee break
	16:20-17:20	Lecture 6:Application of Computational Fluid Dynamics Codes for Nuclear Reactor Design Speaker:Yong HoonJeong Chair:Hidekazu Yoshikawa
May 25	08:30-9:40	Lecture 7: Using structured or unstructured grids without or with adaptive refinement in computational fluid dynamics Speaker: Yitung Chen Chair: Rong-Jiun Sheu
	09:40-10:45	Lecture 8: Predicted water chemistry for current and advanced light water reactors Speaker: TsungKuangYeh Chair: Yitung Chen
	10:45-11:00	Coffee break
	11:00-12:00	Lecture 9: Radiation transport applications in reactor physics and shielding Speaker: Rong-JiunSheu Chair: Yitung Chen
	12:00-14:00	Lunch break
	14:00-15:15	Lecture 10:Loss of coolant accident simulation of nuclear

Date	Time	Presentations
		power station spent-fuel pool Speaker: Min Lee Chair: Lin Shao
	15:15-15:35	Coffee break
	15:35-16:35	Lecture 11: Fuel performance modeling and simulation Speaker: Wenfeng Liu Chair: Jing Jiang
	16:35-17:30	Lecture 12: Development of a physical component based nuclear power plant simulator to support nuclear I&C activities Speaker:Jing Jiang Chair: Lin Shao
	17:30-17:35	Wrap up by Hidekazu Yoshikawa

Table 2 List of speakers

No.	Speaker	Title/Affiliation
1	Yitung CHEN	Professor Mechanical Engineering Department University of Nevada, Las Vegas
2	Xiaomeng DONG	PhD Student College of Nuclear Science and Technology Harbin Engineering University
3	Yong Hoon JEONG	Associate Professor Nuclear and Quantum Engineering KAIST
4	Jing JIANG	Professor Department of Electrical and Computer Engineering University of Western Ontario
5	Min LEE	Vice President/Secretary General/Distinguished Professor Department of Engineering and System Science National Tsing Hua University
6	Lei LI	Lecturer College of Nuclear Science and Technology Harbin Engineering University
7	Wenfeng LIU	Consultant ANATECH Corp.
8	Masanori NAITOH	Director Nuclear Power Engineering Center The Institute of Applied Energy
9	Lin SHAO	Associate Professor Department of Nuclear Engineering Texas A&M University Longjiang

No.	Speaker	Title/Affiliation
		Scholar Harbin Engineering
10	Rong-Jiun SHEU	Associate Professor Institute of Nuclear Engineering and Science & Department of Engineering and System Science National Tsing Hua University
11	Sichao TAN	Professor College of Nuclear Science and Technology Harbin Engineering University
12	Tsung-Kuang YEH	Professor Department of Engineering and System Science National Tsing Hua University

Photo 1 Group photo of all attendants.

Photo 2 Snapshot of the workshop room.

2.2 Organization of this report

The contents of all the twelve presentations at the workshop are summarized in the subsequent chapters by classifying into the following three categories: (i) Scope of virtual reactor project, (ii) Specific elemental aspects in developing virtual reactor, and (iii) Possible application areas of virtual reactor.

According to Prof. Zhang Zhijian's saying at his opening address of IWNSST2014, the first subject of scope of virtual reactor project is the HEU's on-going leading R&D project in China with the corporation of several universities in China together with the international collaboration between several universities and ORNAL in U.S.A...

3 Scope of virtual reactor project

3.1 Introduction and Research Progress of Numerical Virtual Reactor

Dr. Lei Li presented the overall Introduction and Research Progress of Numerical Virtual Reactor. The synopsis of his presentation is given in the following.

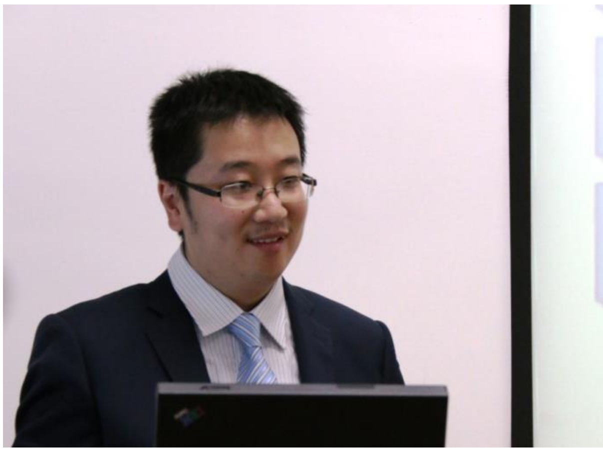

Photo 3 Dr. Lei Li.

In China, the concept of "Numerical Virtual Reactor (NVR)" had been first proposed by Nuclear Power Simulation Research Center (NPSRC), HEU. The "Numerical Virtual Reactor" is defined as a scientific computing system which is made up of science-based, high-fidelity models of neutron transportation, thermal-hydraulic, material performance in reactor vessel and nuclear steam supply system, efficient and robust solver, and HPC platform. The main objective, strategic plan, key technology, research areas, research roadmap and ways of cooperation of NVR are introduced in details. From 2012 to now, NPSRC has undertaken the project of "Research on Key Technology for Numerical Virtual Reactor" which was funded by China government. The latest progress of the project has been also presented in this presentation.

The detail of his presentation is described in his paper [1] in this issue.

3.2 Introduction of CASL, MOOSE and NEAMS

Following Dr. Lei Li's presentation, Mr. Xiaomeng DONG introduced the CASL, MOOSE and NEAMS projects now conducted in U.S.A. The synopsis of his presentation is given in the following.

Photo 4 Mr. Xiaomeng DONG.

The material of his presentation was originally prepared by Dr. Dean WANG, Associate Professor, Nuclear Engineering Program University of Massachusetts Lowell, U.S.A., but due to his sudden unavailability to attend this workshop, Mr. Xiaomeng DONG presented the contents on behalf of Dr. Dean WANG.

CASL is the abbreviation of Consortium for Advanced Simulation of Light Water Reactor, MOOSE for Multiphysics Object-Oriented Simulation Environment, and NEAMS for Nuclear Energy Advanced Modeling and Simulation. Among these three programs, the content of CASL was mainly presented in detail, while the other two just brief introduction of the project. As for the CASL, the background, organization, and goals were first introduced and then the other important aspects such as challenge problems and road map were described in detail. While some information about their goals and capabilities for MOOSE and NEAMS.

The details of CASL, MOOSE and NEAMS are described in [2].

4 Specific elemental aspects in developing virtual reactor

4.1 Laser diagnostic technique in verification and validation of advanced simulation of light water reactor

Prof. Sichao TAN introduced the Laser diagnostic technique related with the verification and validation of the simulation codes. The synopsis of his presentation is given in the following.

Photo 5 Prof. Sichao TAN.

To achieve better simulation of light water reactor, the more accurate core thermal hydraulics codes should be developed, which needs precise verification and validation. The whole field measurement technique is necessary to provide detailed experimental data for improvement of thermal hydraulics codes. However, the traditional measurement techniques cannot realize the whole field measurement, so advanced measurement techniques are needed. The laser diagnostic technique is a potential technique to meet this measurement demand.

The laser diagnostic technique is a series of measurement techniques based on laser scanning, which includes laser-scanning pressure-sensitive paint, molecular tagging velocimetry, particle image velocimetry, laser Doppler velocimetry and laser induced fluorescence. The main advantages of these techniques are that they are non-intrusive and they can realize full-field and instantaneous measurement.
In his presentation, he first introduced the background of laser diagnostic technique, and then the principle and advantages of the technique.

Several application cases were then given which include visualization experiments, temperature measurement experiments, concentration measurement experiments, and void fraction measurement experiments. He stressed that laser diagnostic technique is promising measurement technique, and that the improvement and development of the technique would provide powerful support for advanced simulation of light water reactor.

The detail of his presentation is described in his paper [3] in this issue.

4.2 Issues concerning neutron-atypical artifacts introduced by ion irradiation experiments for simulation of neutron irradiation in metals

Prof. Lin Shao made a review of using ion accelerators for simulation of neutron irradiation effects of various metal materials used in NPP. The synopsis of his presentation is given in the following.

Photo 6 Prof. Lin Shao.

For both reactor safety analysis and reactor designs, it is required to achieve extremely high displacement per atom (dpa) for reactor in-core components, up to 400 dpa and beyond. However, current advanced testing reactors are unrealistic to reach such damage levels. The only realistic option is to use ion accelerators to speed up testing.

To use ion-bombardment as a surrogate irradiation to simulate neutron damage, it is necessary to identify and understand neutron-atypical artifacts. We examine several artifacts which influence micro structural development, requiring care in data interpretation. We focus on Fe irradiated by Fe self-ions to isolate these artifacts from competing spatial/temporal changes in composition. The first artifact arises from combined influence of interstitial-injection and defect-imbalance, leading to void formation in near-surface region with absence of voids in damage peak region. The second artifact concerns the choice of beam scanning mode. For rastered beams, voids become smaller and denser with increasing rastering speed. A defocused beam leads to lower void density but larger void sizes. In general, swelling is depressed with rastering. The third effect comes from pre-injection of helium atoms. Although helium co-implantation can promote void nucleation, it is found that void formation is suppressed if helium loading is too high. These neutron atypical factors must be considered in order to develop a reliable accelerator based testing procedure to simulate high dpa irradiation in true reactor environments.

The detail of his presentation is described in his paper [4] in this issue.

4.3 Application of Computational Fluid Dynamics Codes for Nuclear Reactor Design

Prof. Yong HoonJeong gave a comprehensive overview on computational fluid dynamics methodology. The synopsis of his presentation is given in the following.

Photo 7 Prof. Yong Hoon Jeong.

Development of computing power allows more sophisticated numerical simulation of single phase heat and mass transfer and makes two-phase simulation more likely. Direct numerical simulation of single phase heat and mass transfer is accepted as a reliable replacement of high fidelity experiments with limited scale of the problem. In my presentation, various turbulent models are validated against the direct numerical simulations to expand the role of computational fluid dynamics in the design and analysis of sodium cooled fast reactor core. In addition to the application of CFD in design and analysis, an application of a Computational Multi Fluid Dynamics code – CUPID- is introduced for expanding the role of CFD in two-phase system design and analysis. Examples of natural circulation were also presented

The detail of his presentation is described in his paper [5] in this issue.

4.4 Using structured or unstructured grids without or with adaptive refinement in computational fluid dynamics

Prof. Yitung Chen gave constructive review on organization method of grids for computational fluid dynamic. The synopsis of his presentation is given in the following.

Photo 8 Prof. Yitung Chen.

An optimized and good quality of computational grids for computational fluid dynamics (CFD) is essential for every production flow solver including in the applications of mechanical, nuclear, aerospace, marine, chemical, civil, biomedical engineering, but there are a lot of different computational mesh types out there. The numerical error in modeling and simulation is often related to mesh quality since the physical domain is discretized to numerical or computational domain. The first production codes for solving the Euler or Navier-Stokes equations and so on are so-called “structured” grids. In a structured CFD grid, user can construct a mapping function that will transform a curvilinear mesh to a uniform Cartesian grid. This allows a given point's neighbors to be easily identified and efficiently accessed, which allows for speedy CFD codes.

On the other hand, it means that user cannot just plop down additional points into the CFD grid if user decides to need more in some area. The points go into a mesh, and user must add a whole line at a time (in 2-D—a plane at a time in 3-D) in order to preserve the structure of the mesh.

Structured meshes come in three basic flavors: H-grids, O-grids, and C-grids. Many of the successful algorithms in unstructured mesh generation have found their roots in the field of computational geometry. Computational geometry is the theoretical science concerned with defining or postulating the existence of specific geometrical constructs (*i.e.*, particular triangulations), devising algorithms for generating these constructs, and analyzing the complexity of these algorithms (usually asymptotic worst case complexities).

Unstructured CFD grids do not have a direct mapping between where things are in memory and how they connect in the physical space. Unlike structured CFD grids, the cell at location 'n' in memory may have no physical relation to the cell next to it at location 'n+1'. This means that an unstructured solver has to do a bit more work to figure out where neighboring cells are, but it allows for a lot of freedom in constructing a CFD grid. User can add resolution where user needs it, and coarsen the grid wherever user doesn’t. User can always convert a structured grid to an unstructured one, but not necessarily the other way around.

Unstructured grids offer maximum flexibility: in theory, user can put a grid around anything, no matter how complicated. Also, user only has to put grid resolution where it is actually needed.

In a structured grid, because everything is connected together, user often has to cluster points out in the middle of nowhere in order to achieve the resolution user requires in other parts of the domain. Unstructured grids allow user to avoid other undesirable features that frequently pop up with structured grids.

So, why would anyone want to use a structured CFD grid? There are several reasons. The selection of proper structured or unstructured grids will be demonstrated. The mesh adaptive techniques of r-, h-, and p-method will also be reviewed. An example of parallel multiblock mixed finite element methods for elliptic problems on unstructured grids were also discussed in his presentation.

4.5 Fuel performance modeling and simulation

Dr. Wenfeng LIU gave a review on Fuel Performance Modeling and Simulation technology. The synopsis of his presentation is given in the following.

Photo 9 Dr. Wenfeng LIU.

Fuel performance issues are important to the safe and economic operation of nuclear reactors and to the mitigation of the consequences in accident conditions. Fuel performance modeling and simulation with the aid of a computer code plays an essential role in predicting and understanding the related fuel behaviors. A number of Light Water Reactor (LWR) fuel performance issues such as fission gas release and swelling, fuel cracking and relocation, cladding creep and elongation, cladding corrosion and hydriding, fuel thermal properties degradation, and fuel failures under normal operation, transient, and accident scenarios were reviewed. The state of the art of fuel performance code development are introduced, which include the development and implementation of material/behavior models, the frame work of the finite element code, and the verification and validation of material/behavior models and fuel code. A number of challenging issues in the modeling and simulation of fuel behavior in the coupling of a fuel code with a core simulator, development of three-dimensional code, and in bridging the modeling efforts at different scales are briefly discussed.

The detail of his presentation is described in his paper [6].

5 Possible application areas of virtual reactor

5.1 Application of mechanistic code SAMPSON for LWR severe accident analysis to Fukushima Daiichi NPP

Dr. Masanori Naitoh’s broad presentation covered not only the modeling of LWR severe accident but also its application for verifying it by real severe accident. The synopsis of his presentation is given in the following.

Photo 10 Dr. Masanori Naitoh.

SAMPSON is a fully mechanistic system code for analysis of LWR severe accident. The accident progressions of the Fukushima Daiichi Nuclear Power Plant (NPP) Units 1, 2, and 3 were analyzed with the improved SAMPSON code.

The major improvements were incorporation of new modellings for phenomena specific to Fukushima Daiichi NPP; (1) direct steam release from the damaged portion of the gasket of the safety relief valve (SRV) and from the buckling portion of the in-core monitor guide tube (ICMGT) in the core region, (2) direct corium release from melt portion of the ICMGT at the reactor pressure vessel (RPV) bottom to the pedestal, and (3) performance of the reactor core isolation cooling system under two-phase flow conditions.

The analysis results show that (1) Unit-1: almost all of the core materials had melted and fell down onto the pedestal floor before the start of the alternative water injection by fire pumps, (2) Unit-2: about 30% of the core materials had melted down into the RPV lower plenum and stayed there (*i.e.* no RPV bottom break), and (3) Unit-3: the core meltdown continued for a while even after the start of alternative water injection, and almost all of the core materials had melted and fell down onto the pedestal floor.

The results on Units -2 and -3 still have some uncertainties since the SRV working period of Unit-2 and the timing of the termination of the high pressure coolant injection system of Unit-3 are still unknown.

The detail of his presentation is described in his paper [7] .

5.2 Predicted water chemistry for current and advanced light water reactors

Prof. Tsung-Kuang Yeh made a comprehensive overview on water chemistry in light water reactor. The synopsis of his presentation is given in the following.

Photo 11 Prof. Tsung-Kuang Yeh.

Intergranular stress corrosion cracking (IGSCC) on stainless steel and nickel-base alloy components has been a major material degradation issue for decades for all light water reactors (LWRs) around the world. To ensure operational safety, an optimization on the coolant chemistry in the primary coolant circuit of a nuclear reactor is essential no matter what type or generation the reactor belongs to. In light of the safety demand and the lack of essential water chemistry information in a LWR, the only feasible approach to accomplish the foregoing task of understanding water chemistry state is to conduct a series of theoretical analyses. In this study, a radiolysis model was therefore developed for analyzing the concentrations of electroactive radiolysis products in the coolant. The simulation would produce predicted results pertinent to the water chemistry variation and the corrosion behavior of structure materials in the primary coolant system of a LWR.

The detail of his presentation is described in his paper [8] in this issue.

5.3 Radiation transport applications in reactor physics and shielding

Prof. RongJiunSheu gave a comprehensive lecture on the application of radiation transport equation in reactor physics and shielding. The synopsis of his presentation is given in the following.

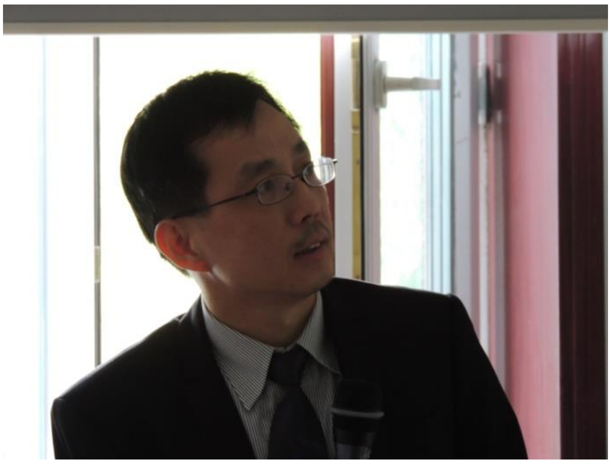

Photo 12 Prof. RongJiun Sheu.

There were three topics in hispresentation on transport on reactor physics and radiation shielding: (1) neutronic studies of very high temperature reactors (VHTR) and molten salt reactors (MSR), (2) evaluation of the fast neutron fluenceon reactor pressure vessel (RPV), and (3) source terms and radiation shielding for the transportation and dry storage of spent nuclear fuels (SNF). Compared with conventional LWRs, the fuel element of VHTRs presents a doubly heterogeneous geometry that should be taken in account carefully in neutronic simulations. MSRs are distinguished by the circulation of liquid fuel in and out of their cores, which provides unique advantages for online fuel addition and medical isotope extraction. However, these features complicate the corresponding reactor physics analysis to be compared with solid-fuel designs. Accurate determination of the fast neutron fluence at various locations of RPV is important because of radiation damage concern. His study investigates the effects of various simplifications of core modeling on the resulting flux distribution and compares the results of two independent transport calculations using RAMA and MAVRIC. A complete radiation shielding analysis for an interim SNF storage involves many difficulties: source term and geometry modeling, deep penetration calculation, radiation streaming, and also skyshine evaluation. To make a complicated and deep-penetration calculation computationally practical or feasible, efficient algorithms are always desirable and indispensable in most cases. His lecture introduced the high-fidelity source term evaluation and radiation shielding calculations for a planning of SNF transportation and dry storage in Taiwan.

The detail of his presentation is described in his paper [9] in this issue.

5.4 Loss of coolant accident simulation of nuclear power station spent-fuel pool

Prof. Min Lee introduced the loss of coolant accident simulation of spent-fuel pool in nuclear power station. The synopsis of his presentation is given in the following.

Photo 13 Prof. Min Lee.

The core melt down accident of Fukushima Nuclear Power Station on March 11th, 2011 alerted nuclear industry that the long term loss of cooling of spent fuel pool may need some attention. The target plant analyzed is the Chinshan Nuclear Power Station of Taiwan Power Company. The 3-Dimensional RELAP5 input deck of the spent fuel pool of the station is built. The results indicate that spent fuel of Chinshan Nuclear Power Station is uncovered at 6.75 days after an accident of loss cooling takes place and cladding temperature rises above 2200℉ around 8 days. The time is about 13 hours earlier than the results predicted using simple energy balance method. The results also show that the impact of Counter Current Flow Limitation (CCFL) and radiation heat transfer model is marginal.

The detail of his presentation is described in his paper [10].

5.5 Development of a physical component based nuclear power plant simulator to support nuclear I&C activities

Prof. Jing Jiang presented a physical component based nuclear power plant simulator. The synopsis of his presentation is given in the following.

Photo 14 Prof. Jing Jiang.

Prof. Jing Jiang presented his lecture which is not a sort of virtual reactor but a materialized nuclear power simulator to support I&C research and development. This simulator relies on physical sensors and actuators, and hardware simulated dynamic processes. From an I&C perspective (low voltage, current and power), the simulator behaves as if the I&C system is connected to a real process. The simulator consists of 6 physical processes, and can be configured to form 15 different control loops to support various I&C research needs. The simulator has been deliberately overly instrumented with 25 different industry-grade sensors and 30 industry-grade actuators so that it can be easily interconnected with external hardware I&C devices.

This simulator allows researchers to test variety of instrumentation and control algorithms in a dynamic environment in real-time, similar to that encountered in a real nuclear power plant. The biggest advantage of using physical simulator, as opposed to software-based simulator, is that one does not have to deal with timing issues when connecting physical control devices for closed-loop testing and validation process. This simulator also facilitates development of new control and diagnostic schemes, *e.g.* smart sensors, applications of wireless sensor networks, in a dynamic environment. Our experience so far has indicated that such a simulator is a must-have item in research and development of I&C systems for nuclear power plants.

The detail of his presentation is described in his paper [11] .

6 Conclusion

The 19th International Workshop on Nuclear Safety and Simulation Technology (IMNSST2015) was held on May 24-25, 2015, at Harbin Engineering University, in Harbin, China. The two-day workshop was to promote the key technical communication of numerical nuclear reactor, which is beneficial to the development of numerical nuclear reactor.

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