

THERMAL HYDRAULIC SAFETY RESEARCH AT JAEA AFTER THE FUKUSHIMA DAI-ICHI NUCLEAR POWER STATION ACCIDENT

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ABSTRACT

This paper summarizes thermal-hydraulic (T/H) safety studies being conducted at JAEA based on the consideration of research issues after the Fukushima Dai-Ichi Nuclear Power Station accident. New researches have been initiated after the accident, which are related to containment thermal hydraulics and accident management (AM) measures for the prevention of core damage under severe multiple failure conditions. They are conducted in parallel with those initiated before the accident such as a research on scaling and uncertainty of the T/H phenomena which are important for the code validation. Those experimental studies are to obtain better understandings on the phenomena and establish databases for the validation of both lumped parameter (LP) and computational fluid dynamics (CFD) codes. The research project on containment thermal hydraulics is called the ROSA-SA project and investigates phenomena related to over-temperature containment damage, hydrogen risk and fission product (FP) transport. For this project, we have designed a large-scale containment vessel test facility called CIGMA (Containment InteGral Measurement Apparatus), which is characterized by the capability of conducting high-temperature experiments as well as those on hydrogen risk with CFD-grade instrumentation of high space resolution. The effectiveness of AM measures against accidents with severe multiple system-failures are investigated using the ROSA/LSTF facility that is a T/H model of the four-loop Westinghouse type PWR. Scaling and uncertainty of the T/H phenomena are investigated with several separate effects test facilities including those for the core heat transfer under steady-state and transient conditions. This paper describes the plans for those researches and results obtained so far.

KEYWORDS

Thermal hydraulics, ROSA, LSTF, Containment, Code validation

1. INTRODUCTION

It was widely recognized in Japan that nuclear regulation regarding prevention and mitigation of severe accidents should be strengthened since the Fukushima Dai-Ichi Nuclear Power Station accident in 2011. The importance of severe accident research, therefore, was reemphasized to predict the accident progression, develop the accident management (AM) measures and establish technical basis for the regulation. In addition to this recognition, the nuclear regulation authority (NRA) in Japan recently clarified a possibility to develop their own system analysis thermal hydraulic (T/H) code in a report showing the policy on the safety study in 2013[1] and initiated some studies for the development.

With consideration of the above background, the thermal hydraulic safety research group in JAEA is currently conducting researches for 1) the containment thermal hydraulics, 2) the accident management (AM) measures to prevent the core damage under severe multiple system-failure conditions, and 3) scaling effects and uncertainty of the important phenomena. The first two researches were initiated after the Fukushima accident, considering the importance of improvement of the analysis capability for the severe accident related phenomena and the effectiveness of AM measures. Although the AM measures have been studied for more than twenty years using the ROSA/LSTF facility, the new study using ROSA/LSTF focuses on accidents with more severe system failures as observed in the Fukushima accident. The research for scaling effects and uncertainty is mainly for the code validation and the improvement of the best estimate plus uncertainty (BEPU) method, which was conducted before the Fukushima accident. We will conduct several separate effects tests for core heat transfer and the important phenomena that significantly affect the peak cladding surface temperature (PCT).

2. CONTAINMENT THERMAL HYDRAULICS

The ROSA-SA (Severe Accident) project was initiated in 2013 for the investigation on T/H phenomena related to over-temperature containment damage, hydrogen risk and FP transport which are considered very important in terms of mechanisms for the containment damage and source term transport [2]. So far the most focused activity for this project has been the design of a large-scale containment test facility called CIGMA (Containment InteGral Measurement Apparatus). The CFD analyses also have been conducted to identify the technical issues important for the code/model validation and improvement, which include the erosion of density stratification by imping jet, and condensation with noncondensables.

The CFD analyses will be used for the detailed validation of designs of safety equipment in the containment and AM measures for the nuclear regulation, while those are not suitable for the analyses on the accident propagation considering many sensitivity effects and the risk evaluation. For those purposes, we have to rely on the Lumped-Parameter codes such as the MELCOR. Therefore, the LP code models will be validated and improved in the ROSA-SA program using results of the experiments and/or the CFD analyses.

Regarding the research on the FP transport, we will conduct the pool scrubbing tests focusing on the relationship between the decontamination factor and two-phase flow behavior, for which small test setup has been constructed. We will also conduct tests on aerosol behavior in large space in the later phase of the project, although details are not yet planned.

2.1. The CIGMA Facility

The motivation of our group to investigate the containment thermal hydraulics comes mainly from presumed over-temperature containment failure and occurrences of hydrogen explosions during the Fukushima accident. Since the T/H behavior related to hydrogen risk has been extensively investigated in research projects using test facilities such as PANDA [3,4], MISTRA [5], and THAI [6], characteristics of those existing facilities were carefully investigated for the design of the CIGMA facility to avoid redundant experiments and obtain new technical findings. From those considerations, the CIGMA facility was designed to have the capability of conducting high-temperature experiments as well as those on hydrogen risk with CFD-grade instrumentation of high space resolution

Table 1 summarizes the design specification of CIGMA compared to existing facilities of THAI, MISTRA, and PANDA. The test section of the CIGMA facility has a cylindrical geometry with 2.5m in diameter and 10m in height with nozzles for fluid injection and discharge, and large glass windows with 650 mm in diameter for optical measurement (see Fig.1 for the test section of the CIGMA facility). The

Table 1 Comparison of design specifications CIGMA and existing facilities

		THAI	MISTRA	PANDA	CIGMA
Organization		GRS	CEA	PSI	JAEA
Height	m	9.2	7.3	25(total)	10
Diameter	m	3.2	4.25	4	2.5
Volume	m ³	60	100	183*1	~ 50
Pressure	MPa	1.4	0.6	1.0	1.5
Temperature	K	453	473	473	573~473*2
Power	MW	0.1		1.5	0.2
Instrumentation		~200	~370	~1000	
Thermocouple		>160	>300		~600
Gas sampling		~20	~50	~100	~100
Window		20		6	~15
Velocimetry		PIV/LDV	PIV/LDV	PIV	PIV/LDV

note

*1: two vessel + interconnection pipe

*2: 573~473 K for boundary wall depending on pressure, and up to 973K at gas injection nozzle

facility has systems for outer surface cooling and inner spray cooling, and support systems to provide electricity, water, steam and gases.

The CIGMA facility allows the high temperature gas injection up to 973 K, of which pressure boundary can withstand up to 573 K at atmospheric pressure and 473 K at 1.5MPa. These temperature and pressures conditions seem to be much higher than those of the existing database on containment T/H behavior. For example, the maximum pressure and temperature at the gas injection nozzle was 2.6 bar and 423 K for PANDA, and 1.1 bar and 421 K for MISTRA for the SETH 2 project conducted using those facilities [7]. The CIGMA test, therefore, can enlarge the T/H conditions of existing databases. Such enlargement is considered important because even the state-of-the art CFD codes/models rely on empirical relationships for modeling of turbulence, diffusion of scalar properties and so on.

The instrumentation with high space resolution is indicated by the number of thermocouples and sampling probes per volume, which are largest among the facilities listed in the table 1. The gas sampling is for the measurement of the molar fraction of a gas component using the quadrupole mass

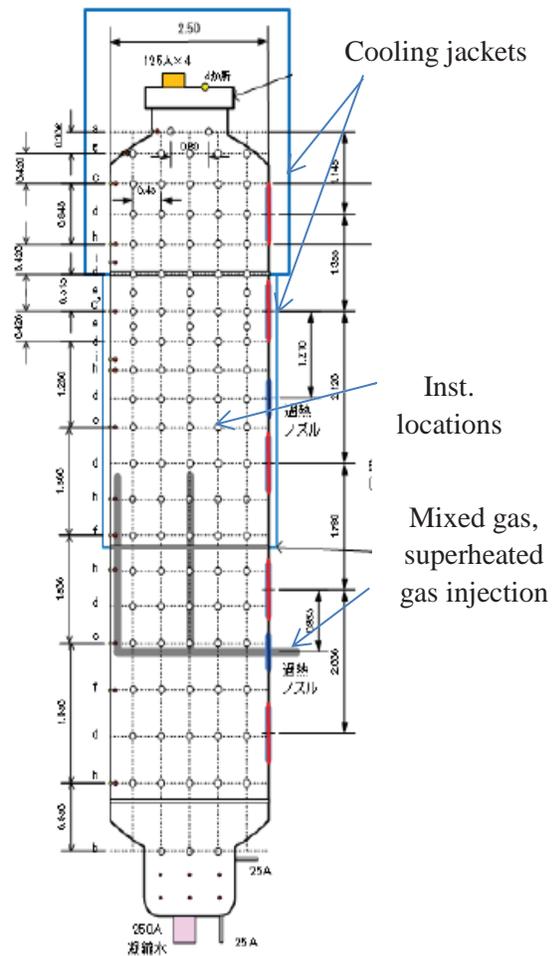


Figure 1 The test section of the CIGMA facility

spectrometer (QMS). The glass windows are for the optical measurement using the Laser-Doppler Velocimetry (LDV) and Particle Image Velocimetry (PIV). Those measurements are to produce database useful for the validation and improvement of the CFD code/model.

Figure 2 illustrates experiments planned at CIGMA. The experiments will focus on T/H phenomena including the erosion of density stratification due to impinging jet or plume, wall temperature response to high-temperature impinging jet, and cooling on the containment outer surface or by spray together with effects of configuration and geometries including internal structure of bulk head and walls.

The experiments also investigate the hydrogen gas behavior during the containment vent and nitrogen substitution, and T/H responses to systems operated outside the design conditions. For example, we will investigate the effects of spray cooling under very small spray flow conditions since we cannot assume systems for AM measures to be operated inside the design conditions as observed in the Fukushima accident. The fabrication of the components started in 2014 and the first test will be conducted in 2015.

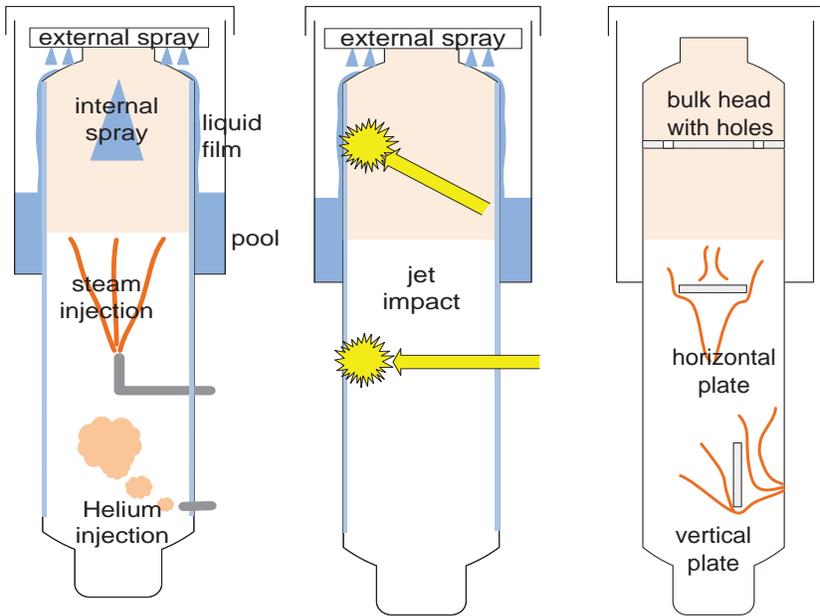


Figure 2 Planned experiments using CIGMA

2.2. CFD Analysis on the Erosion of Density Stratified Layer by Impinging Jet

Because of importance in predicting the distribution of hydrogen-gas in the containment, helium gas distributions affected by various T/H phenomena have been investigated extensively in the previous research projects. For example, the erosion rate of stratified layer by impinging jet or plume was investigated in the international standard problem ISP-47, where one experiment conducted using the THAI facility indicated that the experimentally observed density stratification was not reproduced using the CFD codes by all participants [8].

In order to identify technical issues and improve physical models, the CFD analyses were conducted on the erosion of stratified layer by imping jet using the Reynolds Averaged Navier Stokes (RANS) model and the Large Eddy Simulation (LES) model [9,10]. It is known that the RANS analysis is advantageous in computational cost and suitable for the calculation of large geometry, while the LES analysis is believed to be more accurate when using sufficiently fine mesh. The RANS turbulence model was, therefore, investigated and modified in comparison with the LES analysis.

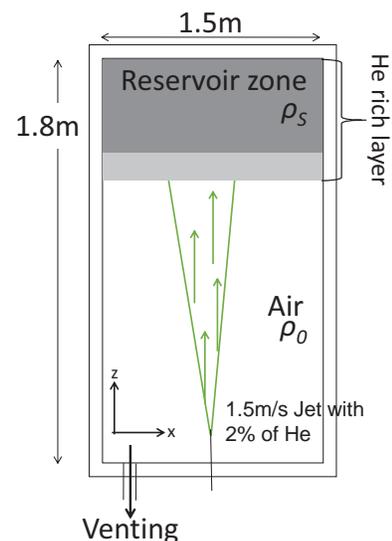


Figure 3 Analyzed condition

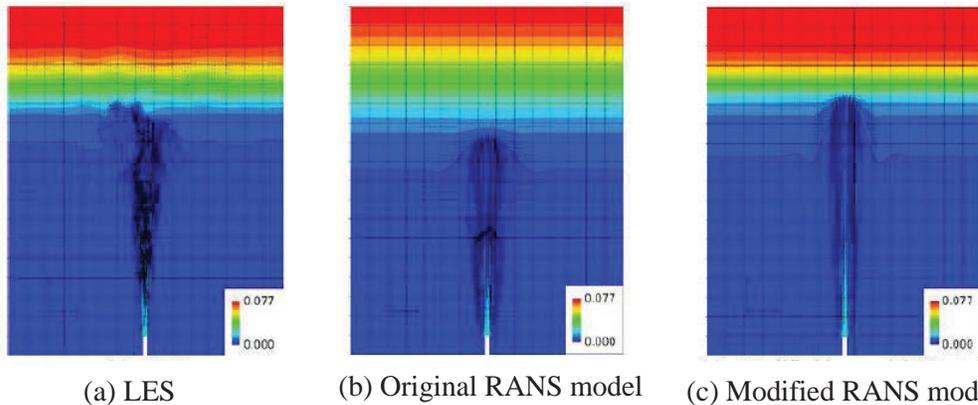


Figure 4 Comparison of three calculated distributions of the helium mass fraction by the LES model, the Low-Re $k-\epsilon$ model, and the modified model at 60 sec. (reprinted from figures in Ref.9 with some changes)

The analyzed geometry was a containment of rectangular parallelepiped with 1.8m of height and 1.5m of length and width as shown in Fig.3, which corresponds to our small-scale experimental facility and is much smaller than the actual containment to suppress the computational time to affordable limit. The initial and boundary conditions were determined in reference to representative previous experiments including the below-mentioned PANDA benchmark exercise; the light gas region having the helium mass fraction of 0.077 (equivalent to 0.4 for molar fraction) was initially located above the air region in the containment. The gas having helium mass fraction of 0.02 was injected through the vertical nozzle attached to the bottom plate, while the gas in the containment was discharged from four holes on the bottom plate modeled as the pressure boundary condition.

The code used for the analysis was OpenFOAM, which is distributed by the OpenFOAM foundation [11]. The used turbulent model was the low-Reynolds number $k-\epsilon$ model for the RANS analysis (referred to as the original RANS model, hereafter); it was the standard Smagorinsky model [12] with near-wall treatment using the Van Driest damping function [13] for the LES analysis. The number of computational cells was $5.4e5$ for RANS and $5e6$ for LES. The comparison results showed that the erosion rate was much larger by the RANS analysis than by the LES analysis. The original RANS model was, therefore, modified to take into account effects of jet stagnation at the impinged region and buoyant force in the mixing region. After investigating several turbulent models, we developed the modified RANS model, which consists of the Kato model [14] for kinetic energy production for flow stagnation and the Katsuki model [15] for turbulence dumping in stratification.

Figure 4 shows the comparison of three calculated distributions of the helium mass fraction by the LES model, the original RANS model, and the modified RANS model, where the helium mass fraction at 60 sec is indicated by a color distribution corresponding to those between 0 and 0.077. The figure clearly indicates that the calculated erosion is almost similar between the modified RANS model and the LES model, while it is much larger for the original RANS model. These differences corresponded to differences in turbulent diffusion behavior; when compared to the LES results regarding intensity and extent, the turbulent diffusion was much larger for the original RANS model, and was similar for the modified RANS model.

In order to compare the modified RANS model with experimental data, we participated in the PANDA benchmark exercise sponsored by the OECD/Nuclear Energy Agency (NEA) and Paul Scherrer Institute (PSI) [16,17]. This benchmark exercise focused on the erosion of density stratification by impinged jet discharged from the bottom, and was participated by 19 organizations using several kinds of CFD codes

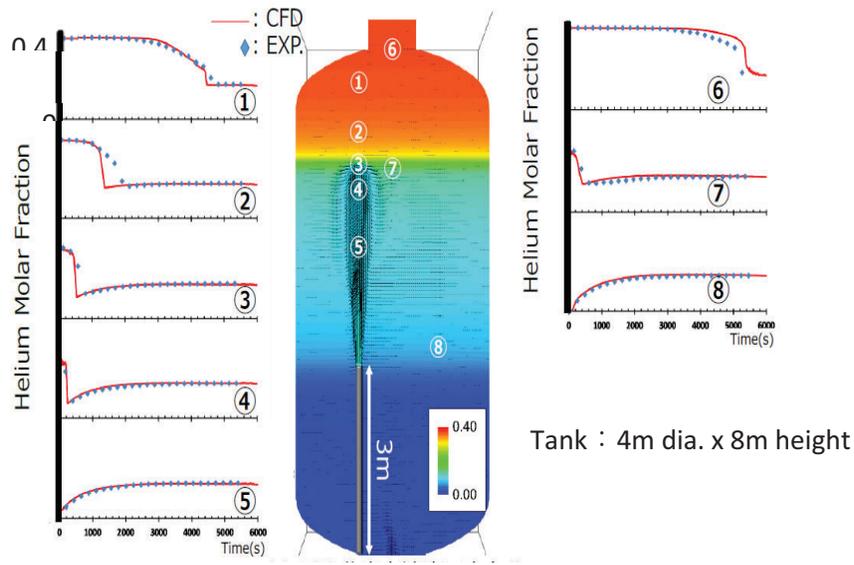


Figure 5 Comparisons of measured and CFD calculated helium molar fractions (reprinted from figures in Ref.16 with some modification)

[18]. Figure 5 shows comparisons of our post-test results using the modified RANS model and the experimental data of the helium molar fractions [16]. The comparison shows fairly good agreement between calculations and experimental data measured at several locations.

Although the good agreement was obtained for the post-test analysis, the pre-test (blind) analysis showed poor results as shown in Fig.6, which indicates unrealistic gradual decrease of helium fraction without showing decrease due to arrival of injected gas with low contained helium concentration. The difference in the calculation condition was only the selection of the computational grid geometry; tetrahedral mesh was used for the blind analysis, and hexahedral mesh was used for the post test analysis. The effect of this difference has been discussed intensively in the community of this research field, although we did not pay sufficient attention before the submission of pretest analysis. Since the use of tetrahedral mesh has several advantages over the hexahedral mesh such as easiness to create the computational grid, we will continue to study the cause of this problem and try to find a suitable numerical scheme to overcome this problem.

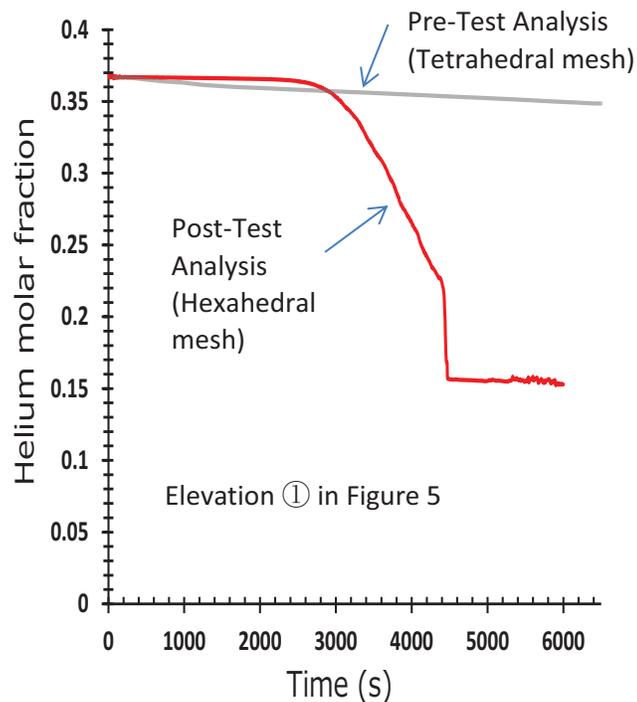
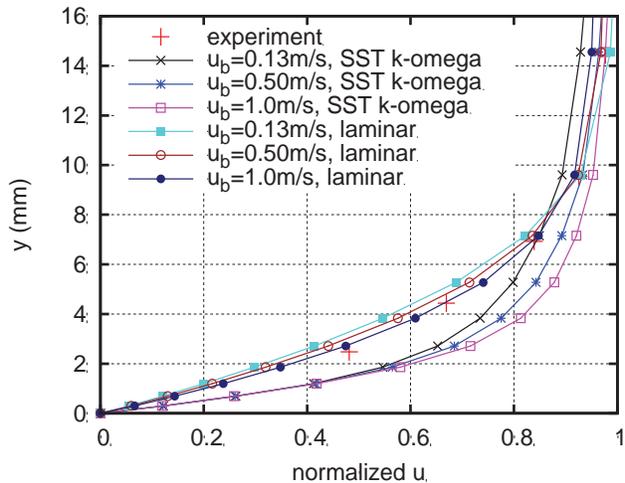
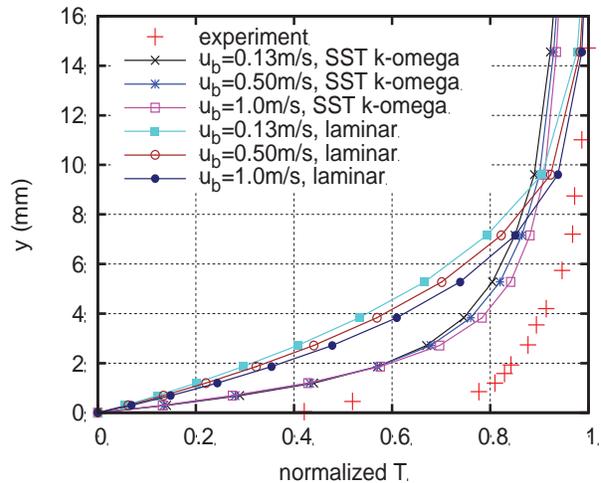


Figure 6 Comparison of pre-test and post-test analyses

Since the use of tetrahedral mesh has several advantages over the hexahedral mesh such as easiness to create the computational grid, we will continue to study the cause of this problem and try to find a suitable numerical scheme to overcome this problem.



Figures 7 Comparison of measured and calculated velocities. (reprinted from a figure in Ref.[19])



Figures 8 Comparison of measured and calculated temperatures. (reprinted from a figure in Ref.[19])

2.3. CFD Analysis on Condensation with Noncondensables

Thermal hydraulic behavior in the containment during severe accidents is affected significantly by condensation of steam on the surface of inner wall, sprayed liquid droplets, internal structure, and heat exchangers such as the fan cooler. In order to clarify technical issues in predicting condensation, experimental data in literature was analyzed using OpenFOAM [19]. The used experimental data were those by H. Kang et al., which investigated condensation on nearly horizontal cooling surface under presence of air at atmospheric pressure [20]. Their study was used because it had detailed measurement on distributions of temperature and velocity near the condensing wall together with their analysis.

Our analysis was based on 1) condensation rate determined by diffusion of steam in the vicinity of the cooling wall, 2) thermodynamic equilibrium, 3) no phase change in the bulk region, 4) no liquid film, i.e., condensation modeled as mass sink on surface, and 5) liquid surface temperature given as a boundary condition which is estimated from the data. The results indicated that the distribution of velocity in the direction parallel to the cooling surface was calculated well by the laminar flow model, while the calculated thickness of the temperature boundary layer was much larger than the experimental data as shown in Figures 7 and 8. Although we tried several models and boundary conditions to have better agreement, all the attempts were failed using the above assumptions. Since the authors of this paper insisted the necessity of assuming thermodynamic nonequilibrium and generation of mists in the boundary layer to predict well the data, equilibrium assumed by our model might be a cause of the disagreement. The measurement error could also be a cause for the disagreement because the temperature measurement in a thin thermal layer is difficult in general. Since several previous papers have indicated the condensation rate can be predicted well assuming thermodynamic equilibrium, this is regarded as an important technical issue for modeling the condensation.

In order to further investigate condensation, we have constructed

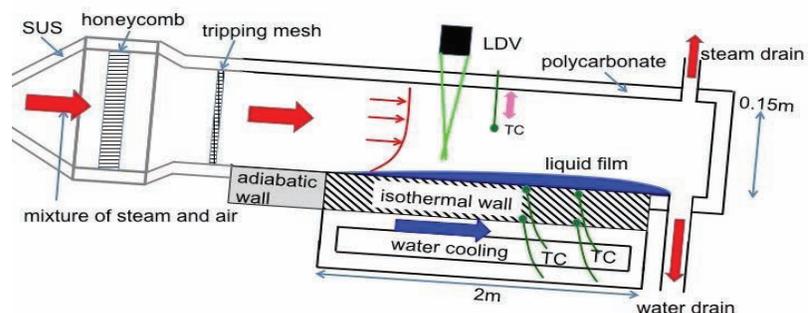


Figure 9 Test section of condensation test facility

an experimental test facility, of which test section is illustrated in Fig.9. The experiment is conducted at atmospheric pressure, changing the wall cooling rate, concentration of steam/air mixture, gas flow rate, and slope of the condensing wall that can be adjustable between horizontal and vertical. The fluid velocity distribution is measured using LDV; the temperature distribution near the wall is measured using a small diameter thermocouple. The preliminary tests were already conducted for the check of instrumentation and equipment.

3. THE ROSA/LSTF EXPERIMENTS

The ROSA/LSTF is a scaling facility simulating the four-loop Westinghouse type PWR which was constructed in 1985 for the investigation on T/H phenomena during the small-break loss-of-coolant accident (LOCA) considering technical issues after the TMI accident. The facility was designed with a volumetric scaling ratio of 1/48 and a height ratio of 1/1 comparing to the reference reactor, Tsuruga-2 operated by the Japan Atomic Power Company (see Fig.10). The ROSA/LSTF facility is the largest PWR simulation facility in the world and has been used for several international and domestic research programs including ROSA-IV, ROSA-V, ISP-26, ROSA-AP600, OECD ROSA, ROSA-PCCS and some programs with domestic vendors and electric utilities.

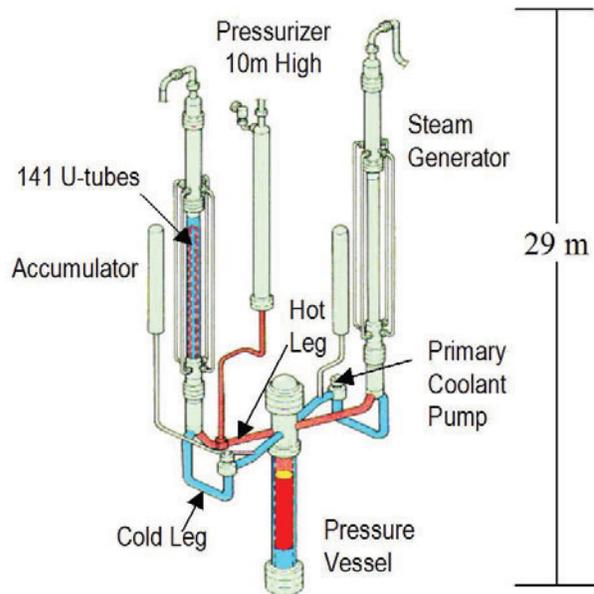


Figure 10 The ROSA/LSTF facility

By using this facility, tests will be conducted to investigate the effectiveness of AM measures for accidents with severe multiple system failures as observed in the Fukushima accident. Currently, we investigate adverse effects of the nitrogen gas ingress from the accumulator injection system (ACC) on the primary depressurization by the intentional SG secondary depressurization as an AM measure [21,22]. The nitrogen gas ingress may occur if the isolation of the ACC system fails after the liquid injection. The resultant accumulation of nitrogen gas in the SG U-tubes adversely affect the heat transfer, which may prevent primary depressurization and thus liquid injection from systems relying on the low-head pump such as a fire engine.

The experiment TR-LF-14 was conducted for this purpose, which assumed the total loss of AC power supply to safety injection systems and the failure of the turbine-driven feedwater system. The AM measures used in this test were manual depressurization of the SG secondary side after observing the core heat-up, and water injection to the SG secondary after the pressure decreased below 1 MPa.

The primary and secondary side pressures observed in TR-LF-14 are shown in Fig.11. The primary pressure stayed at almost the initial value as long as the decay power was transferred to the SG secondary side. The heat transfer to the SG secondary caused cyclic opening of the SG safety valve and steam discharge through the valve. When the SG secondary side became empty of liquid, the decay heat removal through the SGs was terminated, causing the primary pressure to increase and then oscillate responding to cyclic opening of the primary side safety valve. The actuation of the safety valve caused the loss of coolant from the primary loop. When the superheat was observed at the core exit, the SG safety valve was manually opened to depressurize the SG secondary side, and then water was injected to the SG secondary side, which removed heat from the primary and thus depressurized the primary. The figure 11 shows that the primary depressurization rate was accelerated in response to the increase of the

injection rate at ~15000 sec.

When the primary pressure decreased below the initial accumulator pressure of 4.51MPa, the liquid injection started from the ACC system. Because of the assumed failure of the isolation of the ACC system after the coolant injection, the nitrogen gas ingress occurred and accumulated in the SG U-tubes, adversely affecting the heat transfer through the U-tubes. Although the SG secondary side liquid level increased to a level that covered all the U-tubes before 21000 sec, relatively large differential pressure remained between the primary and the secondary. This large pressure difference may prevent the coolant injection from systems relying on a low-head pump. In future experiments, AM measures to mitigate the effects of unintentional ingress of nitrogen gas will also be investigated, which includes opening of the pressure relief valve at pressurizer.

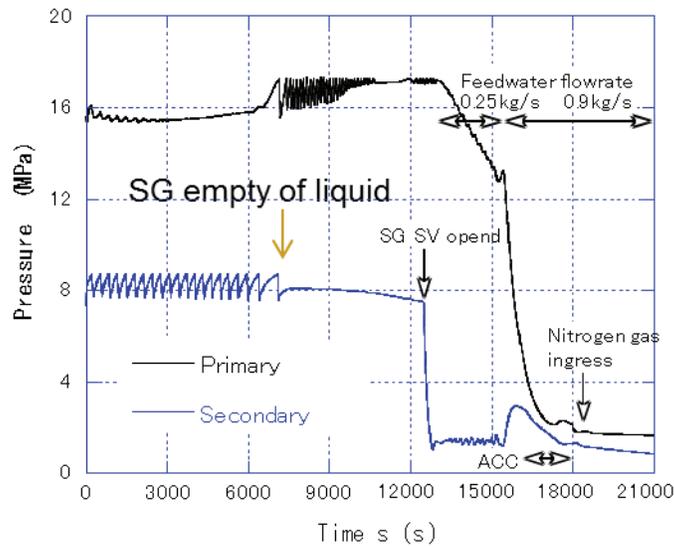


Figure 11 Primary and secondary pressures during TR-LF-14

4. SEPARATE EFFECTS TESTS

Except for the above-mentioned researches, we will conduct various separate effects tests with purposes of the code validation and improvement of the best estimate plus uncertainty (BEPU) method. The recent clarified policy of NRA regarding a possibility of the development of the system analysis code also emphasizes the importance of those researches.

Planned researches for this purpose include heat transfer tests and hydraulic behavior tests focusing on the scaling effects and uncertainty. The core heat transfer tests will be conducted using a single tube heat transfer test facility and a 4x4 bundle test facility. The test section of the single tube heat transfer test facility consists of two vertically connected tube sections: test section and preheating section, which can be heated independently (see Fig.12). This configuration enables to independently control quality and wall temperature in the test section, and is beneficial in measuring droplet deposition rate with a heat balance method and investigating parameter effects of wall temperature, flow rate and quality under both steady and transient condition [23]. The bundle experiments will focus, among others, on the blowdown heat transfer considering its large effect on the uncertainties of PCT.

For the experiments focusing on the scaling effects, the

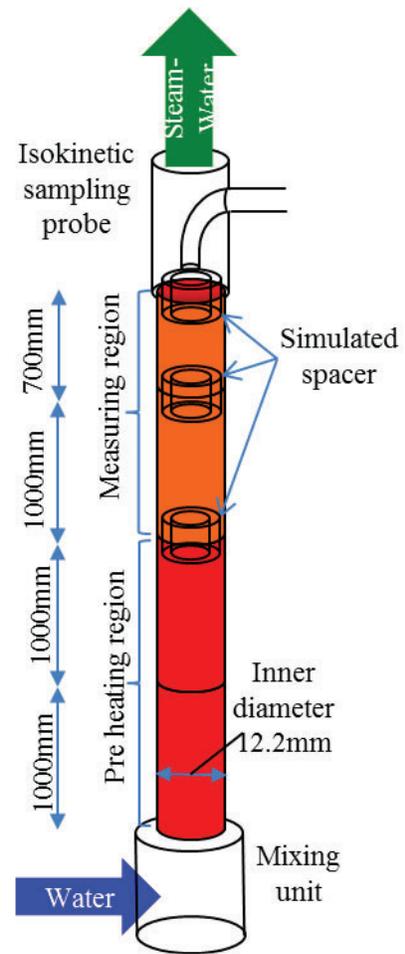


Figure 12 Single-tube heat transfer test facility

countercurrent flow limitation (CCFL) in the hot leg including the inclined portion at the SG inlet will be investigated by conducting tests under a wide range of pressure conditions and using fluids of air-water or steam-water. To effectively conduct experiments, the high pressure thermal-hydraulics test loop will be constructed.

5. CONCLUSIONS

The Fukushima Dai-Ichi Nuclear Power Station accident reemphasized importance of the research on the severe accident in Japan. The thermal-hydraulic safety research group in JAEA, therefore, started the ROSA-SA project in 2013 on containment thermal hydraulics related to overtemperature damage, hydrogen risk and FP transport. For this project, the large-scale test facility called CIGMA has been designed and constructed, which is characterized by the capability of conducting high-temperature and pressure experiment with CFD-grade instrumentation of high-resolution. The CIGMA test, therefore, can enlarge the T/H conditions of existing databases, which is considered important because even the state-of-the-art CFD codes/models rely on empirical relationships for modeling of turbulence, turbulent-induced scalar diffusion and so on. The experiments will focus on T/H phenomena and effects including the erosion of density stratification due to impinging jet or plume, wall temperature response to high-temperature impinging jet, cooling on the containment outer surface or by internal spray, and internal structures such as bulk head and wall. The CFD analyses were also conducted to identify the technical issues and validate and modify the turbulent model using the experimental data in literature and those obtained in the OECD/NEA sponsored benchmark exercise. The first CIGMA test is schedule in 2015.

Also by considering the lessons learnt from the accident, the investigation was initiated for AM measures against accidents under more severe multiple-failure conditions using the ROSA/LSTF facility. Currently, we are focusing on the nitrogen gas ingress from the accumulator to the primary loop. Such unintentional ingress causes degradation of heat transfer through the SG and thus adversely affects the primary depressurization by the intentional SG secondary depressurization which is one of important AM measures. Insufficient primary depressurization may adversely affect the performance of coolant injection systems relying on a low-head pump such as a fire engine.

Furthermore, we will conduct separate effects tests for the validation of the code/models and the improvement of the BEPU method. Planned researches for this purpose include core heat transfer tests and hydraulic behavior tests focusing on the scaling effects. The heat transfer test in a single tube will be conducted this year.

ACKNOWLEDGMENTS

The CIGMA facility and the separate effects tests facilities including the single-tube heat transfer test facility and the bundle test facility are developed under the auspices of the Nuclear Regulation Authority (NRA), Japan. One figure in this paper contains the experimental data obtained in the third OECD/NEA CFD benchmark Exercise (IBE-3) which are conducted by the Paul Scherrer Institute (PSI) using the PANDA facility. The authors appreciate the PSI staffs for performing the PANDA experiment. The authors also appreciate Dr. E. Studer of the French Alternative Energies and Atomic Energy Commission (CEA), Saclay for his helpful comments on the computational mesh for the CFD analysis.

REFERENCES

1. “Genshiryoku kisei iinnkai niokeru annzen kenkyuu ni tsuite,” (translated as “On the safety study by the nuclear regulation authority,”), Nuclear regulation authority, Japan, *in Japanese* (2013).

2. Y. Sibamoto, et al., "Research Program of Flow Behavior in Containment Vessel for Severe Accident," *2013 Annual Meeting of Atomic Energy Society of Japan*, N41, in Japanese (2013).
3. D. Paladino and J. Dreier, "PANDA: A Multipurpose Integral Test Facility for LWR Safety Investigations," *Science and Technology of Nuclear Installations*, ID:239319 (2012).
4. R. Zboray and D. Paladino, "Experiments on Basic Thermal Hydraulic Phenomena Relevant for LWR Containments: Gas mixing and Transport induced by Buoyant Jets in a Multi-compartment Geometry," *Nucl. Eng. Des.*, **240**, pp. 3158-3169 (2010).
5. M. Caron-Charles, et al., "Steam Condensation experiments by the MISTRA Facility for field containment code validation," ICONE-10-22661 (2002).
6. "OECD/NEA THAI Project, Final Report Hydrogen and Fission Product Issues relevant for Containment Safety Assessment under Sever Accident Conditions," NEA/CSNI/R(2010)3 (2010).
7. "OECD/SETH-2 Project PANDA and MISTRA Experiments Final Summary Report," NEA/CSNI/R(2012)5 (2012).
8. H. J. Allelein et al., "International Standard Problem ISP-47 on Containment Thermal Hydraulics Final Report," NEA/CSNI/R(2007)10 (2007).
9. S. Abe et al., "RANS and LES Analyses on a Density Stratified Layer Behavior of Multicomponent Gas by Buoyant Jet in a Small Vessel," *Proceedings of the International Conference on Nuclear Engineering ICONE-22*, July 7-11, 2014, Prague, Czech Republic (2014).
10. S. Abe et al., "A study on improvement of RANS analysis for erosion of density stratified layer of multicomponent gas by buoyant jet in a containment vessel," *Proceeding of 10th International Topical Meeting on Nuclear Thermal Hydraulics, NUTHOS-10*, NUTHOS10-1181, Okinawa, Japan, Dec. 15-19 (2014), also will be published in *Journal of Energy and Power Engineering*.
11. "The OpenFOAM Foundation," <http://www.openfoam.org/index.php> (2015).
12. J. Smagorinsky, "General circulation experiments with the primitive equations", *Monthly Weather Review*, **91** (3), pp. 99-164 (1963).
13. E. R. Van Driest, "On Turbulent Flow near a Wall," *Journal of Aeronautical Sciences*, **23**(11), pp. 1007-1011 (1956).
14. M. Kato and B. E. Launder, "The Modeling of Turbulent Flow around Stationary and Vibrating Square," *9th Symposium on turbulent shear flows* (1993).
15. T. Katsuki, et al., "Wind Tunnel Experiment and Numerical Simulation of Atmospheric Boundary Layer under Various Atmospheric Stability," *Journal of Environmental Engineering, Architecture Institute of Japan*, **74**, pp. 735-743, in Japanese (2009).
16. S. Abe et. al., "JAEA approach OECD/NEA PANDA Benchmark, Erosion of a stratified layer by a buoyant jet in a large volume," *presented in the poster session at the CFD4NRS-5, OECD/NEA & IAEA workshop* (2014).
17. S. Abe et. al., "RANS analyses on erosion behavior of density stratification consisted of helium-air mixture gas by a low momentum vertical buoyant jet in the PANDA test facility, the third International Benchmark exercise (IBE-3)," submitted to *Nucl. Eng. Des.*
18. M. Andreani, A. Badillo and R. Kapulla, "Synthesis of the OECD/NEA-PSI CFD Benchmark Exercise," *CFD4NRS-5, OECD/NEA & IAEA workshop* (2014).
19. M. Ishigaki, et. al., "Numerical simulation of thermal flow with steam condensation on wall using the OpenFOAM code," *Proceedings. of CFD4NRS-5, OECD/NEA & IAEA workshop, CFD4NRS-5* (2014).
20. H.C. Kang and M.H. Kim, "Characteristics of film condensation of supersaturated steam-air mixture on a flat plate," *Int. J. Multiphase Flow*, **25**(8), pp.1601-1618 (1999).
21. T. Takeda et al., "ROSA/LSTF Experiment on a PWR Station Blackout Transient with AM Measures and RELAP5 Post-test Analysis, ROSA/LSTF," ICONE22-30205 (2014).
22. T. Takeda and I. Ohtsu, "ROSA/LSTF Experiment on AM Measures during A PWR Station Blackout Transient with Pump Seal Leakage and RELAP5 Post-test Analysis," submitted to ICONE-23 (2015).

23. Y. Sibamoto et al., "Core Heat Transfer Coefficient Immediately Downstream of the Rewetting Front during Anticipated Operational Occurrences for BWRs," *J. Nucl. Sci. Technol.*, **48**(3), pp. 440-453 (2011).