SEVERE ACCIDENT RESEARCH IN JAPAN AFTER ACCIDENT AT FUKUSHIMA DAIICHI NPS

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ABSTRACT

After the Fukushima Daiichi accident in March 2011 several investigation committees in Japan issued reports with lessons learned from the accident, in which some recommendations on severe accident research are included. The review of specific severe accident research issues was started after Fukushima accident in Atomic Energy Society of Japan (AESJ). AESJ has recently developed a new Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap (TH-RM) for LWR safety improvement and development after Fukushima accident by thoroughly revising the 1st version (TH-RM-1) prepared in 2009. The revision has been made by the work of three sub working groups (SWGs) by considering the lessons learned from Fukushima accident. The TH-RM has identified twelve important thermal-hydraulic technology development subjects, 8 of which are severe accident related subjects. At the same time Research Expert Committee on Evaluation of Severe Accident established in AESJ in 2012, has published Phenomena Identification and Ranking Tables (PIRTs) for both thermal-hydraulic and source term issues in severe accident based on findings from Fukushima accident by utilizing PIRT methodologies. The present paper describes the review of the severe accident research before Fukushima accident, lessons learned on severe accident research from Fukushima accident, severe accident research issues reviewed after Fukushima accident in AESJ and finally current severe accident research activities mostly based on this review after Fukushima accident in Japan.

KEYWORDS
Severe accident; Fukushima Daiichi; Lessons learned; R&D issues; Roadmap; PIRT

1. INTRODUCTION

Severe accident research in Japan was started after TMI-2 accident in 1979 with small-scale experiments and analysis and it was accelerated after Chernobyl accident in 1986 with relatively large-scale experiment and analysis until around 2003. In response to “Recommendation of accident management for severe accident of light water nuclear power plant” by Nuclear Safety Committee of Japan in 1992, basic policy of accident management measures due to internal events was voluntarily proposed by the electric power companies and they were approved by the Government in 1994. After the deployment of accident management measures at nuclear power plants until around 2002, severe accident research was drastically reduced in terms of number of experts and budget. Also severe accident due to external events has not been paid careful attention for the accident management measures, even though the knowledge and methodologies on severe accident initiated by the external events have been accumulated in reactor safety society in Japan in recent years.

After the accident at Fukushima Daiichi Nuclear Power Station (Fukushima accident) in 2011 several investigation committees have been established in Japan, such as those by the Government, Diet and private sectors including Tokyo Electric Power Company, and Atomic Energy Society of Japan (AESJ). They have issued investigation reports with lessons learned from the accident [1-5]. Several measures, such as enhanced emergency power supply capabilities and improved severe accident management, have already been in place and some mid/long term measures are being implemented at nuclear power plant sites in accordance with new regulatory standards established by Nuclear Regulatory Authority of Japan. Among those lessons, several recommendations have been made on severe accident research. Specific severe accident research items have also been reviewed in AESJ in some committees and working group [6-8]. The present paper describes the review of the severe accident research before Fukushima accident, lessons learned on severe accident research from Fukushima accident, accident research issues reviewed after Fukushima accident in AESJ and finally current severe accident research activities after Fukushima accident in Japan. [9-10].

2. REVIEW OF SEVERE ACCIDENT RESEARCH IN JAPAN BEFORE FUKUSHIMA ACCIDENT

2.1 Major Severe Accident Phenomena

The major phenomena during severe accidents of LWRs are core damage progress in reactor pressure vessel (RPV), molten core cooling, direct containment heating (DCH), fuel coolant interaction (FCI) including steam explosion, molten core/concrete interaction (MCCI), hydrogen deflagration and detonation, RPV failure, containment vessel (CV) failure, fission product (FP) release from fuel, and FP transport in reactor coolant system (RCS) and CV [11]. Severe accident phenomena in LWRs are generally characterized by their physically and chemically complex processes involved with high temperature core melt, multi-component and multi-phase flows, transport of radioactive materials and sometimes highly non-equilibrium state [10]. Review of thermal-hydraulic researches in severe accidents in LWRs has already been published by Kataoka mostly based on the papers appeared in Journal of Nuclear Science and Technology [12]. Review of severe accident research before Fukushima accident in Japan is briefly described in the following sections.

2.2 Fuel Degradation

Right after the TMI accident there was very few information on fuel degradation behaviors. Therefore experimental programs using research reactors, such as PBF (INEL, USA), ACRR (SNL, USA), NRU (CRNL, Canada), and PHEBUS (CEA, France) were initiated. In Germany electrically heated rod bundle fuel degradation experiment called CORA was conducted. Also investigation of TMI-2 degraded core was started as TMI-2 R&D program by USNRC.

In Japan fuel rod degradation experiment was conducted using Nuclear Safety Research Reactor (NSRR) and specifically fuel degradation behaviour at the reflooding of the core was investigated at Japan Atomic Energy Agency (JAEA) [13]. The oxidation behavior of the Zircalloy cladding tube was clarified at temperatures ranging 1000–1260°C, and it was shown that there was fuel degradation due to the thermal shock by the reflooding after the cladding was exposed to high-temperature steam for a relatively long time.

2.3 In-Vessel Phenomena

OECD TMI VIP program [14] was conducted in order to investigate accident progression in the lower head in TMI-2 and the heat load onto the RV lower head was evaluated. It was found that the existing cooling mechanism could not explain the observed effective cooling of the lower head. Therefore several experimental and analytical studies have been conducted to investigate the molten core coolability in the lower head. In-vessel retention (IVR) has also been investigated experimentally and analytically.

In Japan, for example, JAEA conducted in-vessel molten core coolability experiment in ALPHA program using molten alumina as a melt simulant pouring on to lower head geometry [15] as shown in Fig. 1. It was shown that the gap about 1 mm width between solidified alumina layer and vessel wall was formed and that this supports the hypothetical gap flooding cooling mechanism [16]. Experimental studies supposing in relation to IVR are made to investigate the characteristics of pool boiling heat transfer from downward heating surface and obstacle effects of penetrations at Osaka University [17].

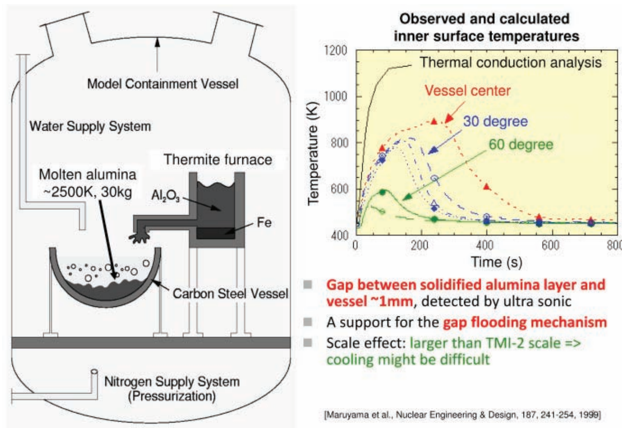

Figure 1. In-Vessel Molten Core Coolability Experiment [15].

2.4 Ex-Vessel Phenomena

In-vessel or ex-vessel large-scale fuel coolant interaction (FCI), or steam explosion, by a contact of high temperature core melt and low temperature water could be a threat to the containment integrity due to its pressure loads, shock wave and missile attack. Many experimental and analytical studies have been conducted for FCI. Large-scale experiments using corium have been conducted with KROTOS and FARO at JRC Ispra, and TROI at KAERI.

In Japan with ALPHA experiments at JAEA, several parameters, such as ambient pressure, amount of molten material and water temperature, on the occurrence of the steam explosion have been investigated using up to 50 kg of simulated molten materials [18]. The interfacial behavior between high temperature molten liquid and low temperature water is experimentally investigated by using a molten material droplet and external pressure pulse at the University of Tsukuba [19].

Molten core-concrete interaction (MCCI) could be another threat to the containment integrity. Large amounts of molten corium may enter the reactor cavity after the reactor pressure vessel failure. One of the important MCCI related issue is the coolability of the melt by injected water as one of accident measures to terminate MCCI progress. In MACE [20] and OECD/NEA MCCI Projects [21], large-scale MCCI and coolability experiments have been conducted using molten corium with electrical heating. Analytical works have also been conducted for the prediction of MCCI behaviors.

In Japan the analysis for the WITCH/LINER experiments was performed to investigate the heat transfer characteristics between the gas-agitated steel melt and the vertical surface by JAEA [22]. The applicability of heat transfer correlations for a gas-agitated fluid system was examined through the numerical analysis of the one-dimensional heat conduction taking into account the crust formation due to the solidification of the steel melt. The FCI and MCCI have been studied experimentally in COTELS project as a joint study between Nuclear Power Engineering Corporation (NUPEC, Japan) and National

Nuclear Center, NNC (Republic of Kazakhstan) using one of the testing complexes at NNC [23]. The testing complex includes three experimental facilities for debris coolability tests. To get the molten corium, the electric induction melting furnace was used to produce 60 kg of corium containing UO_2, SS, Zr, and ZrO_2.

There has been increased interest in hydrogen production, distribution and combustion in LWRs since TMI accident. There is major concern as these events threaten containment integrity. Within the total range of possible premixed combustion events, namely ignition, flame propagation, deflagration, deflagration-to-detonation transition (DDT) and detonation, where particular uncertainty exists for the DDT. There have been many experimental and analytical researches on these hydrogen issues, including the effectiveness of hydrogen control systems, such as ignitors and passive recombiners.

In Japan NUPEC carried out large-scale hydrogen mixing and distribution tests to investigate hydrogen distribution behavior in model containment [24]. One of the tests have been utilizes as international standard problem (ISP-35) of OECD.

For the probable reason, containment integrity may be lost due to over-pressure or over-temperature at the seals of penetrations, airlocks or deformation of containment structure during severe accident. Several containment integrity tests were conducted at SNL at over-pressure conditions for both steel and concrete type containment.

In Japan, experiments to investigate the containment integrity have been conducted at NUPEC for the structural behavior using scaled containment model [25]. JAERI conducted penetration leakage characterization test , simulating relevant parts of electrical penetration assembly used in Japanese PWR containments [26].

2.5 Fission Product Release and Transport

For the fission product release from the fuel during severe accident, experimental studies have been conducted at ORNL and IRSN, and database of the experimental parameters on the FP release has been accumulated.

In Japan VEGA experiment at JAEA, shown in Fig. 2, has been conducted to investigate the effects of fuel temperature, ambient pressure up to 10 bar, atmospheric condition and MOX fuel on FP release [27]. It was confirmed that the release of CsI is obviously suppressed by higher ambient pressure, which gave the first experimental evidence.

For the FP transport in RCS and CV experimental and analytical studies have been conducted. Integral experiments from the FP release from the fuel and FP transports in RCS and CV has been conducted in PHEBUS-FP program at IRSN Cadarache, France.

In Japan, JAEA conducted high pressure pool scrubbing experiment up to 6 MPa simulating pool scrubbing in pressurizer of PWRs [28]. Industries (Toshiba, Hitachi and TEPCO) conducted pool scrubbing experiment with a cylindrical pressure vessel, 1 meter in diameter and 5 meters high. The aerosol removal efficiencies were systematically measured and a simplified model for aerosol removal effects by pool scrubbing has been developed [29]. WIND project has been conducted at JAEA to investigate the re-suspension and re-evaporation of FP aerosol through the primary pipes, including steam generator tubes [30]. Also iodine chemistry in containment has been investigated at JAEA under high radiation dose using Co-60 [31].

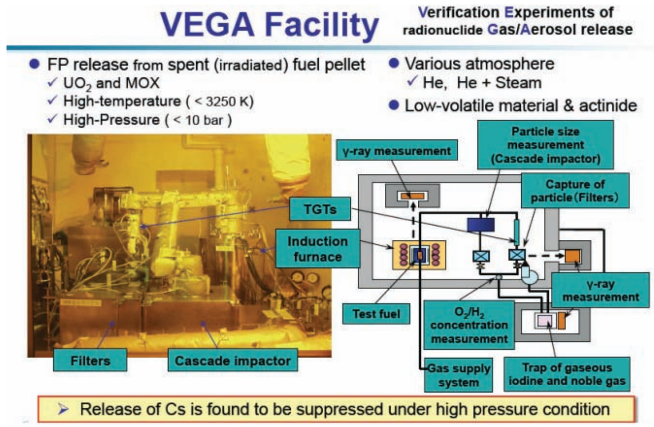

Figure 2. VEGA Experiment on FP Release from Irradiated Fuel [27].

2.6 Severe Accident Simulation Codes

Integral codes simulate the overall nuclear power plant response, that is, the response of RCS, CV and FP release and transport, and finally source term released to the environment, by using integral models for a self-consistent analysis of the accident. They include a combination of experimental correlations and phenomenological models for the relevant phenomena. In Japan THALES [32] code has been developed by JAEA in this category. THALES code is an integrated severe accident analysis code in order to simulate the accident progression and transport of radioactive material for probabilistic safety assessment (PSA) of a nuclear power plant.

Mechanistic (or detailed) codes are characterized by best-estimate phenomenological models to enable an accurate simulation of severe accident behaviour. The main advantages of these coded are to give detailed insight into the progress of severe accident to design and optimize mitigation measures. These codes can be used for the benchmarking of the integral codes. In Japan, SAMPSON code [33] has been developed by Institute of Applied Energy (IAE) in order to pursue the most detailed mechanistic code in this category.

Specific (or dedicated) codes aim at simulating a single phenomenon. These codes may be simple with fast-running or very complex with large calculation time, depending on their objectives. In Japan JASMINE code [34] has been developed at JAEA for steam explosion and melt dispersal.

2.7 Summary of Severe Accident Research in Japan before Fukushima Accident

In Japan relatively wide variety of severe accident research was conducted among institutes, industries and universities, especially after the Chernobyl accident. In mid 1990s severe accident measures, initiated by internal events, were voluntarily proposed by the utilities based on the encouragement by the Government and some of the research results worldwide, including those of Japan, were reflected to the implementation of severe accident measures. Since the assessment of severe accident initiated by external events was technically difficult at that time, it was agreed that it will be evaluated in the future. However after the above severe accident measures were actually deployed to the nuclear power plant sites in early 2000s, the needs on severe accident research was drastically reduced partly by some misunderstandings, and associated human resources and budget were sharply reduced in Japan. This trend was accelerated by the safety check of the nuclear power plants after the earthquake near Kashiwazaki NPS in 2007 and continued just before the Fukushima accident.

3. SEVERE ACCIDENT RELATED LESSONS LEARNED FROM FUKUSHIMA ACCIDENT

After the Fukushima accident several investigation committees have been established and the following reports have been issued in Japan:
(1) Independent Investigation Commission, Feb 2012 (Non-government Committee) [1],
(2) Tokyo Electric Power Company, June 2012 (TEPCO Committee) [2],
(3) National Diet of Japan Investigation Commission, July 2012 (Diet Committee) [3],
(4) Investigation Committee on the Accident, July 2012 (Government Committee) [4], and
(5) Atomic Energy Society of Japan, March 2014 (AESJ) [5].

In some investigation committee reports, safety research, especially severe accident related research, based on lessons learned from the Fukushima accident has been recommended. For example, in AESJ reports, in recommendation Ⅳ(Common items), item (1) entitled "(1) Enhancement of nuclear safety research base" says "In order to assure safety, it is important to clarify the fundamental concept of the nuclear safety, to set safety goals, to effectively contribute to safety improvement based on PRA, and to continuously pursue plant designs, severe accident measures and emergency measures, which properly apply the defense-in-depth concept. Nuclear safety research constitutes the base of the continuous efforts for these safeties." Under this item the report also recommends "Nuclear safety research should be a driving force to promote a better understanding of the overview of the safety approach and to pursue continuous advancement of both software and hardware for the diversified safety improvement." and lastly it recommends "to prepare a comprehensive map of technical issues to be addressed by the discussions on the vision for the achievement of safety goals and by facing the current technology. For the resolution of these technical issues it is highly recommended to develop mid and long-term roadmap, as well as short-term roadmap."

It should be mentioned that under the same recommendation Ⅳ(Common items), "(2) Enhancement of international collaboration" and "(3) Nuclear human resource development" are also recommended, since both activities are closely related with research activities, especially with severe accident research.

4. REVIEW OF SEVERE ACCIDENT RESEARCH ISSUES IN AESJ

4.1 Thermal-hydraulic Roadmap

The Atomic Energy Society of Japan (AESJ) developed a new Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap (TH-RM) for LWR safety improvement and development after Fukushima accident by thoroughly revising the 1st version (TH-RM-1) prepared in 2009 under good collaboration of utilities, vendors, universities, research institutes and technical support organizations for regulatory body. This revision has been made by three sub working groups (SWGs), namely "safety assessment", "fundamental technology" and "severe accident", by considering the lessons learned from Fukushima accident [8]. The "safety assessment" SWG pursued the development of computer codes mostly for safety assessment. The "fundamental technology" SWG pursued safety improvement and risk reduction via accident management measures by referring the technical map for severe accident established by "severe accident" SWG. Phenomena and components for counter-measures and/or proper prediction are identified by going through severe accident progression in both reactor and spent-fuel pool of PWR and BWR. Twelve important thermal-hydraulic technology development subjects have been finally identified. Eight subjects out of twelve are severe accident related as below:

(1) Development of core catcher (ex. Material database of refractory material/ Performance evaluation and verification)

(2) Verification of drywell air cooling system performance under severe accident condition (ex. Non-condensable gas effect)
(3) Passive containment cooling system (PCCS)
(4) Coolant injection to reactor well cavity for containment top-head flange cooling and development of sealing material with high heat-resistance
(5) Autocatalytic recombiner (PAR)
(6) Hydrogen removal system for severe accident convergence termination (ex. Catalysis for H_2 removal in severe accident late-phase condition)
(7) Filtered venting system (+Experimental data for FP transport, Iodine behavior, deposit, vaporization)
(8) Development of instrumentation and measurement devices for severe accident conditions

It is noted that Work Description Sheet was developed for each of identified and selected R&D subjects. External hazards are also considered how to cope with from thermal-hydraulic safety point of view.

4.2 Thermal-hydraulic and Source Term PIRTs

Research Expert Committee on Evaluation of Severe Accident established in AESJ in 2012, in collaboration with the above mentioned SWGs, started to investigate severe accident related issues for the estimation of the melted core status in the Fukushima-Daiichi units and for the improvement of source term estimation considering Fukushima accident. In this Committee important specific severe accident phenomena had firstly been extracted based on the knowledge of severe accident researches, and the key phenomena and/or the phenomena with large uncertainties had secondly been selected through the brainstorming and discussion among experts. Phenomena Identification Ranking Table (PIRT) was finally obtained and reports have been issued for both thermal-hydraulic [6] and source term aspects [7].

By reviewing the current MAAP evaluation models with the obtained thermal-hydraulic PIRT, it was found that 95 of the 386 high-ranked specific phenomena were not considered in MAAP 5.0.1. While 62 of these phenomena will have been addressed in the MAAP enhancement project and 25 others are not suitable to be analyzed by MAAP, 8 important phenomena should be considered in post-MAAP enhancement project with additional experiments or fundamental studies.

The source term PIRT is divided into 3 phases for time domain and 9 categories for spatial domain. The 68 specific phenomena have been extracted and the importance from the viewpoint of the source term has been ranked through brainstorming and discussions among experts.

5. SEVERE ACCIDENT RESEARCH AFTER FUKUSHIMA ACCIDENT

As discussed in Chapter 4, specific important severe accident research issues have been identified after the Fukushima accident mostly based on the review work in AESJ. In this section several examples of ongoing severe accident research activities, which are basically in accordance with the important issues identified in Chapter 4, are introduced:

5.1 Severe Accident Research at Institutes

At JAEA, new researches have been initiated after the Fukushima accident, which are related to containment thermal hydraulics and accident management measures for the prevention of core damage under severe multiple failure conditions [35]. Those experimental studies are to obtain better understandings on the phenomena and establish databases for the validation of both lumped parameter and 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 transport. For this project, a large-scale containment vessel test facility called CIGMA

(Containment InteGral Measurement Apparatus)**,** shown in Fig. 3, has been designed for the conducting high-temperature experiments as well as those on hydrogen risk with CFD-grade instrumentation of high space resolution.

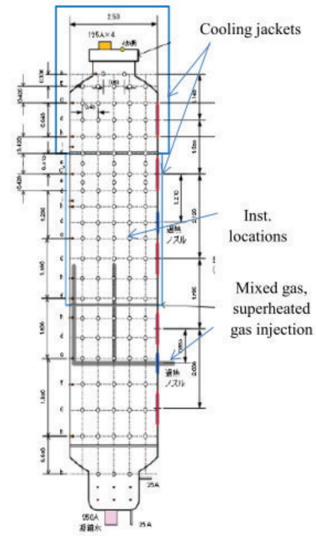

Figure 3. CIGMA Test Facility at JAEA.

An integral code system for severe accident analysis, THALES2/KICHE, shown in Fig. 4, has also been developed at JAEA. The core melt progression and the transportation of radioactive materials within RCS and CV are analyzed with THALES2 in conjunction with KICHE for the iodine reaction kinetics in aqueous phase. The applications of THALES2/KICHE have been made in various analytical studies for severe accident progression, including analyses for the Fukushima accident in order to obtain technical knowledge on the source term into the environment and the core damage state [36].

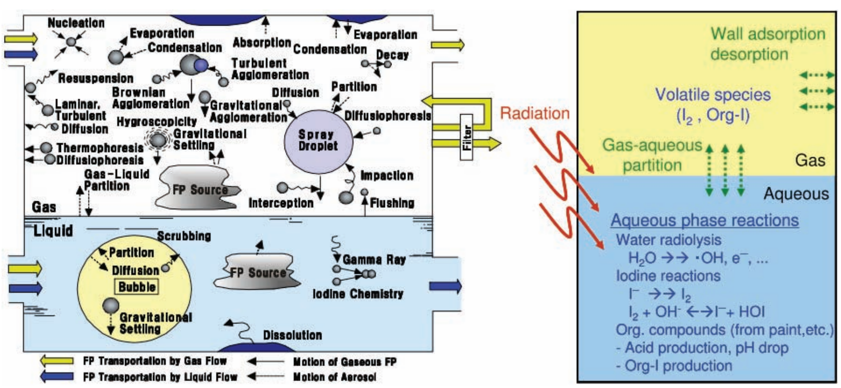

Figure 4. THALES-2 (left) and KICHE (right).

It is noted that OECD/NEA's BSAF (Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station) Project hosted by JAEA has been conducted among 8 countries from November 2012 using currently available severe accident analysis integral codes in order to improve severe accident codes and analyze accident progression and current core status in detail for preparation of fuel debris removal, as a part of the R&D projects for the mid-to-long term response for decommissioning of the Fukushima Daiichi Nuclear Power Station, units 1 through 4.

At Institute of Applied Energy (IAE), SAMPSON code development for detailed severe accident analysis has been accelerated after Fukushima accident [37]. SAMPSON is designed as a large scale simulation system of inter-connected hierarchical modules, as shown in Fig. 5. It intends to minimize the use of empirical correlations, to eliminate the tuning parameters as much as possible and to maximize the use of mechanistic models and theoretical-base equations. Also SAMPSON is capable of detailed multi-dimensional plant analysis. Recently validation and verification activities have been greatly progressed due to the participation of industries and universities.

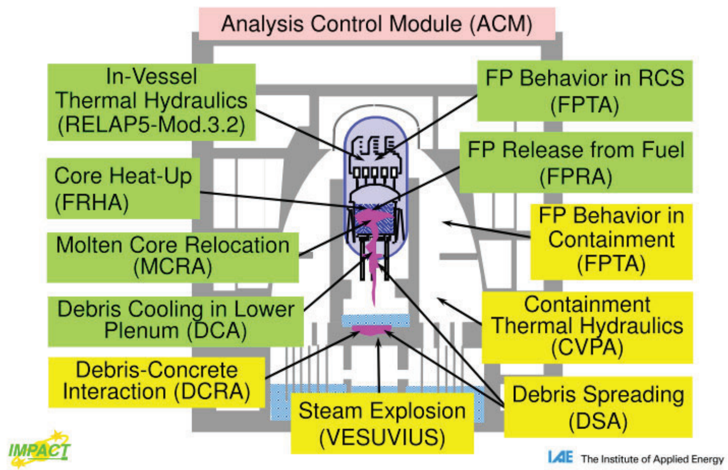

Figure 5. SAMPSON : Severe Accident Analysis Code with Modular Structure.

5.2 Severe Accident Research at Industries

TEPCO has investigated the accident progression behaviors and source terms of Fukushima accident using the measured data at the accident and analytical codes, such as MAAP code. Recently TEPCO has issued latest report on the findings of their continuing inquiry into some of the important technical remaining questions on the Fukushima accident. Specifically TEPCO investigated the accident progression of Fukushima Daiichi Unit 1 (1F1) based on the behavior of fuel range water level indicator readings. They have found that the accident progression of 1F1 was estimated based on the measured values and the knowledge obtained by existing accident analyses, and that the simulation based on the estimated accident progression approximately reproduced the entire trend of the measured data [38].

At Hitachi, the development of inherently safe technologies for large scale BWRs has been conducted. They include a passive water-cooling system, infinite-time air-cooling system and hydrogen explosion prevention system and an operation support system [39]. The passive water-cooling system and infinite-time air-cooling system achieve core cooling without electricity. These systems are intended to cope with a long-term station black out. Both these cooling systems remove relatively high decay heat for the initial 10 days after an accident, and then the infinite-time air-cooling system is used alone to remove attenuated decay heat after 10 days. The hydrogen explosion prevention system consists of a high-temperature resistant fuel cladding made of SiC cladding and a passive autocatalytic recombiner.

At Toshiba in collaboration with PWR utilities and IAE, testing plan for critical heat flux measurement during in-vessel retention has been conducted in order to develop a CHF correlation for various PWRs [40]. The ranges of the test parameters and test method to simulate local conditions in the cooling channel around RV were developed, and the test equipment was designed. Development on flat and high thermal conductivity core-catcher is promoted at Toshiba considering installing into operating plants [41]. The flow pattern visualization test results have been conducted. With the inclination angle 0 degree conditions, the flow patterns of air and water in the test section become stratified flow. With the inclination angle 5 degree conditions, the flow patterns of air and water in the test section become slug flow.

5.3 Severe Accident Research at Universities

In order to clarify the effects of complicated structures in BWR lower plenum on the jet behavior, visualization experiments have been conducted at The University of Tsukuba [42] in collaboration with JAEA. It was clarified that the jet tip velocity depends on the condition whether complicated structures exist or not and also clarified that the structures prevent the core of the jet from expanding. Steam injector, a passive jet pump which operates without power source or rotating machinery with high heat transfer performance due to the direct-contact condensation of supersonic steam flow onto subcooled water jet, has been developed as a safety system during severe accident as shown in Fig. 6 [43]. Pool scrubbing and filtered venting during severe accident conditions have also been investigated with basic experimental facilities [44].

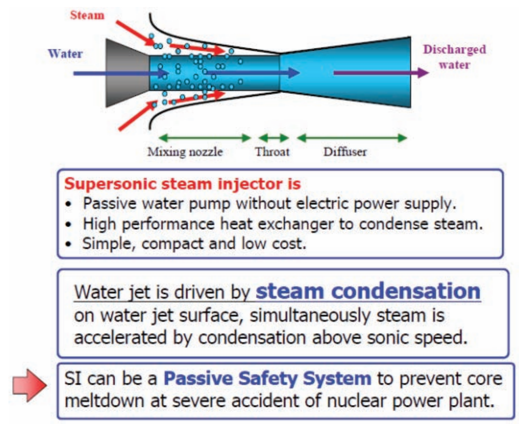

Figure 6. Concept of Supersonic Steam Injector [43].

At The University of Tokyo, in order to understand the rapid depressurization in Unit 1 of Fukushima accident, experimental investigation on the buckling failure of stainless steel tube columns under external pressure has been conducted [45]. The results show different behaviors of the buckling failure temperature depending on the buckling mode. For the first mode of buckling, the buckling failure temperature increased with decrease of pressure applied but it did not change after a certain pressure condition. However, the buckling failure temperature was linearly increased with the decrease of the pressure for the second mode of buckling. The sensitivity analysis using SAMPSON code has also been performed in order to demonstrate that the passive depressurization system with an optimized leakage area and failure condition is more efficient in managing a severe accident. It was clarified that the passive depressurization systems can depressurize the reactor coolant system to prevent or to mitigate the effects of direct containment heating instead of the safety/relief valves (SRV) if SRV is inoperable or it is stuck in the closed position [46].

In Hokkaido University, high efficiency Filtered Containment Venting System (FCVS) is being developed as shown in Fig. 7 [47] in order to prevent containment vessel rapture and to prevent release of radioactive materials to the environment. Hokkaido University has tested wet type FCVS using venturi scrubber in water pool and dry type FCVS using metallic fiber filter for 1st stage, Silver Zeolite named AgX for 2nd stage. Since the AgX needs superheat steam, it is confirmed through TRAC analysis that it is possible to heat up steam orifice, upper stream of the AgX filter module. It was found very important to suppress the geysering, because it affects to operate FCVS system stable and suppress the water droplet carry over with FP.

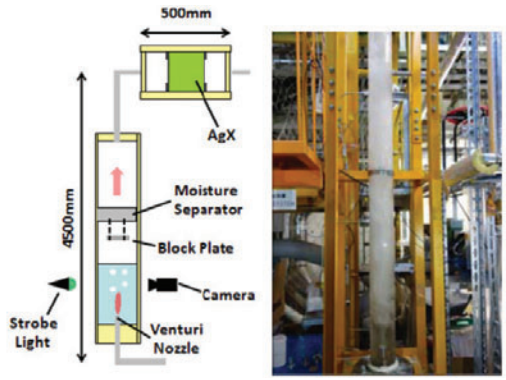

Figure 7. FCVS Test Facility at Hokkaido University [47].

At The University of Electro-Communications, Diffusion of liquid jet discharged from pressurized vessel during severe accident has been investigated [48]. Experiments are conducted to elucidate the effects of the shape of ejection hole and the pressure in the vessel on the characteristics of liquid jet. Numerical analysis using SAMPSON code is also carried out to investigate the effects of the liquid jet diameter on the spreading characteristics of debris on the floor. It was found that the diameter and atomization of liquid jet are influenced significantly by the nozzle shape. The effect of discharge pressure is also significant even if the pressure is several bars.

At Kyoto University, small scale model experiments have been conducted to investigate the effect of counter-current flow limitation (CCFL) in the gap between RPV wall and core debris, and cracks inside core debris for in-vessel coolability with the use of inclined test section with 1 to 4 mm gap width [49]. Also in order to investigate the heat transfer characteristics between porous crust above molten pool and the coolant above the crust with non-condensable gas flowing through the crust during molten core concrete interaction MCCI [50], basic small scale experiment has been conducted using simulated crust, heaters and argon gas as non-condensable gas, as typically illustrated in Figs. 8.

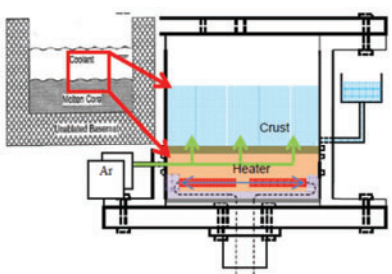

Figure 8. MCCI Experiment at Kyoto University.

6. SUMMARY

(1) Before the Fukushima accident in Japan, considerably active research efforts were conducted in wide varieties of severe accident fields and the research results along with those from worldwide were utilized for the implementation of the severe accident measures. However after the severe accident measures initiated from internal events were deployed at plant sites in early 2000s, severe accident research was drastically reduced in terms of experts and budget.

(2) Several investigation committees were established in Japan after the Fukushima accident and they issued investigation reports with lessons learned from the accident. Some of those reports have emphasized the importance of enhancement of severe accident research, and associated strengthening of international collaboration and nuclear human resource development.

(3) For the identification and prioritization of severe accident research issues systematic approach has been conducted in Atomic Energy Society of Japan. Roadmap of thermal-hydraulic, including severe accident, has recently published. Also thermal-hydraulic and source term PIRTs have been established and published.
(4) Important severe accident research has been conducted after the Fukushima accident in institutes, industries and universities basically in accordance with the outcome of the above roadmap and PIRTs, such as molten core coolability, advanced severe accident analysis capabilities, development of reliable passive core cooling system, analysis of hydrogen behavior and investigation of hydrogen measures, enhancement of removal function of radioactive materials of containment venting, and advanced instrumentation for the diagnosis of severe accident.
(5) Severe accident research activities in Japan based on lessons learned from the Fukushima accident and the recent reviews on important severe accident issues by AESJ, including the assessment of advanced containment design which excludes long-term evacuation in severe accident situations, should be revitalized in order to reestablish and to greatly improve the safety and reliability of nuclear power plants in future.

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