Perspectives on severe accident research in Japan after accident at Fukushima Daiichi nuclear power station

SUGIMOTO Jun[1]

1. *Academy for Global Nuclear Safety and Security Agent, Tokyo Institute of Technology 2-12-1 N1-1, Ookayama, Meguro-ku, Tokyo 152-8550, Japan*

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. In response to the recommendation in investigation report by Atomic Energy Society of Japan (AESJ), the review of specific severe accident research issues was started in AESJ. In 2015 AESJ has developed a new Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap (TH-RM) for Light Water Reactor (LWR) safety improvement and development based on the lessons learned from Fukushima accident. At the same time, Research Expert Committee on Evaluation of Severe Accident established in AESJ, 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. The present paper describes the lessons learned on severe accident research from Fukushima accident, severe accident research issues reviewed in AESJ, several examples of the current severe accident research activities mostly based on this review and some concluding remarks.

Keywords: severe accident research; Fukushima Daiichi; lessons learned from Fukushima accident; MAAP

1. Introduction

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 in 2013. Among those lessons, several recommendations have been made on severe accident research. In response to the recommendation in investigation report by AESJ, specific severe accident research items have been reviewed in AESJ in some committees and working group[6-12]. The present paper describes the lessons learned on severe accident research from Fukushima accident, accident research issues reviewed after Fukushima accident in AESJ, several examples of the current severe accident research activities mostly based on this review and some concluding remarks.

2 Severe accident related lessons learned from Fukushima accident

After the Fukushima accident several investigation committees have been established and the reports have been issued in Japan[1-5]. In some investigation committee reports, safety research, especially severe accident related research, based

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on lessons learned from the Fukushima accident has been recommended. For example, in the AESJ report[5], 50 recommendations have been issued in category groups I through V. Especially in recommendation category group 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 Probabilistic Risk Assessment (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 category group 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.

3 Review of severe accident research issues in AESJ

3.1 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. Ten subjects out of twelve are severe accident related as below:

(1) Alternative cooling system (*ex.* Effect of solutes in cooling water on heat transfer)
(2) Development of new reactor material (*ex.* SiC cladding/ Material with high temperature stability and low H_2 generation)
(3) Development of core catcher (*ex.* Material database of refractory material/ Performance evaluation and verification)
(4) Verification of drywell air cooling system performance under severe accident condition (*ex.* Non-condensable gas effect)
(5) Passive containment cooling system (PCCS)
(6) Coolant injection to reactor well cavity for containment top-head flange cooling and

development of sealing material with high heat-resistance

(7) Passive autocatalytic recombiner (PAR)

(8) Hydrogen removal system for severe accident convergence termination (*ex.* Catalysis for H_2 removal in severe accident late-phase condition)

(9) Filtered venting system (+Experimental data for FP transport, iodine behavior, deposit, vaporization)

(10) 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.

3.2 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]

In the process of establishing thermal-hydraulic PIRT, Modular Accident Analysis Program (MAAP) model enhancement items to improve the simulation capability for molten corium behavior in the accidents at the Fukushima Dai-ichi NPP were employed, since MAAP code has been relatively widely used in the industries in Japan. The importance ranks of the identified phenomena were evaluated for each time phase through brainstorming and discussion with the experts in the above mentioned Committee[6]. 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, which was established in 2012 and funded by the Ministry of Economy, Trade and Industry in Japan[6], 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[7].

4 Severe accident research after Fukushima accident

As discussed in Chapter 3, specific important severe accident research issues have been identified after the Fukushima accident mostly based on the review work in AESJ. In this Chapter several examples of ongoing severe accident research activities, which are basically in accordance with the important issues identified in Chapter 3, are introduced:

4.1 Severe accident research at Institutes

At Japan Atomic Energy Agency (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[13]. Those experimental studies are to obtain better understandings on the phenomena and establish databases for the validation of both lumped parameter 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 transport. For this project, a large-scale containment vessel test facility called CIGMA (Containment InteGral Measurement Apparatus), shown in Fig. 1, has been constructed by JAEA for the conducting high-temperature experiments as well as those on hydrogen risk with CFD-grade instrumentation of high space resolution.

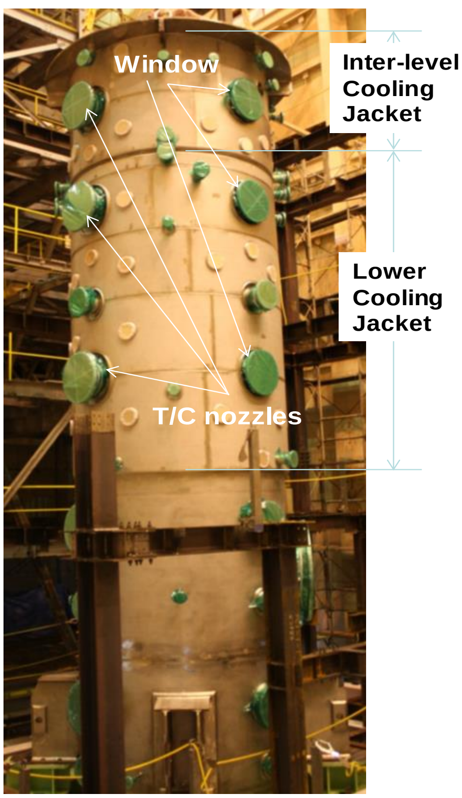

Fig. 1 CIGMA test facility at JAEA[13].

At Institute of Applied Energy (IAE), Severe Accident Analysis Code with Mechanistic, Parallelized, Simulations Oriented towards Nuclear Field (SAMPSON) code development for detailed severe accident analysis has been accelerated after Fukushima accident.[14] SAMPSON is designed as a large scale simulation system of inter-connected hierarchical modules. 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.

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. From Japan side, four institutes have been participating in BSAF Project; Central Research Institute of Electric Power Industry (CRIEPI) with MAAP 5.0, IAE with SAMPSON, JAEA with THALES 2, and Nuclear Regulatory Authority (NRA) with MELCOR 2.1 codes, respectively[15].

4.2 Severe accident research at Industries

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.[16] 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.[17] 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.[18] 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.

4.3 Severe accident research at Universities

In order to clarify the effects of complicated structures in BWR lower plenum on the molten material jet behavior, visualization experiments have been conducted at The University of Tsukuba[19] 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.[20] Pool scrubbing and filtered venting during severe accident conditions have also been investigated with basic experimental facilities.[21]

At The University of Tokyo, experimental and numerical study of thermal stratification and natural circulation in suppression pool has been conducted[22-23]. It was found that the development (formation and disappearance) of thermal stratification was significantly affected by the steam flow rate. 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.[24]

In Hokkaido University, high efficiency Filtered Containment Venting System (FCVS) is being developed in order to prevent containment vessel rapture and to prevent release of radioactive materials to the environment.[25] 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.

At The University of Electro-Communications, Diffusion of liquid jet discharged from pressurized vessel during severe accident has been investigated.[26] 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.[27] 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)[28], basic small scale experiment has been conducted using simulated crust, heaters and argon gas as non-condensable gas.

5 Concluding remarks

In order to further enhance the safety of light water reactors for future, especially in the field of severe accident, the following remarks will be emphasized and considered:

(1) Identified severe accident research issues described in Chapter 3 should, according to its importance, surely be funded, conducted and reflected to the regulation, design, implementation, operation and maintenance at the plant sites. Since it is expected to obtain new findings from the investigation of components of Fukushima Daiichi accident, such as reactor pressure vessel and containment vessel, it is necessary to review the research needs and priorities based on these findings and status of ongoing research activities. This kind of periodical “Rolling” of the severe accident research issues, already started in AESJ, will be important in future.

(2) At the same time, road maps for severe accident research described in Chapter 3 should be compared with those of another nuclear countries or international organizations. Since the national situations, such as external hazards, are generally different with one another, the road maps will never be the same. However a certain kind of international consensus, by understanding each countries or organizations’ situation, would be mutually beneficial. For that purpose international conferences on rector safety or thermal hydraulics will effectively be utilized.

(3) In order to effectively conduct severe accident research in research institutes, industries or universities, human resource development of young generations is essentially important in the mid and long run. For that purpose the Government should clearly support the vital necessity of the nuclear associated underlying safety in mid and long terms. At the same time utilities, research institutes and universities should pursue advanced and dreamful technologies for future so that the young generations will hopefully enter into the nuclear field.

List of acronyms

AESJ: Atomic Energy Society of Japan
CFD: Computational Fluid Dynamics
CIGMA: Containment InteGral Measurement Apparatus
CRIEPI: Central Research Institute of Electric Power Industry
IAE: Institute of Applied Energy
JAEA: Japan Atomic Energy Agency
LWR: Light Water Reactor
MAAP: Modular Accident Analysis Program
MCCI: Molten Core Concrete Interaction
NRA: Nuclear Regulatory Authority
PAR: Passive autocatalytic recombiner
PCCS: Passive Containment Cooling System,
PIRT: Phenomena Identification and Ranking Table
PRA: Probabilistic Risk Assessment

SAMPSON: Severe Accident Analysis Code with Mechanistic, Parallelized, Simulations Oriented towards Nuclear Field
SWG: sub working group
TH-RM: Thermal-Hydraulics Safety Evaluation Fundamental Technology Enhancement Strategy Roadmap

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