Perspectives on phenomenology and simulation of severe accident in light water reactors

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Abstract: Severe accident phenomena in light water reactors (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. Severe accident phenomenology is usually categorized into four phases; (1) fuel degradation, (2) in-vessel phenomena, (3) ex-vessel phenomena and (4) fission product release and transport. Among these, ex-vessel phenomena consist of five subcategories; 1) direct containment heating, 2) fuel coolant interaction (steam explosion), 3) molten core concrete interaction, 4) hydrogen behaviour and control and 5) containment failure/leakage. In the field of simulation of severe accident, severe accident analytical codes have been developed in the United States, EU and Japan, such as MAAP, MELCOR, ASTEC, THALES and SAMPSON. Many different kinds of analytical codes for the specific severe accident phenomena have also been developed worldwide. After the accident at Fukushima Daiichi Nuclear Power Station, review of severe accident research issues has been conducted and several issues are reconsidered, such as effects of BWR core degradation behaviors, sea water injection, pool scrubbing under rapid depressurization, containment failure/leakage and re-criticality. Some new experimental and analytical efforts have been started after the Fukushima accident. The present paper describes the perspectives on phenomenology and simulation of severe accident in LWRs, with the emphasis of insights obtained in the review of Fukushima accident.

Keyword: severe accident phenomenology; severe accident; simulation code; Fukushima Daiichi NPP accident

1 Introduction

Although there was the first major study that developed a probabilistic risk assessment methodology [1], the systematic severe accident research was practically started after Three Mile Island (TMI) -2 accident in 1979 mostly in the USA and some European countries. Severe accident research has been accelerated after Chernobyl accident in 1986 among most nuclear countries. Severe accident research in Japan was started after TMI-2 accident with small-scale experiments and analysis and it was accelerated after Chernobyl accident with relatively large-scale experiment and analysis.

The objectives of severe accident research are (1) to clarify the phenomenological progression of severe accident, (2) to evaluate the effectiveness of accident management (AM) measures, and (3) to quantify the risk of nuclear reactors as an assessment of the safety margin. Severe accident phenomenology in terms of fuel degradation, in-vessel phenomena, ex-vessel phenomena and fission product release and transport, has been developed through these research activities [2-3]. Severe accident analytical codes have been developed based on the knowledge obtained through severe accident research in order to quantitatively describe and predict the severe accident progress, and also to evaluate the effectiveness of accident management measures.

After the accident at Fukushima Daiichi Nuclear Power Station (Fukushima accident) in 2013 several investigation committees have been established in Japan, such as by the Government, Diet and private sectors including Tokyo Electric Power Company. They have issued investigation reports with accident progression, causes of the accident, effects of the accidents and lessons learned from the accident [4-10]. Although the details of the accidents, such as locations and degree of the core and containment damages are not well understood, the basic scenarios of accidents at unit 1 through 4 have been almost clarified in these reports. Based on the lessons learned, several measures, such as enhanced

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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 Japan.

After the Fukushima accidents specific severe accident research issues have been reviewed by some committees and working groups of Atomic Energy Society of Japan (AESJ) mostly based on the findings obtained from the Fukushima accidents [11]. The present paper describes the perspectives on phenomenology and simulation of severe accident in light water reactors (LWRs), with the emphasis of insights obtained in the review of Fukushima accident. [12].

2 Severe accident phenomenology

Figure 1 illustrates important phenomena during severe accidents of LWRs, such as 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. 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. Each phenomenon is described in the following sections.

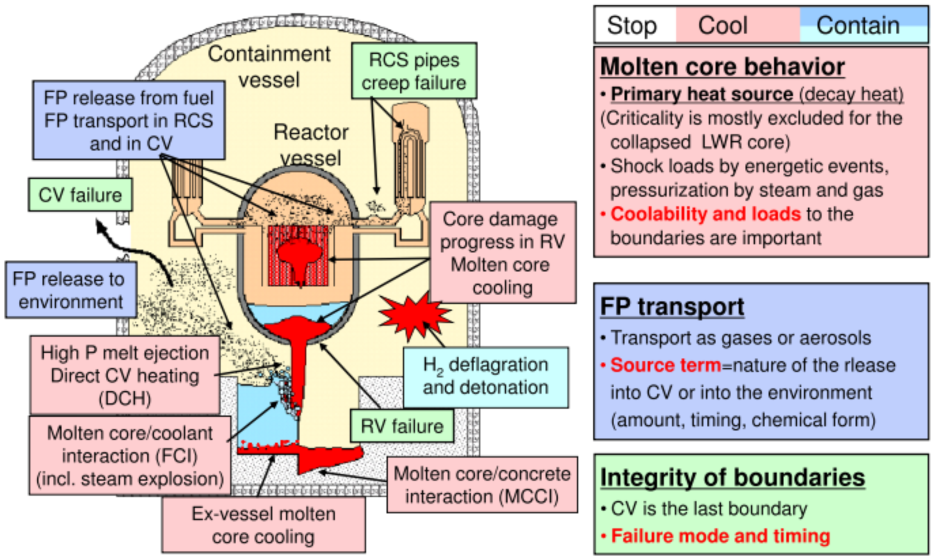

Fig. 1 Important phenomena during severe accident.

2.1 Fuel degradation

Right after the TMI accident there was very few information on fuel degradation behaviors. Therefore experimental program using research reactors, such as PBF (INEL, USA) [13], ACRR (SNL, USA) [14], NRU (CRNL, Canada) [15], and PHEBUS (CEA, France) [16] were initiated. In Japan fuel rod degradation experiment was conducted using NSRR research reactor [17]. In Germany electrically heated rod bundle fuel degradation experiment called CORA [18] was conducted. Also investigation of TMI-2 degraded core was started as TMI-2 R&D program by USNRC [19]. Table 1 summarizes the specification of these experiments.

The followings are the summary of research on fuel degradation:

(1) With the lack of cooling capability of the fuels, the fuel is heated up due to decay heat and the fuel is melt down if the temperature exceeds melting point.

(2) Temperature increase of the fuel is relatively slow with decay heat. However if the temperature exceed above about 1200 K, rapid heat-up was observed due to steam-zircalloy reaction, which causes hydrogen generation.

(3) Control rods drop relatively early due to its low melting point, clad and structural materials follow. Those are accumulated above the grid spacers and the lower support plate.

(4) Fuel (oxides) may react with other materials and melt at temperatures lower than its melting point (3120K) by forming eutectic mixtures.

(5) It was found that TMI-2 molten fuel was heated up to about 3100K by metallurgical examination.

(6) Core collapse and relocation process has large uncertainty. TMI-2 results indicated the multiple paths of relocation of molten core.

(7) At reflooding of damaged core, rapid heat-up and hydrogen

generation due to water-zircalloy reaction is possible.

(8) Compared with PWR geometry, information on fuel degradation behavior for BWR geometry is very limited.

Table 1 Fuel degradation experiments

Name	Organization / Country	Scale of facility	
		Number of pins in bundle(s)	Length of pins (m)
OECD/LOFT	INEL/US	121	1.6
PBF/SFD	INEL/US	32	0.9
ACRR	SNL/US	16	0.9
FLHT	AECL/Canada	12	3.6
PHEBUS	CEA/France	21	0.8
NSRR	JAEA/Japan	4	0.5
CORA	KfK/Germany	25 (electric)	2.0

2.2 In-vessel phenomena

OECD TMI VIP program [20] 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 study has been conducted to investigate the molten core coolability in the lower head. 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 as shown in Fig. 2. 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 [21].

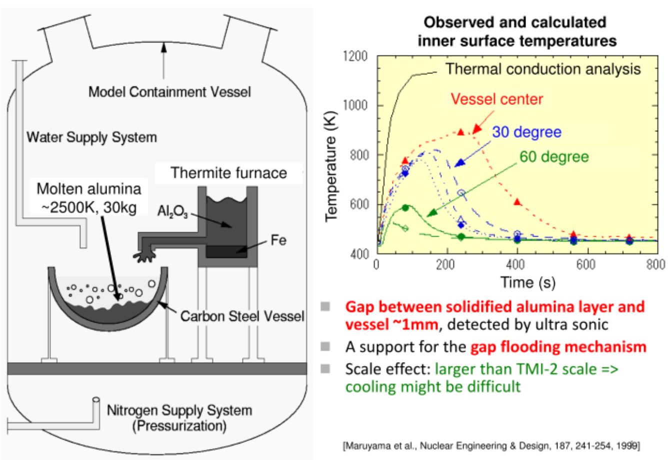

Fig. 2 In-vessel molten core coolability experiment [21].

The followings are the summary of research on in-vessel phenomena:

(1) The molten core cooling and the thermal load to the RPV lower head were studied in the TMI-2 post accident examination, and effectiveness of in-vessel retention, as an accident management, was investigated.

(2) It was found from TMI-2 lower head examination, that the lower head experienced the maximum temperature at about 1370K.

(3) It was suggested from TMI-2 investigation that if RPV fails, nozzle failure is not likely, but middle or large scale creep is more likely.

(4) A hypothesis on the cooling mechanism of the molten core in the TMI-2 lower head was suggested, which assumes cooling by flooding of overlying water to the vessel surface through gaps and cracks before the occurrence of large scale relocation. Experimental results clearly support this hypothesis.

(5) Concerning in-vessel retention (IVR) as an accident management, PWR cavity flooding is proposed to cool and stabilize the molten core inside the RPV. Feasibility has been verified for up to middle size plants (about 600MWe class), but not for larger plants (*e.g.* about 1400MWe).

2.3 Ex-vessel phenomena

The experimental and analytical studies have been conducted for the evaluation of the direct containment heating (DCH) as illustrated in Fig. 3, which starts with high pressure melt ejection (HPME), typically in high pressure scenario of PWR, followed by entrainment of melt droplets by steam blown out, and finally resulting in the rapid heating of containment vessel atmosphere by metal oxidation reaction. For example Surtsey Test Facility at SNL was used to perform scaled experiments that simulate high pressure melt ejection accident [22]. These experiments were designed to investigate melt dispersal from a reactor cavity and the resulting containment loads if the reactor pressure vessel lower head fails while the reactor coolant system is still at elevated pressures.

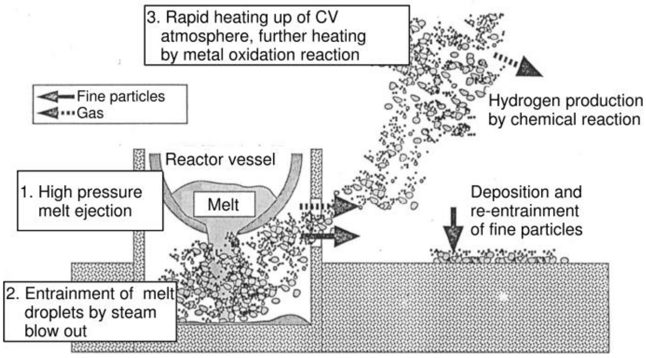

Fig. 3 Conceptual scheme of DCH (Direct containment heating).

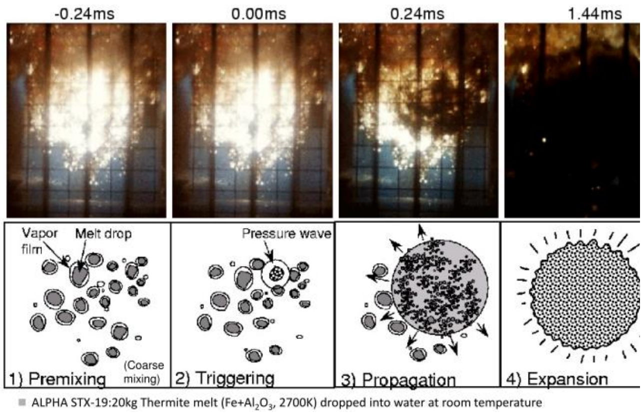

Fig. 4 High speed photographs of steam explosion.

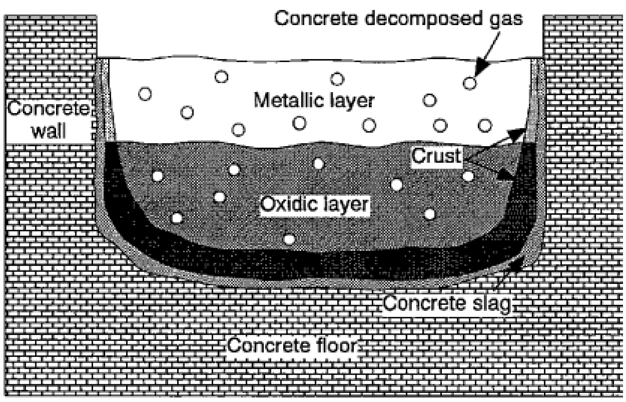

Fig. 5 Conceptual scheme of molten core concrete interaction[31].

Many small-scale and large-scale experiments and analytical studies have been conducted for the fuel coolant interaction (FCI), including steam explosion. In-vessel or ex-vessel large-scale 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. As shown in Fig. 4 with ALPHA experiments at JAEA [23-24], the steam explosion usually has four processes; premixing, triggering, propagation and expansion. Large-scale experiments using corium have been conducted with KROTOS [25] and FARO [26] JRC Ispra, and TROI [27] at KAERI. International research project on steam explosion called SERENA [28] by OECD/NEA has been still conducted using various analysis codes. It may be noted that in Fukushima accident there has been no evidence of a large-scale steam explosion according to the recorded pressure transient data.

Molten core-concrete interaction (MCCI) as typically shown in Fig. 5 could be another threat to the containment integrity. Large amounts of molten corium may enter the reactor cavity after the reactor pressure vessel failure. Because of the continuous release of decay heat in the corium there is a potential for a melt-through of the concrete foundation of the containment by ablation of the concrete. Also concrete ablation generates gas release, especially H_2, H_2O, CO and CO_2, into the containment atmosphere. Whereas the production of steam and carbon dioxide contributes to the pressure increase in the containment, the release of hydrogen and carbon monoxide may eventually lead to the formation of explosive gas mixtures in the containment atmosphere. Both effects have impact on the boundary conditions for long-term leakage processes and may even lead directly or indirectly to an over-pressurization failure of the containment. 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 [29] and OECD/NEA MCCI Projects [30], 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. For example 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 [31]. In Fukushima accident MCCI probably occurred, especially in unit 1. It is supposed that debris cooling during MCCI was established probably by alternative water injection.

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 research on these hydrogen issues, including the effectiveness of hydrogen control systems, such as ignitors and passive recombiners.

It should be noted that the hydrogen explosion really happened in the reactor buildings at Fukushima unit 1, 3 and 4, although the hydrogen explosion may not be anticipated in BWR containment with inert nitrogen inside.

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. In Fukushima accident it is presumed that the containment integrity was lost due to over-temperature, especially for unit 2, and hydrogen leakage to reactor building occurred at unit 1 and 3. Experiments to investigate the containment integrity have been conducted at SNL [32], NUPEC [33] and JAERI [34].

The followings are the summary of research on ex-vessel phenomena:

(1) Concerning DCH, detailed analysis with 3-D computational fluid dynamics (CFD) code is available. Studies so far did not show very high pressurization to jeopardize the containment integrity. Also implemented accident management measures lowered its possibility by quickly depressurizing RCS.

(2) Concerning FCI, in-vessel steam explosion is not likely to occur due to high pressure and temperature environment. Ex-vessel steam explosion loads should be evaluated if there is a chance of melt drop into relatively cold water.

(3) Stopping of MCCI by top flooding is still not well verified. There is a possibility of crust hampering the penetration of water.

(4) Pressurization tests of scaled CV model showed that the models endured up to 3-4 times of design pressure [34]. Steel CVs can rupture at high pressure, but concrete CVs fail by leakage. It is shown that high temperature failure is more plausible than high pressure failure.

(5) Concerning hydrogen issues (concentration, distribution, deflagration, DDT, detonation) the concentration limit is about 4% for H_2 and 5% for O_2 in dry condition. Local concentration depends on break position and compartment arrangement; a detailed analysis needs 3D CFD (mixed gas, phase change, combustion). Hydrogen behavior could have an influence on FP chemistry.

2.4 Fission product release and transport

For the fission product release from the fuel during severe accident, experimental studies have been conducted at ORNL, IRSN and JAEA, and database of the experimental parameters on the FP release has been accumulated. For example VEGA experiment at JAEA, shown in Fig. 6, has been conducted to investigate the effects of fuel temperature, ambient pressure up to 10 bar, atmospheric condition and MOX fuel on FP release [35]. It was confirmed that the release of CsI is obviously suppressed by higher ambient pressure, which gave the first experimental evidence. Analytical models have been developed based on these experimental findings.

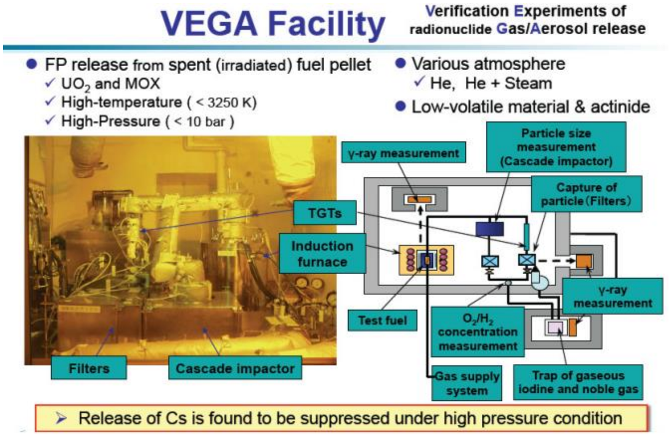

Fig. 6 VEGA experiment on FP release from irradiated fuel[35].

For the FP transport in RCS and CV experimental and analytical studies have been conducted. Among these, 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 [36]. Also iodine chemistry in containment has been investigated at JAEA under high radiation dose using Co-60 [37]. Integral experiments from the FP release from the fuel and FP transports in RCS and CV has been conducted in PHEBUS-FP program[38], shown in Fig. 7, as an international research program using PHEBUS research reactor at IRSN Cadarache, France. In this program test took place in two successive phases: 1) a "degradation" phase, lasting several hours, during which, through an increase in the power for the PHEBUS core, the temperature of the test fuel increased to the point of liquefaction and displacement of materials, leading to release of fission products and transport into the circuit and the containment vessel, and 2) a “containment” phase, lasting several days, during which the quantities of interest in understanding transport phenomena, materials deposits and iodine chemistry in the circuit and the containment vessel were measured.

The followings are the summary of research on fission product release and transport:

(1) Important FPs which have large inventory and energy, are 1) noble gases (Kr, Xe) which is easy to escape to the environment, 2) iodine which has complex chemical behavior and high volatility and thus has large biological impacts, and 3) Cs, Sr etc. which have high or middle volatility and slow decay constant and thus have an impact on soil contamination.

(2) FP release from high temperature fuel depends on several parameters, such as fuel temperature, atmosphere and pressure. The database of FP release has been accumulated and analytical models have been developed.

(3) FP transport in RCS is likely dominated by thermodynamics due to high temperature atmosphere. Iodine will mostly have the chemical form as CsI and Cesium mostly as CsOH. FP gas and aerosol may undergo through deposition due to thermophoresis and turbulence, and re-vaporization due to decay heat or re-suspension due to high gas velocity.

(4) Concerning FP transport in CV, CsI aerosols will fall down mostly due to gravitational settling and dissolve in water. Radiation chemistry effects may produce volatile iodine I_2 or organic iodine. This process is affected by many factors, such as pH, organic impurity and oxygen concentration.

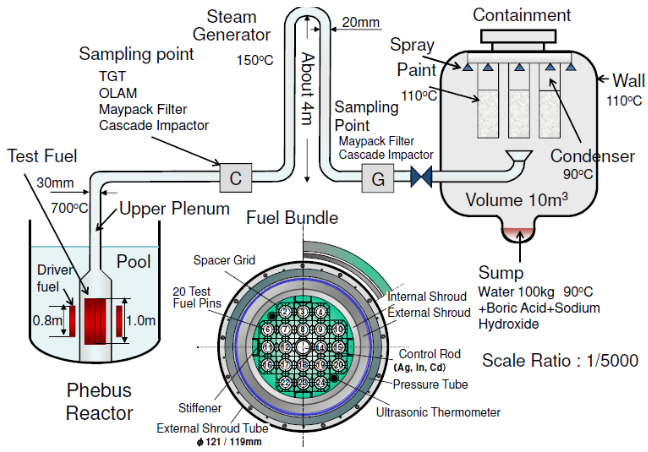

Fig. 7 PHEBUS-FP experiment.

3 Severe accident simulation codes

In the field of simulation of severe accident, three classes of codes can be defined depending on their scope of application; integral codes, mechanistic codes and specific codes [2].

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. They are mainly used for level-2 probabilistic risk analysis (PRA) with fast running capability. The internationally used codes are MAAP (Fauske & Associatees, USA) [39], MELCOR (SNL, under USNRC, USA) [40], and ASTEC (jointly developed by IRSN, France, and GRS, Germany) [41]. THALES [42] 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. Figure 8 shows physical and chemical modes to simulate FP behaviours used in THALES.

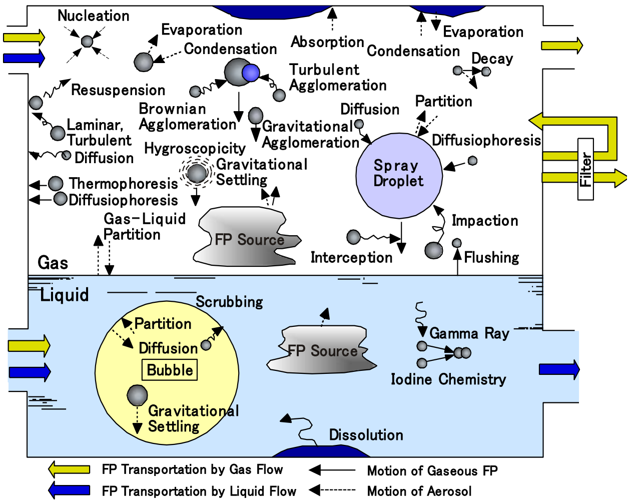

Fig. 8 Analytical models in THALES code.

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. Computation time depends on the scope of the application and on the level of space or time discretization. The internationally used codes are: for RCS behaviour and core degradation, ATHLET-CD (GRS, Germany) [43], SCDAP/RELAP5 (INEL, USA) [44], RELAP/ SCDAPSIM (ISS, USA) [45] and ICARE/ CATHARE (IRSN, France) [46]; and for CV CONTAIN (USA) [47] and COCOSYS(GRS) [48]. SAMPSON code [49] has been developed by Institute of Applied Energy (IAE, Japan) 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. Typical issues for which specific codes are required include: steam explosion and melt dispersal MC3D (IRSN, France) [50], TEXSAS (UW, USA) [51] and JASMINE (JAEA, Japan) [52] and structure mechanics CAST3M (CEA) [53] and ABAQUS (USA) [54].

For the assessment of these codes, verification and validation processes are needed. In the validation process usually three kinds of experiments are utilized through the comparison between the calculation and measurement; separate effect experiments, coupled-effect experiments and integral experiments.

It may be noted that OECD/NEA's BSAF (Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station) Project [55] hosted by JAEA has been initiated 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.

4 Post-Fukushima severe accident research in Japan

In Atomic Energy Society of Japan (AESJ) identification and prioritization of severe accident research issues have been reviewed by Sub-working Group on severe accident since 2009 in terms of significance of consequences, uncertainties of phenomena and maturity of assessment methodology. After the Fukushima accident re-investigation was started with the consideration of additional effects of Fukushima accident and the group identified important issues, such as effects of BWR core degradation behaviors, sea water injection, pool scrubbing under rapid depressurization, containment failure/leakage and re-criticality. [56].

In January 2012, Research Expert Committee on Evaluation of Severe Accident was established in AESJ and in collaboration with the above mentioned Working Group started to investigate severe accident related issues mostly for the improvement of severe accident simulation codes. The Committee has been working to establish phenomena identification ranking table (PIRT) for the modeling of the analysis of Fukushima accident. The first version of PIRT on thermal hydraulic field has been established [57]. Also the first version of PIRT on source term has been almost established by this Committee.

In October 2013, Sub-working Group on basic technology has been newly established in order to develop road map in severe accident research issues by utilizing outputs of the Sub-working group and Committee mentioned above.

The followings are several important subjects on severe accident research mostly based on author's personal view and AESJ's activities through the insights obtained from the review of Fukushima accident [12]:

(1) Investigation of damaged core and components
(2) Advanced severe accident analysis capabilities and associated experimental investigations
(3) Development of reliable passive cooling system for core/containment in case of long-term station blackout
(4) Analysis of hydrogen behavior and investigation of hydrogen measures
(5) Enhancement of removal function of radioactive materials for containment venting
(6) Advanced instrumentation for the diagnosis of severe accident
(7) Assessment of advanced containment design which excludes long-term evacuation in any severe accident situations

Some of new experimental and analytical studies after Fukushima accident have been initiated based on above AESJ's activities and also to meet new licensing requirements. For example, highly reliable passive cooling system for core and containment necessary for the existing and future reactors are developed at University of Tsukuba, which efficiently drives water jet by steam condensation on water jet surface, simultaneously steam is accelerated by condensation above the sonic velocity, without the need of power supply called "supersonic steam injector (SI)" [58] as shown in Fig. 9. In Kyoto University, small scale model experiments have been conducted 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 as illustrated in Fig. 10 [59]. The formed crust during MCCI is simulated by metallic materials with porosity between 30 to 50 %, and argon gas is used as non-condensable gas simulant. The overall heat transfer coefficient from the heater simulating decay heat to bulk water is measured with parametrically varied crust characteristics, such as average hole diameter and porosity, and argon gas velocity.

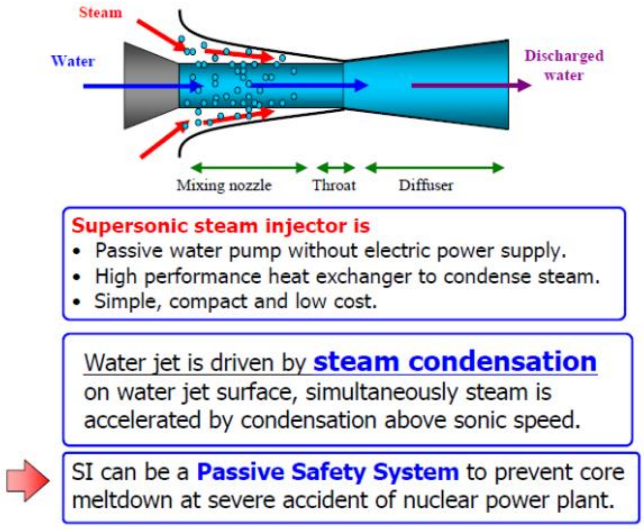

Fig. 9 Concept of supersonic steam injector (SI) [58].

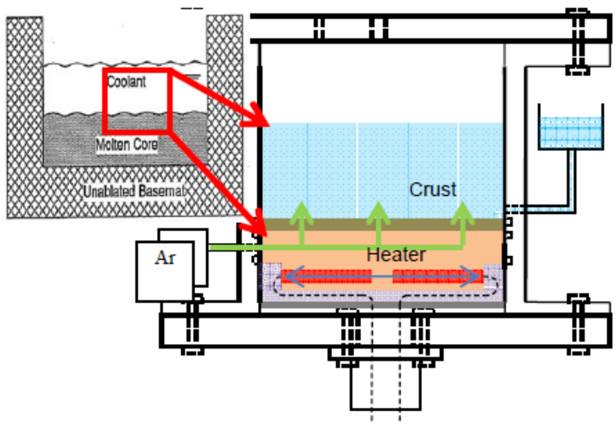

Fig. 10 MCCI experiment at Kyoto University [59].

5 Summary

Although some details of Fukushima accident progression have not been clarified yet, such as the degree and locations of the core degradation or the containment breech, the overall process of severe accident has been basically well understood and analyzed, mostly due to accumulated findings and knowledge obtained in severe accident research and code development efforts worldwide after 1980s. The followings are a brief summary of the perspectives on phenomenology and simulation of severe accident in LWRs, with the emphasis of insights obtained in the review of Fukushima accident:

(1) Severe accident phenomenology has been developed based on researches on severe accident in terms of fuel degradation, in-vessel phenomena,

ex-vessel phenomena, and fission product release and transport.

(2) Severe accident simulation codes, such as integral codes, mechanistic codes and specific codes, have been developed mostly based on the findings from severe accident research, and widely used for the sever accident analysis, including for Fukushima accident.

(3) Investigation of post-Fukushima severe accident research issues has been conducted in Atomic Energy Society of Japan. Important severe accident research items should be identified to further improve the analytical capabilities and reflected to existing and advanced reactors to extensively increase the level of safety for future.

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