Research Article
Review of Severe Accident Phenomena in LWR and Related Severe Accident Analysis Codes

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Abstract: Firstly, importance of severe accident provision is highlighted in view of Fukushima Daiichi accident. Then, extensive review of the past researches on severe accident phenomena in LWR is presented within this study. Various complexes, physicochemical and radiological phenomena take place during various stages of the severe accidents of Light Water Reactor (LWR) plants. The review deals with progression of the severe accidents phenomena by dividing into core degradation phenomena in reactor vessel and post core melt phenomena in the containment. The development of various computer codes to analyze these severe accidents phenomena is also summarized in the review. Lastly, the need of international activity is stressed to assemble various severe accidents related knowledge systematically from research organs and compile them on the open knowledge base via the internet to be available worldwide.

Keywords: Chernobyl accident, Fukushima Daiichi nuclear accident, severe accident phenomena, severe accident analysis codes, TMI-2 accident

INTRODUCTION

There had been two big nuclear power plant accidents in the last century: TMI-2 accident in 1979 and Chernobyl accident in 1986. The both severe accidents had significant influence on public acceptance for nuclear power around the world because:

- Radioactive release to the environment had obliged the extensive evacuation of general public living in the neighborhood of nuclear power plant and
- Many years are necessary to decontaminate the plant site.

After the TMI and Chernobyl accidents, extensive researches had been made on "severe accident" around the world. Based on those research works much knowledge had been accumulated in the period of 1990' on understanding various severe accident phenomena and analyzing the consequence of those severe accident phenomena as the computer simulation programs.

However in March 11, 2011, the third big severe accident happened in Fukushima Daiichi plant in Japan. Why Fukushima Daiichi accident happened in Japan although many researches on severe accident had conducted for almost three decades after the days of the TMI and Chernobyl accidents? From a series of update log on Fukushima Daiichi accident in Japan reported by (Shibutani, 2011a), it can be well recognized that the root cause of this recent big severe accident in Japan since Chernobyl accident comes from a biased group thinking shared by many experts of Japanese nuclear society that it is not necessary to introduce any safety measure against severe accident because Japanese technology is so reliable. As the result of such "self complacency" attitude shared by Japanese nuclear community, no proper provision against probable occurrence of enormous earthquake and tsunami which had happened many times historically in Japan. This teaches us a lesson that ignoring severe accident is very dangerous for our utilization of nuclear power.

These days, many developing countries around the world are starting or considering nuclear power development program. This proliferation of nuclear power technology in many countries around the world may be a conspicuous feature of the 21st century to be compared with the days in the 20th century when limited number of industrially developed countries own nuclear power plants. The hot expectation for nuclear power by many developing countries would be their expectation to meet with their increased demand of electricity as well as to cope with global warming prevention. However, we should be careful about the further possibilities of severe accidents with the proliferation of nuclear power around the world. Therefore, the authors of this study have motivated to

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study on severe accidents of nuclear power plants by extensive literature investigation to the published document, to summarize the up-to-date knowledge on severe accident comprehensively as to the evaluation, prediction and prevention and create effective means and will be involved in "nuclear safety" in the nuclear developing countries in the 21 century.

The objective and the content of this study is limited to review the progress of researches on understanding severe accident related phenomena and the developments of the related analytical methods as computer simulation codes by extensive surveys to the published literatures from research institutions thus far involved in the severe accident related researches in the western countries including Japan. The span of this review is limited for Light Water Reactor (LWR) and the phenomena of severe accident are covered for those which would take place within reactor facilities such as reactor vessel, containment, reactor building. The review of the computer codes are also made for the analysis codes on individual phenomena, integrated codes to explain the progression of severe accident phenomena within nuclear power plants.

CLASSIFICATION OF NUCLEAR REACTOR ACCIDENTS AND GENERAL DESCRIPTIONS OF SEVERE ACCIDENT

According to U.S. Nuclear Regulatory Commission (USNRC), there are nine classes of reactor accidents. They are:

- Trivial incidents
- Small release accident outside containment
- Radwaste system failures
- Fission products to primary system
- Fission products to primary and secondary systems
- Refueling accidents
- Spent fuel handling accident
- Accident initiation events considered in design basis evaluation in safety analysis report and
- Class nine accidents which consist of most serious accident. According to USNRC, severe accident seems to correspond to class nine accidents (Lamarsh and Baratta, 2001)

On the other hand of USNRC, IAEA classifies the nuclear reactor accidents into three categories of design basis accident, beyond design basis accident and as described in subsection of "design basis accident", beyond design basis accident and severe accident, respectively.

Design basis accident: The Design Basis Accidents (DBA) are those accidents chosen by the deterministic method or with the help of probabilistic considerations, in order to design all the plant systems, but particularly the safety ones. A postulated accident that a nuclear

facility must be designed and built to withstand without loss to the systems, structures and components necessary to ensure public health and safety(Gianni, 2006) LOCA is one of the design basis accident.

Beyond design basis accidents: Accident conditions more severe than a design basis accident is called beyond design basis accidents. It may or may not involve core degradation (IAEA, 2008). These accidents can neither be termed as DBAs (because of their low probability) nor severe accidents.These accidents are considered with specific attentionwith mitigation measures. They have low probability and small safety margins as compared to DBAs. Some beyond design accidents are:

- Anticipated Transients Without Scram(ATWS)
- Total loss of external and internal electric power supplies (Station Blackout) (William, 2005; Argonne National Laboratory, 1962) and the other accidents caused by earthquakes, tsunamis, fires, flooding, tornadoes and terrorist attacks

Severe accident: According to IAEA's definition, severe accident is that the accident condition is more severe than a design basis accident with involving significant core degradation due to violent core disruption (Reactivity Events) or slow core melting (Inadequate Core Cooling). The significant feature of severe accident by IAEA (2003c) definition is that whatever the accident initiating condition, the outcome of the accident is significant core degradation.

- **Severe accident seen from risk concept:** The terms by severe accidents we understand potential or actual accidents that represent a significant risk to people, property and the environment (PSI Bericht Nr. 98-16, 1998). In the severe accident, due to insufficient core cooling the reactor core would be heated up until severely damaged. When the core is damaged, the large amount of Fission Products (FPs) is released from the fuel and hydrogen gas is generated in the reactor core region. If lower head of a Reactor Pressure Vessel (RPV) is failed then molten core will release into a containment vessel and various severe accident phenomena such as steam explosion, hydrogen burn and Molten Core Concrete Interaction (MCCI), etc., will occur. These phenomena can threaten the integrity of the containment vessel (Akihide, year).which is the last barrier against the radiological risk to the environment.
- **Severe accident and the defense-in-depth concept:** These severe accidents are extremely unlikely because LWRs are designed based on the defense-in-depth concept. The objective of the defense-in-depth is twofold: first, to prevent accidents and second, if the prevention fails, to detect and limit their potential consequences and to

prevent any evolution to more serious conditions (IAEA, 2001).

- **Severe accident and the analysis codes:** The source term is the most important items to quantify the safety margin, potential risks of LWRs and for the evaluation of accident management measures to prevent and mitigate the severe accidents (IAEA, 2003a). For different kinds of severe accident sequence, the source term can be evaluated by proper integrated computer codes that take into account of most of the severe accidents phenomena such as thermal hydraulics, FP release, steam explosion, hydrogen explosion and MCCI, etc. The main tools for the accident analyses are thermal-hydraulic system codes (Gianni, 2006).

  The computer codes which analyze the specific phenomena were developed based on the test results. However the verification of those computer codes is very difficult and expensive because conduction of the integrated experiments is very difficult and costly to simulate various severe accident conditions properly by using radioactive nuclides.

SEVERE ACCIDENT PHENOMENA

Major phenomena caused by core melt accident: When the core remains uncovered for a considerable period of time, the fuel rod temperature will increase which may eventually lead to significant and irreversible core degradation. The mechanism may be chemical or mechanical which produce such degradation and the consequences may be more or less severe depending on the temperature. Various phenomena caused by severe accident are so broad that they range from nuclear fuel behavior and coolant behavior at off-normal operation condition of NPP to Zr-water reaction, hydrogen production, fission product release, formation of molten corium and propagation towards the lower head.

- **Zr-water reaction:** When the core of reactor remain without water for a long time then the nuclear fuel becomes overheated due to residual heat. Steam flow running over the fuel surface initiates an exothermic oxidation of zircollay cladding, resulting in substantial production of hydrogen gas. The chemical reaction between zircalloy and flowing steam is given by the following chemical eq.:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2\,H_2$$
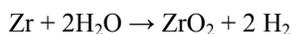
  Various correlation models to describe degree of oxidation, hydrogen generation rate, etc., with the steam temperature change by Zr-water reaction

have been developed experimentally such as Baker-Just model (Baker and Just, 1962).

- **Formation of corium:** Reaction between fuel and its cladding will produce low-melting point eutectics and thus formed material of molten fuel and cladding material is called corium. With heated up further by residual heat, the corium will relocate downwards or upwards in the reactor core, with releasing most of volatile fission product gas in the first place and then semi-volatile products from molten fuel.

- **Corium-water reaction:** Progressively, a corium formed in the reactor core will sink to lower head of the vessel where the corium will contact with the remaining water there. (Corium-water reaction) and then the corium will result in coarse fragmentation which may further cause a violent mechanical phenomena called steam explosion. Lower head or part of core may fail by steam explosion or core may be transformed into a projectile and break the vessel head. Structural element might then be projected towards the containment building and threatening its leak tightness.

- **Hydrogen burning:** Hydrogen produced by core degradation is released into the containment and burns with the contact of oxygen and the provoked pressure and temperature spike can damage the containment building. Combustion can be slow acting (slow deflagration) or more rapid (rapid deflagration, detonation). Hydrogen combustion and ensuing containment loss caused by a reactor core meltdown accident constitutes a risk. Therefore, commitment to making this risk residual has been conducted by the implementation of catalytic hydrogen re-combiners and vented filter systems in the operating plants in several western countries (Petrangeli, 2006).

- **Collapse of reactor containment structures:** Corium eventually causes the rupture of reactor vessel either by thermal erosion, creep or plastic fracture depending on pressure conditions in the coolant system. Under high pressure conditions when the vessel is ruptured, the corium released into the containment may provoke a pressure spike, which results in substantial heat exchange between the air and the corium. Oxidation of the corium's metallic components and combustion of the hydrogen present in the containment building occurred. This phenomenon is called "direct containment heating". The corium is accumulated in the reactor pit and it causes progressive erosion of the concrete basemat, potentially penetrating it. A substantial quantity of incondensable gas is liberated and as the result pressure in the containment increases progressively. Leaks may occur via penetrations in the containment building
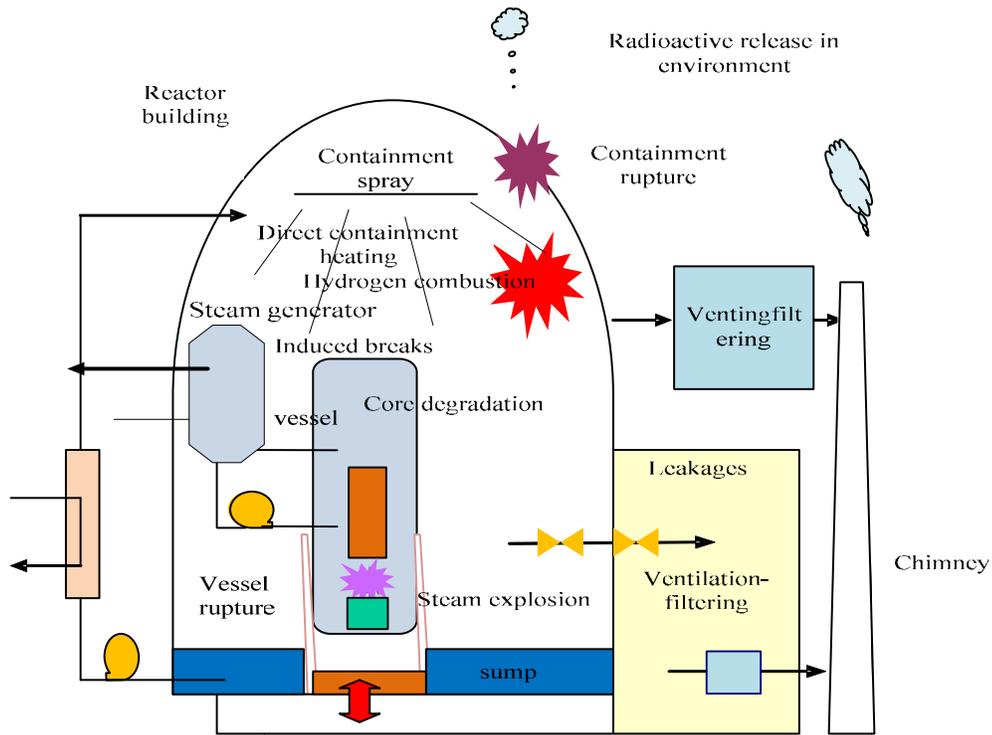
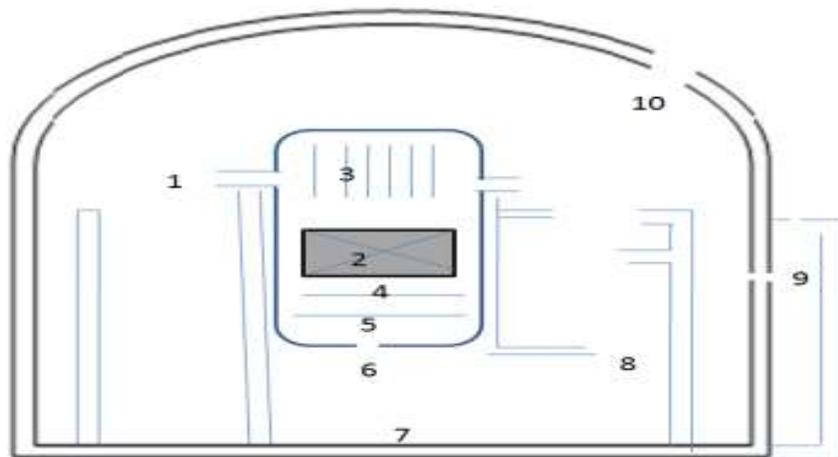Fig. 1: Physical phenomenon during the severe accident [IRSN and CEA-2007/83-351]

Fig. 2: Core melts accident progression

as a result of pre-existing leaks or those generated during containment isolation. For all modes of containment rupture, the release of fission products into environment depends on the conditions affecting their transfer within the facility. Iodine behavior requires particular attention, given its complexity and the significant short term radiological impact. Regarding long-term accident consequences, particular attention must be paid to cesium release (IRSN-CEA 83/351, 2007).

- As the summary of this session, Fig. 1 shows the schematic picture what are liable to occur during severe accident for pressurized water reactor. Figure 2 show the overview of the progression of core melt accident in boiling water reactor with the time scale of major phenomena listed in Table 1 (William, 2005).

Core degradation stage (IRSN-CEA 83/351, 2007; William, 2005): In this section, progression of

Table 1: Core melts accident progression overview

	Accident phenomena	Time
1	Initiation of loss of coolant accident	Start of accident
2	Core uncovered by water	Less than 1/2 hr
3	Volatile fission products released to upper part of vessel /internals and containment	Less than 1/2 hr
4	Lower vessel internal structures fails	1/2-1hr
5	Core debris interaction with residual coolant in vessel lower head	1/2-1 hr
6	Vessel lower head fails	1/2-2 hr
7	Core debris interaction with reactor cavity	2-24 hr
8	Core debris and/or activity release from cavity	12-24 hr
9	Containment leakage	12-24 hr
10	Containment failure	12-24 hr

severe accident phenomena for the stage of core degradation will be described in more detail than in 4.1 by noticing the change of temperature of major materials in the reactor.

Core damage progression will depend on plant design and specific accident scenarios. In this stage, the damage sequence starts with uncovering of reactor core by water and ends with relocation of molten core materials to the lower plenum of the reactor vessel. The progression of the core damage corresponds to the elevated temperature range from approximately 600K (coolant saturation temperature) to over 3100 K (melting point of UO_2). In less than half an hour, the peak core temperature would reach 1100 K. At this

peak temperature, the zircaloy cladding of the fuel rods may balloon and burst. This is the first stage of core damage. The next stage of core damage begins at approximately 1500 K and this is the rapid oxidation process of the zircalloy by steam. In the oxidation process hydrogen gas is produced and a large amount of heat is released. Above 1500 K, the power from this oxidation process exceeds the decay heat unless the oxidation rate is limited by the supply of either zircalloy or steam. When the temperature in the core reaches about 1700 K, molten control rod materials will flow down and solidify in the space between the lower parts of the fuel rods where the temperature is comparatively low. There is generally a pool of water in the lower plenum of the vessel at the time of core relocation. Release of molten core materials into water always generates large amounts of steam. If the molten stream of core materials breaks up rapidly in water, there is also a possibility of a steam explosion. During relocation, any still not oxidized zirconium in the molten material may also be oxidized by steam and in this process hydrogen is produced. Recriticality of the molten core material also may be a concern if the control rod materials are left behind in the core and the relocated material breaks up in unborated water in the lower plenum at temperature 2700K.

Core degradation phenomena with their sequence of events are shown in Table 2.

Table 2: Core degradation phenomena (William, 2005)

| No. of sequence	Phenomena	No. of sequence	Phenomena
1	Clad perforations	9	Flow blockage
2	Clad swelling	10	Clad fragmentation
3	Clad bursting	11	Pellets disruption
4	Clad outer diameter oxidation	12	Fuel rod disruption
5	Clad inner diameter oxidation	13	Fuel melt pooling
6	Fuel melt "candling"	14	Mixed debris melt
7	Fuel/clad eutectics	15	Large scale melting
8	Fuel/grid eutectics	16	Core slumping
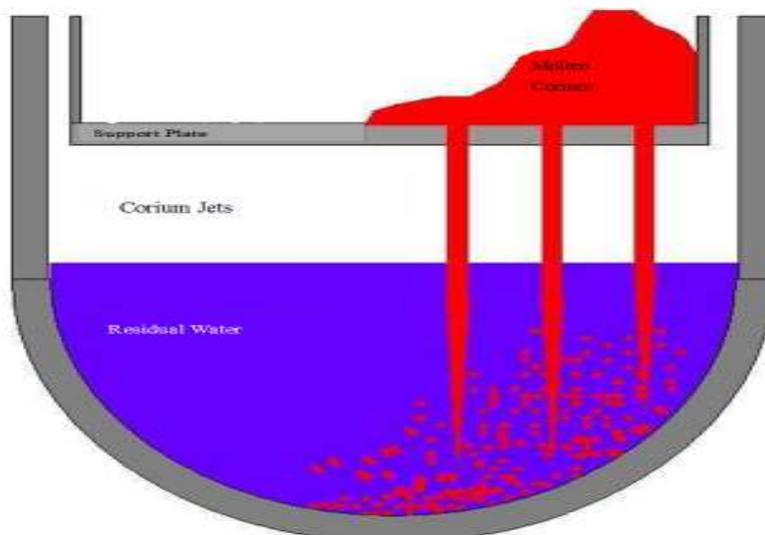
Fig. 3: Condition of core following the accident (TMI-2) (IRSN-CEA 83/351, 2007)
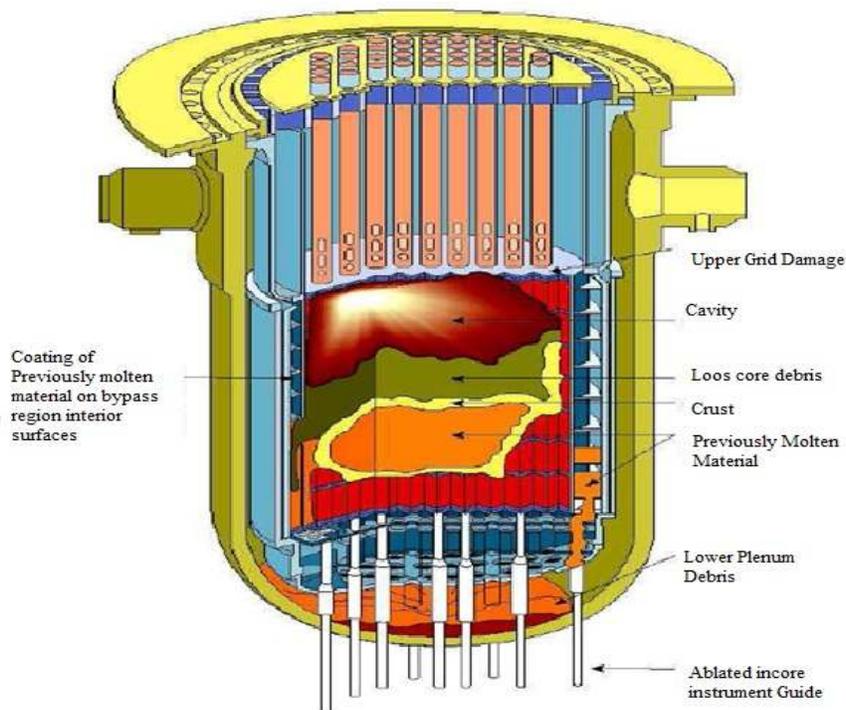
Fig. 4: Molten corium fragmentations with water contact in lower head [IRSN and CEA-2007/83-351]
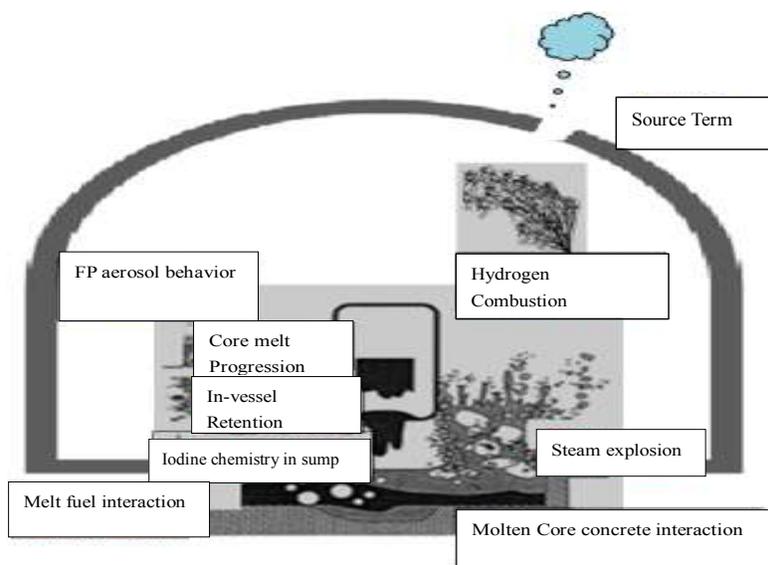

Fig. 5: Major severe accident phenomena (Akihide)

Table 3: Time and temperature of physical phenomena

Time(min)	Temperature (K)	Core state	Effect of Water Addition
110	600	Core uncover	Recovery of water level
<135	<1100	Pre-damage heat up	Pressure increase then decrease, temperature decrease , recovery
135	1100	Ballooning and bursting	Pressure increase then decrease, temperature increase then decrease, recovery
145	1500	Rapid oxidation	Pressure Increase, hydrogen production, temperature increase then decrease, possible recriticality (unborated water),possible recovery
150-210	>1700	Debris bed formation	Pressure increase, hydrogen production, possible recriticality (unborated water), collapse of upper core, possible recovery, possible quiescent response
225	>2800	Relocation to lower plenum	Pressure Increase, hydrogen production, possible steam explosion, possible recriticality, particulate bed formation.

Table 4: Post core accident phenomena

No. of sequence	Phenomena	No. of sequence	Phenomena
1	Large-scale hydrogen generation	7	Aerosols transport radioactivity
2	Thermal-shock-Core fragmentation	8	Steam explosions Missile generation
3	Radiation heat transfer	9	Containment pressurization rupture
4	Hydrogen burning, deflagrationand detonation	10	Ex-containment radioactivity dispersal
5	Corium/vessel interaction-vessel melting	11	Meteorological radioactivity transport
6	Corium/concrete interaction-flammable gases		

Figure 3 show the schematic diagram of reactor core following TMI-2 accident. Of particular concern in the case of core melt accident as is in TMI accident is such a situation as is illustrated in Fig. 4 where large molten pool is formed within the reactor core and then the collapse of a large portion of the rods above the pool (forming a molten corium on the support plate) and then partial corium would relocateas corium jet towards the residual water in the lower head. This corium jet entering into residual water in the lower head will bring about high pressure explosion by the fragmentation of molten corium in the lower head. There are two stages of fragmentation: the first stage of primary fragmentation by premixing involves the continuous phase (corium jet) produce a first generation of droplets (Shibutani, 2011b). In the secondary fragmentation the size of droplets is such that the instabilities causing their division are sufficiently reduced. Table 3 shows the physical phenomena during this stage of accident with time and temperature.

Post-core melts stage (Akihide, year): If lower head of a Reactor Pressure Vessel (RPV) is failed then molten core material accumulated on the bottom of reactor vessel will be released into a containment vessel. This is post-core melt stage where various severe accident phenomena will occur in the containment building such as steam explosion, hydrogen burn, Molten Core Concrete Interaction (MCCI), etc., as shown in Fig. 5. The sequence of various accident phenomena of post-core melt stage is also illustrated in Table 4. The phenomena indicated by the number 10 and 11 are not the phenomena within the containment but the behaviors of radioactive material discharged outside of the containment and dissipating into the surrounding environment.

SEVERE ACCIDENT ANALYSIS CODES

Purpose, objective and usage of severe accident analysis codes: The start of developing practical analysis codes for the various phenomena of severe accidents in light water reactor can be dated back to the days of "Reactor Safety Study" in the US with the name of Rasmussen report or WASH-1400 published in 1975 (USNRC, 1975), when many analysis codes had been developed by a systematical way for evaluating the consequence of various phases of severe accidents phenomena. Those codes were largely based on simplified models with user's tuning model parameters. Since then and especially after TMI and Chernobyl

accidents, various accident analysis codes for light water nuclear power plants had been extensively developed and elaborated in USA, Western Europe and Japan with the conduction of many nuclear safety experiments both by in-pile experiments (integral experiments using nuclear reactor facilities) and by out-of-pile experiments (simulating experiments to test separate effects of accidents) for deep understanding of severe accident phenomena and for detailed evaluation of severe accident as well as for proper management of severe accident. Although it is possible to grasp the overall trends of severe accident conditions by using rather simplified analysis codes, much detailed analysis must be required especially for assessment of accident managements, because multi-dimensional and complex phenomena would appear under severe accident conditions (Naitoh and Hosoda, 2005).

The initial code development took place in the sixties and seventies and it resulted in a set of quite conservative codes for the reactor dynamics, thermal-hydraulics and containment analysis. The limitations of these codes came from insufficient knowledge of the physical phenomena and of the limited computer memory and speed.

In the deterministic safety analysis the assumption varies from pessimistic (conservative) to realistic (best estimate). The assumptions taken for those analyses would cover the selection of physical models, the introduction of these models into the code and the initial and boundary conditions including the performance and failures of the equipment and human action (IAEA - TECDOC-1351, May 2003b). The direct verification of the analysis code is mostly impossible or extremely expensive due to difficulty of integrated experiments which simulate severe accident condition using radioactive nuclides. Therefore, a lot of separate effect tests have been performed to get specific computer codes which analyze the specific phenomena based on the test results (Akihide).

Integrated severe accident codes (IAEA, 2003b): There are many analysis codes for simulating individual phenomena of severe accident. These analysis codes can be categorized into the different groups as shown in Table 5, where various analysis codes are classified into several groups.

On the other hand, the integrated severe accident codes are formed by selecting and combining those individual analysis tools in Table 5. They can be used to model the whole sequence of the severe accident which may occur in the plant system or in the experimental facilities.

Table 5: Analysis codes to be used for severe accident analysis (IAEA, 2003b; Safety Reports Series, 2002)

Groups	Analysis codes	Specific features
Fuel behavior codes	FALCON/FREY FRAPTRAN, SCANAIR, FRAPCON FRAPTRAN	The fuel behavior codes include models, correlations and properties of cladding plastic behavior, phase change, and large cladding deformation (ballooning) as well as fission gas release.
Reactor dynamics codes including those coupled with plant dynamic model	PARCS/RELAP5,HEXTRAN,HEXBU,SMABRE, KIKO3D/ATHLET, RELAP5/PANBORELAP5-3d, NESTLE,PANTHER,HEXTRAN/SMABREDYN 3D/ATHLET	Solution of the time-dependent two-group neutron diffusion equation in three-dimensional Cartesian geometry using nodal methods to obtain the transient neutron flux distribution. The incorporation of full 3-D modeling of the reactor core into system transient codes allows 'best estimate' simulations of interactions between reactor core behavior and plant dynamics.
Thermal-hydraulics, Integrated system, sub channel , porous media, and computational fluid dynamic methods	TRACE,TRACE-P,TRACE-B,SNAP RELAP5, CONTAIN, APROS, ATHLET ,CATHARE, CATHENA	Thermal hydraulics codes are used to analyze loss of coolant accidents (LOCAs) and system transients in light-water nuclear reactors in 1-D and 3-D space. The combined ECCS injections to both cold and hot legs are possible. Sub channel and Porous media codes can be used to analyze localized flow effects in representative fuel assemblies such as to know the influence of spacer grids and flow blockages on local heat transfer. Computational fluid dynamic (CFD) codes can be used to analyze the effect such as mixing in down comer region.
Containment thermal-hydraulics codes	WAVCO, RALOC, COCOSYS GOTHIC, CONTAIN	The analysis of maximum pressure and temperature during the LOCA and SLB accidents, The analysis of minimum back pressure during large break (LB)-LOCA .The analysis of differential pressures on the containment internal structures during LOCA and SLB as a basis for estimating loads on containment internals, The behavior of containment pressure and temperature during severe accidents (containment overpressure, long-term cooling, hydrogen distribution, hydrogen burning, aerosol behavior for ex-vessel analysis).
Severe accident analysis codes	MELCOR, MACCS2, SCDAP/RELAP5- 3D,CONTAIN,MAAP, IFCI,ICARE,ATHLET-CD,VICTORIA,IMPACT, COCOSYS,SAMPSON GOTHIC, WAVCO	Severe accident codes to model the progression of accidents in light-water reactor nuclear power plants. They range from Thermal-hydraulics, to Hydrogen distribution and reduction, to hydrogen burning, and to aerosol models. Severe accident phenomena treated by MELCOR include thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heat up, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. MACCS2 is used to calculate dispersion of radioactive material to the environment and the population.
More detailed mechanistic severe accident analysis codes	TRAC/MELPROG ICARE, CATHARE ATHLET-CD,CFD	The integrated codes apply parametric models to many physical phenomena, but they are not necessarily suitable for cases, where more profound modeling is required. The common feature of these codes is that they have more accurate models and do not differ from the respective thermal-hydraulic system code as long as there is no significant core degradation.
Computational fluid dynamics (CFD) / Fire analysis codes	CFD,PHOENICS CFX,FLUENT TRIO,FINFLO, CAMP code	CFD codes are used in the limited computational area of concern from reactor safety. The CFD and fire analysis codes are used for application of experimental correlations for rough estimates of the fire development, calculation of fire and gases while CFD codes are needed for complicated compartments. The CAMP code evaluates the thermal load onto the RPV lower head and to quantify molten debris conditions at the lower head failure.
Design-Basis Accident (DBA) codes	RADTRAD	DBA codes are used to determine the time-dependent dose at a specified location for a given accident scenario. It also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation.
Health effects/dose calculation	VARSKIN	Health effects/dose calculation codes are used to model and assess the health implications of radioactive exposure and contamination.
Radioactive nuclide transport codes	DandD, Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Codes	Radionuclide transport and decommissioning codes provide dose analyses in support of license termination and decommissioning. DandD code is used for screening analyses for license termination and decommissioning. The RESRAD code applies to the cleanup of sites and the RESRAD-BUILD code applies to the cleanup of buildings and structures.)
Structural analysis codes	Frame 3DD	Structural analysis codes are used to describe the behavior of the vessel, Piping and containment structures under various accident conditions.

The integrated codes give an overall picture of severe accident with respect to the sequence of phenomena and the timing of events. The integrated codes include many parametric models and accordingly their application range is limited for profound understanding and modeling of the phenomena and interpretations have to be made with special care. The STCP (Source Term Code Package) was initially developed as an integrated package in which many separate codes were included. The widely applied and up-to-date integrated codes for severe accident analyses are MAAP4 and MELCOR, which have been developed and maintained under contracts with EPRI and NRC in the USA, respectively. These codes have been developed as an integral code system from the beginning. There are also European and Japanese integrated codes such as ESCADRE, ASTEC and THALES. The detailed explanation of some representative code systems is given in subsection "representative code systems".

The first and most important item for the influence brought by severe accident is the "source term" which gives quantitative definition of the species, chemical form, amount and timing of FP released to the environment. The source term analysis codes were prepared by integration of several codes to analyze specific aspects of FP release in severe accident. Figure 6 shows various severe accident analysis codes
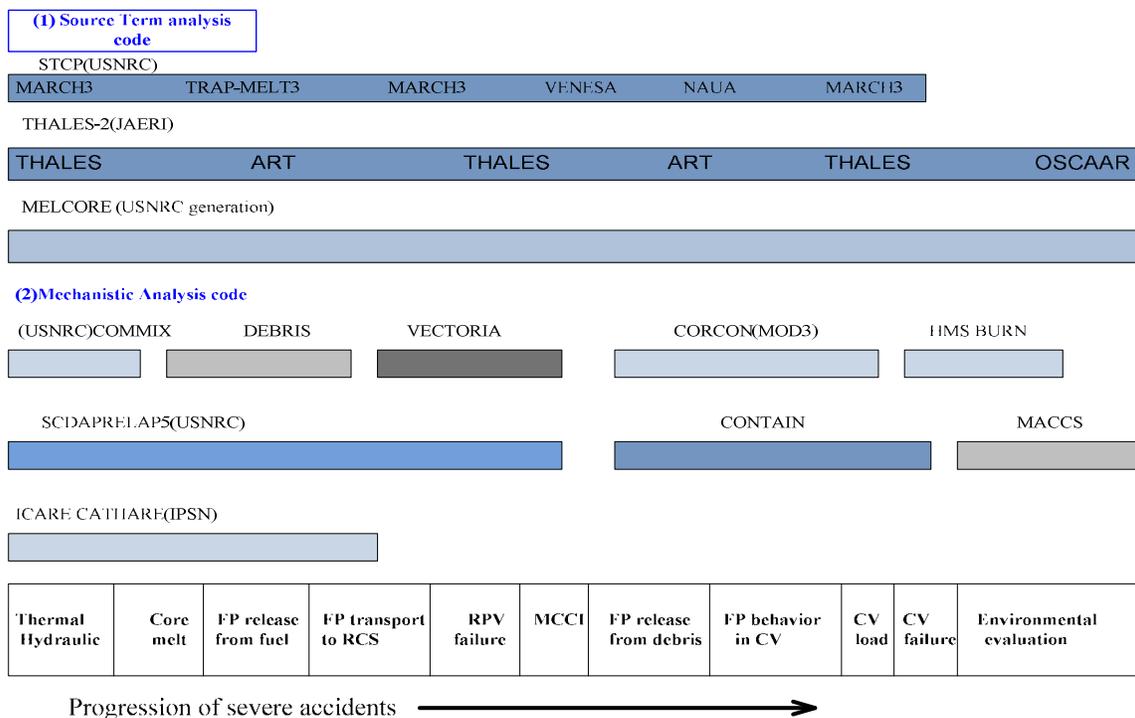
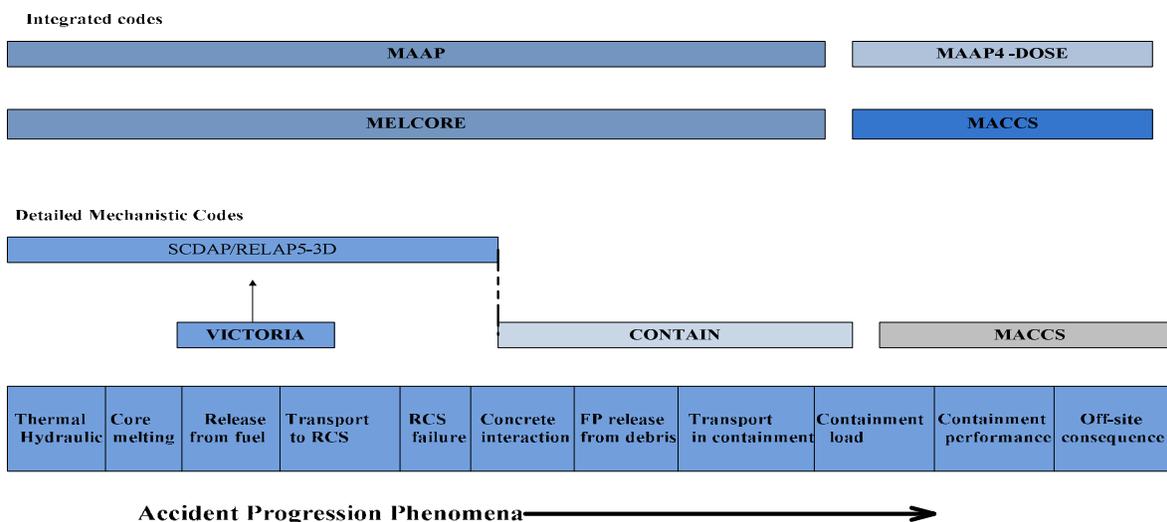Fig. 6: Progression of major severe accident

Fig. 7: Integrated and mechanistic codes (William, 2005)

developed by U.S Nuclear Regulatory Commission (USNRC) and Japan Atomic Energy Research Institute (JAERI) with the classification of source term analysis code and mechanical analysis code. In Fig. 6 the chronological progression of different phenomena of severe accident is also illustrated by the row of horizontal blocks. You can see from Fig. 6 which analysis code covers what parts of severe accident phenomena. Similarly as in Fig. 6 you can see from Fig. 7 how the both of integrated codes and detailed mechanistic analysis codes can be applied for the whole

sequence of severe accident (Baker, 1962). The sequence of severe accident is the same one in both Fig. 6 and 7 but Fig. 7 show the application of integrated and mechanistic codes on whole sequence of accidents.

Representative code systems:

o **MAAP code (William, 2005):** The MAAP code has the capability of integrated RCS and containment analysis for parametric study. It has

the control system/trip logic functions, lumped parameter models easy to make approximations, design specific versions of LWR plants with relatively fixed thermal-hydraulic system representations. The model validation against experimental data can be made by employing special models or versions in the MAAP. Figure 8 shows the representative models to simulate different plant configurations

where,

o   PWR primary system model
o   MAAP 4 containment modeling
o   The MAAP model for steam generator
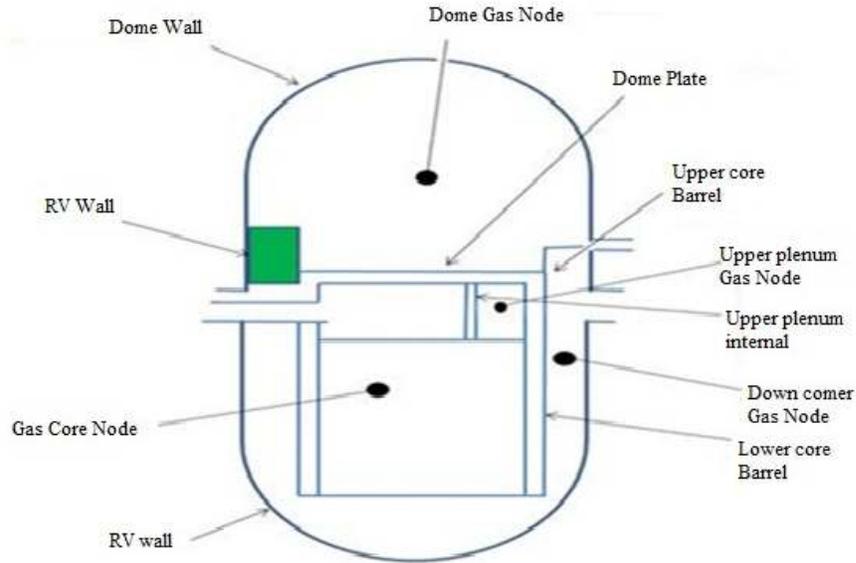
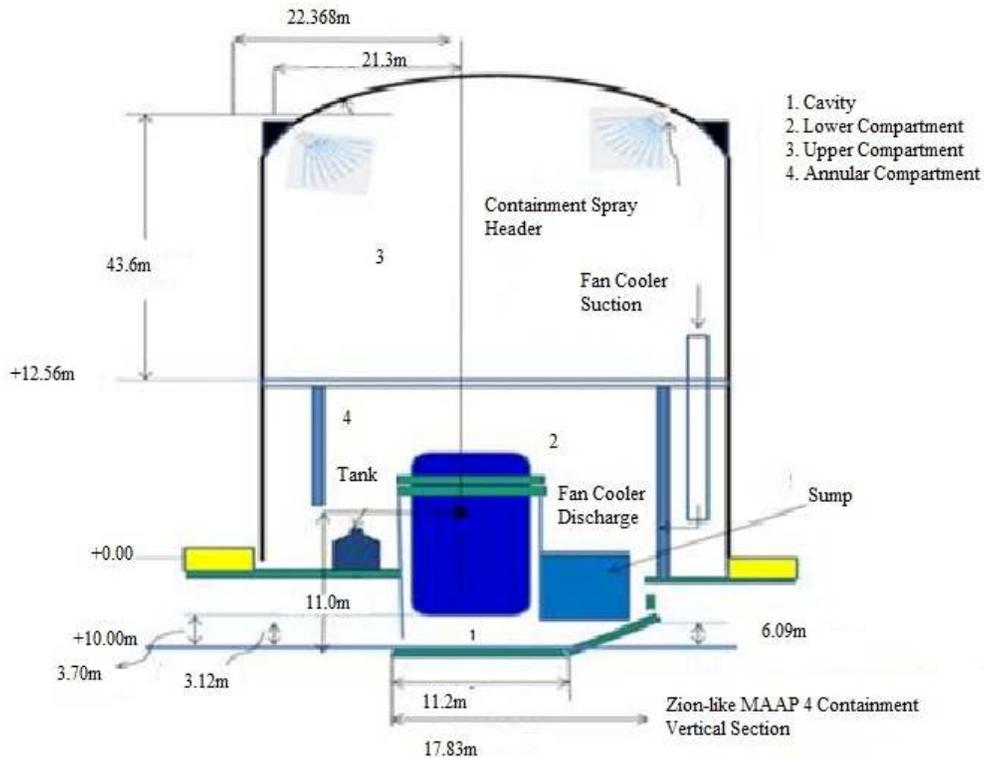Fig. 8: (a) PWR primary system modeling by MAAP4 (William, 2005)

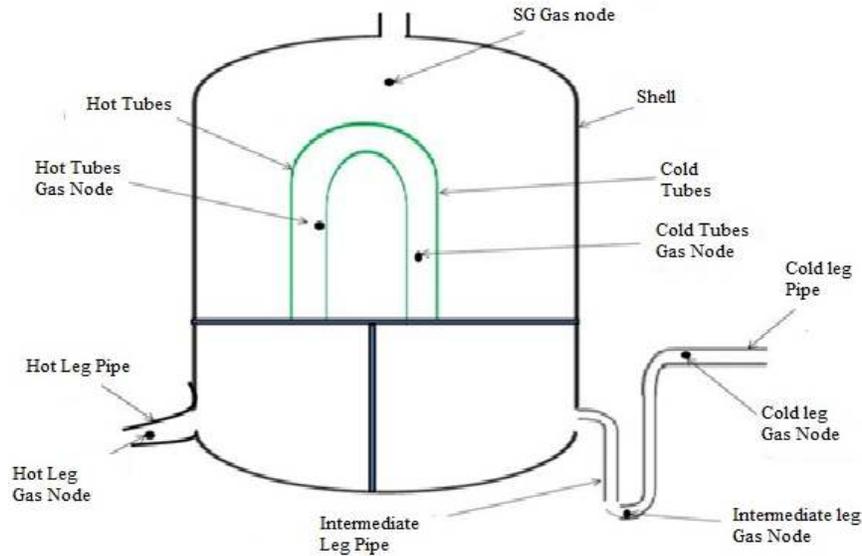Fig. 8: (b) MAAP4 containment modeling (William, 2005)

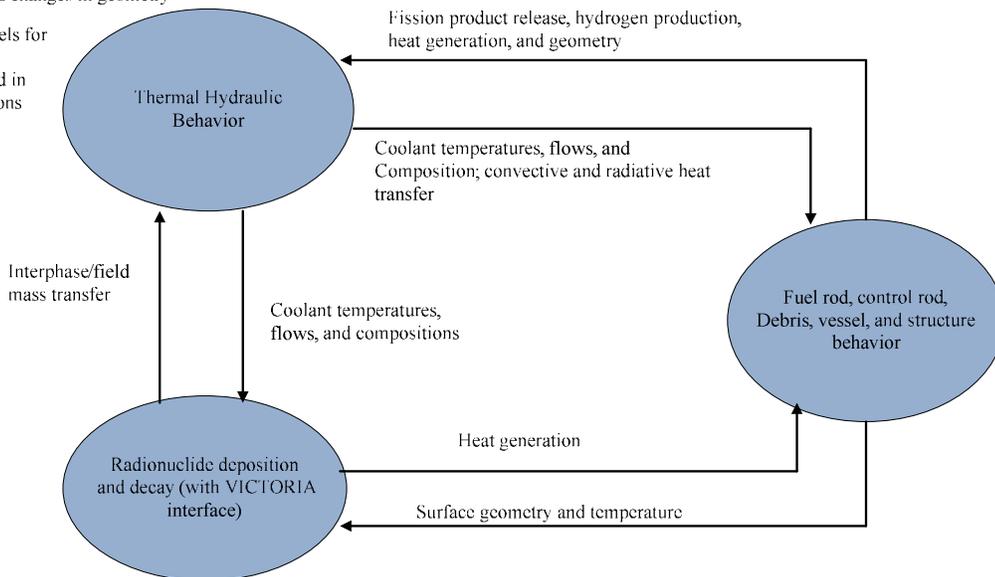Fig. 8: (c) PWR primary system modeling for SG by MAAP4 (William, 2005)

Fig. 9: SCDAP/RELAP5-3D model (William, 2005)

- **SCDAP/RELAP5-3D code (The SCDAP/ RELAP5-3D © Code Development Team, 2002):** The SCDAP/RELAP5-3D code developed by INEEL has been designed for simulating severe accidents as an advanced, state-of-the art, best-estimate computer code. As illustrated in Fig. 9 the SCDAP /RELAP5-3D is a mechanistic code by integrating the analysis models of fission product transport and containment response into the accident simulation system to predict thermal-hydraulic response of the Reactor Coolant System

(RCS), damage progression in the reactor core and heat up of reactor vessel.

- **CONTAIN code (Murata *et al.*, 1997):** The CONTAIN code developed by Sandia National Laboratories (SNL) has been designed for predicting the physical, chemical and radiological conditions inside the containment with the connected buildings of a nuclear reactor in severe accident. It can predict the thermal-hydraulic response inside containments and the release of
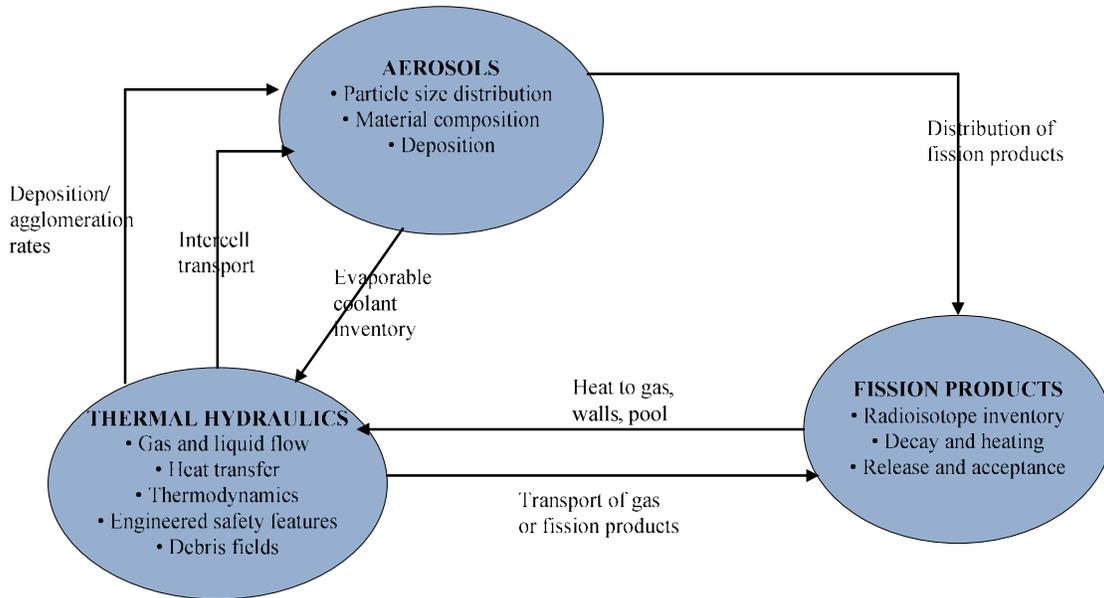
Fig. 10: CONTAIN model (William, 2005)

radioactive nuclides to the environment in the event of containment failure. As is illustrated in Fig. 10 the CONTAIN code consists of the three basic modeling element of:

o Fission product
o Thermal hydraulics
o Aerosols

The modeling capability of CONTAIN is so flexible that it can be applied to the analysis of non-reactor problems such as the migration of radioisotopes in waste repositories and the thermal hydraulic response of non-nuclear facilities under accident conditions.

- **MELCORE code:** MELCOR is a fully integrated severe accidents analysis code also developed by SNL. It can treat the accident sequence from the initiating event, to core uncovery and its damage, fission product release and its transport through the reactor coolant system and containment and finally release to the environment.

    In Fig. 11 several modeling upgrade programs are indicated to improve the capability of the MELCOR modeling of containment phenomena. MELCORE accident consequence code system (MACCS) which is also developed by SNL and this code can simulate the impact of severe accidents at nuclear power plant on the surrounding environment. The phenomena which can be modeled by MACCS are as follow:

    o Atmospheric transport and deposition,
    o Mitigative actions
    o Dosimetry

o Health effects and
o Economic costs (Rollstin *et al.*, 1990)

The ARTIST (Aerosol Trapping in a Steam generator) is experimental program which was conducted at Paul Scherer Institute and can simulate the flow and retention of aerosol-borne fission products in the SG and also provide a unique database to support safety assessments and analytical models (Güntay/LTH, 2001) Similarly the MACE and ANLACE program can model the natural circulation and secondary coolant faults in gas cooled nuclear reactors.

o **CAMP code:** The CAMP code is thermo-fluid dynamics codes which can analysis the molten debris in the lower plenum of RPV and is developed at JAERI. The objectives of CAMP code are to evaluate the thermal load onto the RPV lower head and to quantify molten debris conditions at the lower head failure. The new version of CAMP code can be used for three dimensional geometries. The others function of code are as follow, it is capable of analyzing laminar and turbulent natural convection of single and stratified two component fluids with volumetric heat generation, solid-liquid phase change, heat conduction of vessel wall, interfacial gap formation between debris and water penetration into the interfacial gap(Akihide, year).

Examples of severe accident calculation by MELCORE: The MELCOR calculation of early stages of a station blackout event at Pressurized Water Reactor (PWR) is shown in Fig. 12. In Fig. 12 events are shown with time verses pressure at station blackout event
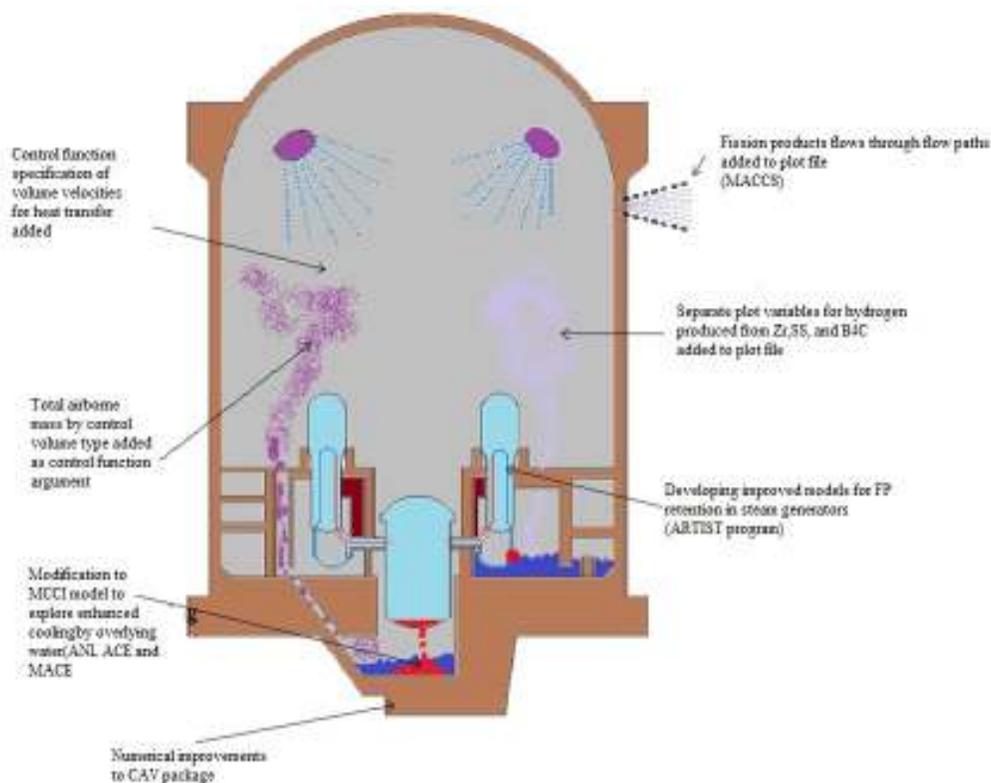
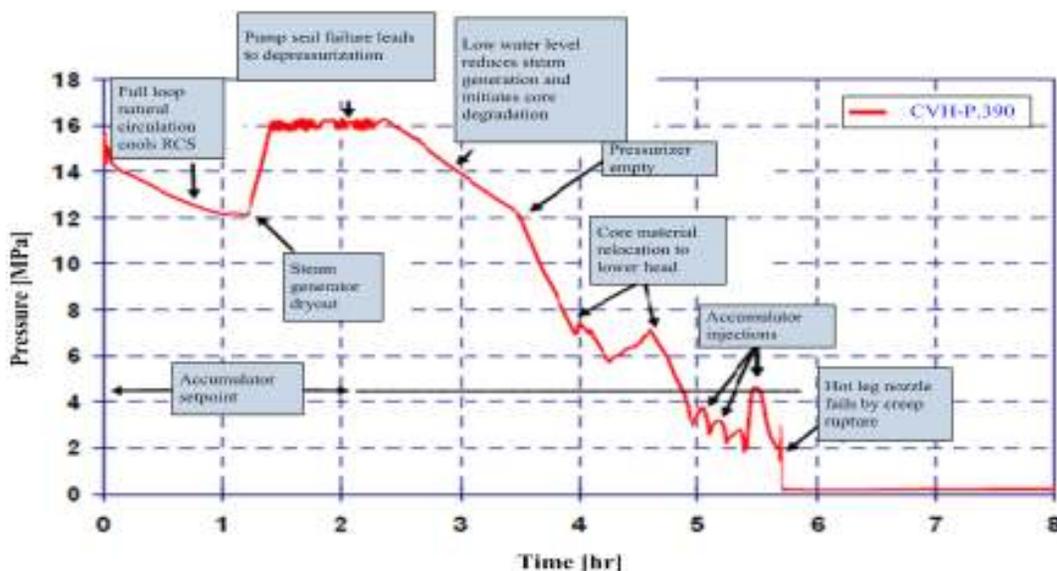Fig. 11: MELCOR modeling of containment phenomena (IRSN-CEA 83/351, 2007)

Fig. 12: MELCOR calculations of early stages of a station blackout event at PWR (Dana, 2008)

(Dana 2008;NRC FIN A3281, 1994). During the early stage of station blackout the depressurization starts which causes the reactor shutdown.

The analysis assumed a depressurized secondary side where heat transfer is dominated by natural circulation of air and steam. The MELCORE calculation leads to early stage steam generator failure.

The low water level reduces steam generation and initiate core degradation. The result of the transient show pressurizer empty and core uncovery and initial clad failure and gap release.

The station blackout is caused by loss of all AC power (loss of offsite power and subsequent failure of the diesel generators). The loss of all off-site and on-

Table 6: Comparison between UCLA experiment data and CAMP code analysis

	Experiment	Analysis
Maximum fluid temperature(K)	300.4	297.3
Vessel inner surface temperature at bottom (K)	290.5	290.9
Upward heat loss (W)	21.2	6.9
Downward heat loss (W)	103.6	125.2

Table 7: Sequence of events and its timing

Events	STCP (min)	THAL ES-2 (min)	MELCOR (min)
Core uncovery	5.3	14.6	18.8
Core melt initiation	40.3	46.6	55.2
Core support failure	59.6	60.4	90.9
Core collapse	56.7	123.5	---
Vessel Failure	79.7	141.6	175.1
Containment Failure	254.4	384.2	574.5

Fig. 13: Comparison of heat transfer coefficients distribution along the vessel wall between UCLA experiment and CAMP analysis

site AC power leads to the loss of all active engineering safety features except the steam powered emergency core cooling systems. In this respect, the MELCORE calculation shown in Fig. 12 gives us the warning that the molten core material relocation is very fast as to already occur before 4 to 5 hours even in case of PWR.

This situation might be almost the same as that happened in Fukushima Daiichi nuclear power station after the big tsunami wave hit the plant site in March 11, 2011.

Comparison with experimental data: The comparison between the CAMP code and experimental data given by University of California Los Angeles (UCLA) is summarized in Table 6 with respect to the fluid temperatures and heat losses from the upper surface of the fluid and from the outer surface of the glass vessel. Except for the heat loss from the fluid upper surface, the analytical results agreed well with the observed ones.

At University of California Los Angles (UCLA) an experiment was performed for natural convection and analyzed with CAMP code. According to this experiment, a volumetric heat generation of molten debris was simulated by microwave heating of Freon-113 (R113) in a hemispherical Pyrex glass vessel. In the present analysis, inner diameter and wall thickness of vessel were 436.5 and 11 mm, respectively. The rate of volumetric heat generation was 6.17kW/m3 and supplied to the fluid with a depth of 21.83 mm at the vessel center axis, which corresponds to an internal Rayleigh number (Rai) of 3.03*1013. To establish a constant temperature boundary, the vessel outer surface was covered with a water pool of a constant temperature. The temperature of upper surface was kept almost constant value of 295.2K. The geometry of the UCLA experiment was reconstructed by an axis-symmetric model of CAMP code. The fluid in the vessel was discretized with 7,500 volumes (Akihide, year). The distribution of local heat transfer coefficients (hlocal) along the vessel inner surface
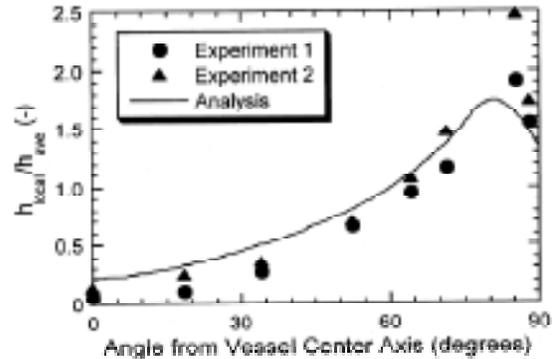
normalized by average one (have) is compared with the experimental data is shown in Fig. 13. With the same conditions experiments 1 and 2 were performed to confirm the reproducibility. The experimental result shows a slight difference at the vessel bottom and top areas, CAMP code well predicted the overall trend of the distribution of heat transfer coefficient

Inter-comparison of calculated results: The sequence of events and its timings of occurrence are shown in Table 7, where the calculation data were given by three codes THALES-2, STPC and MELCORE. From Table 7, it is seen that the STCP calculation gives the occurrence of each event the fastest prediction and then followed by THALES-2 and the MELCOR. The reason why the discrepancy in the three codes can be ascribed to the difference of the analytical models in the codes: The whole core collapse predicted by the STCP calculation gives fastest prediction because molten fuel has to stay at their original location until it meets with the criterion of whole core collapse was met. The melt progression in THALES-2 is slower than that of STCP because the THALES-2 uses candling model for detailed melt progression. But MELCOR calculation gave the slowest accident progression because of the failure model such that the failure criteria were evaluated for each of the radial ring of the core support plate and the RPV bottom head (Akihide, year).

CONCLUDING REMARKS

For almost three decades since the days of Three Mile Island accident (TMI-2) and Chernobyl accident, many researches had been conducted both experimentally and analytically on severe accident phenomena in different organizations in the western countries including Japan. A lot of important phenomena and their understanding had been clarified and the related analytical tools had been advanced by those international research activities towards the end of the last century.

But looking over the occurrence of Fukushima Daiichi accident in March 11, 2011 and looking over the present situation in Japan that the real happened sever accident of the Fukushima Daiichi plants has neither been clarified nor safe containment of the damaged plant has been attained until the time of this review, it was a pity that the international efforts to compile knowledge bases for coping with severe accident prevention had not been effectively utilized in Japanese nuclear society to prevent the occurrence of severe accident on one hand, while the compiled knowledge base on severe accident thus far is not enough for effective severe accident management. It is therefore, important to continue severe accident research for the evaluation of effectiveness of accident management methodology to enhance safety provision against a wider range of severe accident scenarios.

In this study, the authors of this study reviewed the past research results from the limited open references in the world, in order to summarize the present understanding level of various important phenomena related with severe accident in LWR and the related computer codes developments to estimate the consequences of these accidents. However, the range of the presented reviews is limited to the severe accident phenomena within the nuclear power plants.

The authors' review presented in this study does not deal with the following subjects which are also important subjects for the compilation of knowledge base for severe accident management:

- Compilation of material properties, modeling and correlations related with severe accident phenomena
- Compilation of various experimental research products thus far conducted which includes separate effect and integrated experiments in both out-pile and in-pile experimental facilities
- Application of the computer simulation for the interpretation of those experimental studies in (ii) as well as for the analysis of real severe accidents such as TMI, Chernobyl and Fukushima Daiichi,
- Compilation of effective countermeasures to preclude the occurrence and mitigate the consequence of various severe accident related phenomena within the nuclear facility and further,
- Understanding various dissipative behaviors and the influence of various radioactive materials discharged from nuclear power plants to the environment as well as various cause and consequence of various types of radiation exposure and doses in the event of big severe accident with the INES (International Scale of Nuclear Event) as high as level 7

Those items listed above are very broad and versatile and therefore it is very difficult for the authors of this study to do it by themselves and therefore, they would like to conclude this review by proposing that the assembling of the knowledge base on severe accident should be pursued systematically by international authorities such as IAEA, in order to compile the knowledge base as commonly available media such as over internet to be easily accessed and utilized by all interested experts and citizen worldwide.

ACKNOWLEDGMENT

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