INVESTIGATIONS ON THE RESOLUTION OF SEVERE ACCIDENT ISSUES FOR KOREAN NUCLEAR POWER PLANTS

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Under the government supported long-term nuclear R&D program, the severe accident research program at KAERI is directed to investigate unresolved severe accident issues such as core debris coolability, steam explosions, and hydrogen combustion both experimentally and numerically. Extensive studies have been performed to evaluate the in-vessel retention of core debris through external reactor vessel cooling concept for APR1400 as a severe accident management strategy. Additionally, an improvement of the insulator design outside the vessel was investigated. To address steam explosions, a series of experiments using a prototypic material was performed in the TROI facility. Major parameters such as material composition and void fraction as well as the relevant physics affecting the energetics of steam explosions were investigated. For hydrogen control in Korean nuclear power plants, evaluation of the hydrogen concentration and the possibility of deflagration-to-detonation transition occurrence in the containment using three-dimensional analysis code, GASFLOW, were performed. Finally, the integrated severe accident analysis code, MIDAS, has been developed for domestication based on MELCOR. The data transfer scheme using pointers was restructured with the modules and the derived-type direct variables using FORTRAN90. New models were implemented to extend the capability of MIDAS.

KEYWORDS : Severe Accidents, In-vessel Retention, Steam Explosion, Hydrogen Combustion, Severe Accident Analysis Code

1. INTRODUCTION

The development of nuclear power plants is essential for continuous growth of the economy and energy self-sufficiency in Korea, where energy resources are limited. For the development of nuclear power plants (NPPs), security of the public safety, in terms of risks from reactor accidents, is the first thing to be considered. After the Three Mile Island 2 (TMI-2) accident, however, the public voiced much concerned about severe accidents-the core meltdown and subsequent release of radioactive fission products to the surrounding environment. Since then, many experimental and analytical research has been carried out to evaluate the consequences of severe accidents.

In Korea, severe accident research has been promoted since 1992 when the government launched and supported a long-term nuclear R&D program. The research program focused on enhancing technological research capabilities and on the resolution of severe accident issues for Korean NPPs.

During the early stages of the program, research focused

on the small scale separate effect experiments and model development for major severe accidents including high-pressure melt ejection (HPME), resulting in direct containment heating (DCH), and models for reactor vessel lower head failure analysis [1]. The severe accident research program later shifted to study the development of assessment techniques and accident management strategies for reduction of the risk through maintaining the integrity of the reactor vessel and the containment. In support of this, two experimental programs, *Simulation of Naturally Arrested Vessel Attack-In Vessel* (SONATA-IV) and *CONtainment integrity Verification EXperiment* (CONVEX) were performed [2,3].

After the government issued a Severe Accident Policy Statement and requested for the utility to implement a severe accident management (SAM) strategy and provide the engineered safety features for the existing and future developing reactors to mitigate the risk of potential severe accidents, the Korean nuclear industry has established a severe accident management plan for the operating plants and considered the development of

engineered safety features to mitigate the severe accident for the future reactors. The long-term nuclear R&D program, thus, aimed to establish the optimized severe accident management methodology and technical basis to resolve the remaining unresolved severe accident issues such as core debris coolability, steam explosion and hydrogen combustion issues through the analysis and experiments.

To ensure the public safety against severe accidents, the maintenance of the integrity of both reactor vessel and containment is a key. To maintain reactor vessel integrity, the Korean Advanced Power Reactor (APR1400), for instance, has adopted in-vessel retention (IVR) of core debris through an external reactor vessel cooling (ERVC) strategy. Significant amounts of decay heat from relocated core material can be removed through the vessel wall by flooding the reactor cavity which surrounds the reactor vessel. Such a strategy was implemented in a few light water reactors (LWRs), such as Loviisa 1&2 and Westinghouse AP600. For the latter, it was found that there was sufficient margin between the critical heat flux on the water side and the incident heat flux from the uniform corium pool [4]. The margin of safety, however, can decrease, when, for instance, metal layers are present on top of the oxidic corium pool. For the higher power AP1000 and APR1400 reactors, the safety margin is even more reduced due to higher thermal loads on the reactor vessel wall from the molten pool. The molten pool configuration and melt composition affects heat flux distribution on the reactor vessel wall. Recent experimental results of the OECD sponsored MASCA projects showed more complicated melt pool compositions and configurations, such as the three-layer molten pools, to be present [5]. To evaluate IVR-ERVC strategies for APR1400 severe accident management, extensive studies have been performed. In particular, improvements of the insulator design outside the vessel were investigated-ERVC flow paths were studied in scaled experiments of the natural circulation loop. A detailed analysis of in-vessel melt progression in APR1400 reactors using the SCDAP/RELAP5 computer code was also carried out.

During a severe accident, the molten core material can relocate to the bottom of the reactor vessel and be discharged into the reactor cavity after reactor vessel failure. When the reactor vessel lower head or the reactor cavity becomes filled with water, hot molten material can interact with coolant so as to lead to rapid heat transfer from the hot molten material to the surrounding cold water resulting in violent steam generation. This phenomenon is called a *steam explosion*, or a fuel-coolant interaction (FCI) [6,7]. Over the past few decades, intensive research has led to the conclusion that the probability of alpha-mode failures is very low and that the consequence of fuel-coolant interactions does not seem to be as catastrophic as previously thought [8]. Yet, ex-vessel steam explosion issues still remain unresolved. Adoption of an IVR as a

SAM strategy in Korean nuclear power plants [9] posed a safety issue in terms of steam-explosions. It was noted that the ex-vessel steam explosion could be strong enough to threaten structural integrity of the containment or results in a failure of containment penetrations [10].

To investigate this issue, the Korea Atomic Energy Research Institute (KAERI) launched an experimental research program in 2000 [11]: *Test for Real cOrium Interaction with water* (TROI). A series of experiments was performed in the TROI facility using a prototypic material consisting of a mixture of UO_2 , ZrO_2 , Zr, and stainless steel. More than 50 experiments using a molten corium in the range of 5~20 kg were performed. Major parameters such as material composition and void fraction as well as the relevant physics affecting the energetics of steam explosions have been investigated.

Hydrogen combustion issues are a major factor that influencing the safety of nuclear power plants. The large amount of hydrogen generated during severe accidents can result in explosions that can directly load to the containment building, internal structures, and equipment. Much effort has been directed towards controlling hydrogen combustion in the containment during severe accidents [12]. Igniters, which need electric power for operation, have been installed as a method to prevent destructive burns in nuclear power plants by ensuring ignition near the limits of flammability [13]. Meanwhile, passive autocatalytic recombiners (PARs), which recombine the hydrogen present in the containment without the need for external power or operation, are alternatively installed in the European Pressurized Water Reactor (EPR) [14]. APR1400 employs a combination of igniters and PARs for hydrogen control [15]. The radical objective of these hydrogen control systems is to control the hydrogen concentration in the containment building, thereby preventing the deflagration-to-detonation transition (DDT).

Two research activities have been performed for controlling hydrogen in Korean advanced NPPs. The first estimated hydrogen concentration, using the three-dimensional analysis code, GASFLOW, in severe accidents and evaluated DDT probabilities in the containment. The second focused on development and design countermeasures, such as quenching meshes, that allow operators to avoid flame acceleration and DDT.

Among many integrated severe accident computer codes, MELCOR and MAAP from USA and ASTEC from Europe are popular owing in part to the maturity and ease of use. Their development for current and next generation reactors analysis is on-going. In Korea, MAAP and MELCOR are also being used for integral safety analysis with input manipulation. In the meantime, there has been a need to develop computer codes in order to reservoir the domestic analytical and experimental data since the long term R&D project inception. In-house development of the code was expected to lead to a

thorough understanding of the models and foster national expertise. Considering the code-release schedule and the applicability to various configurations of plants, MELCOR 1.8.4 was selected as a reference and served as a basis for MIDAS development. The integrated computer code can be used to simulate overall accident progression and to suggest the initial and boundary conditions to the specific issues rather than to resolve the issues directly.

2. INVESTIGATION ON IN-VESSEL PHENOMENA

Severe accidents in PWR typically begin with core uncover initiated by loss of reactor coolant and failures of reactor safety systems. In-vessel melt progression can be divided into the several stages depending on thermal-hydraulics and material interactions:

- Inadequate core cooling with an initiating event
- Core heat up with core uncover,
- Onset of exothermic oxidation of cladding,
- Onset of clad melting and fuel liquefaction,
- Slumping of molten material into the lower plenum,
- Failure of the reactor vessel bottom head.

The rise in core temperature due to insufficient cooling can lead to exothermic zircaloy oxidation transients, which deliver heat to clad and fuel at a very high rate, and a large amount of hydrogen is produced and released to the containment. As core temperature rises, melting begins with the structural and control rod materials and progresses to clad, fuel eutectic, and fuel. The core boundary can become severely deformed, and melt pools may form as was the case with the TMI-2 reactor. Eventually, the molten core material may be discharged or slumps to the lower plenum, and can thermally and mechanically attack the core support structure. Failure of the core plate or core barrel brings corium to the lower head.

When molten core material comes into contact with coolant, a variety of different FCIs can occur. They can range from quiescent boiling to explosive fragmentation of the fuel with rapid steam generation. If the vessel is not breached by steam explosions, a fraction of the core melt may be quenched in the lower head. Subsequent reheating with the decay heat would occur. Even with forced circulation, melting in the interior of a fragmented debris bed can occur if the dryout heat flux is below the decay heat. A molten pool would form, and the melt could attain a very high temperature. Natural convection in the molten pool transports energy downward and radially to the vessel lower head. Failure of the bottom head would occur when the temperature of the steel increased to the point where the stress level exceeded the strength of the material. The mode of vessel breach can strongly influence the timing and nature of potential loads imposed on containment. It can also influence subsequent concentrations of fission product, particularly in aerosol form, in the containment.

Over the years, much experimental research has investigated the early phase of in-vessel melt progression. Representative experimental programs are CORA, PBF, LOFT, and PHEBUS. The state of knowledge with respect to early in-vessel melt progression confirms the picture conveyed by TMI-2; however, late phases of in-vessel accident progression has not received much attention. There exist several severe accident analysis codes: SCDAP/RELAP5, ATHLET/CD, MELCOR, and ASTEC. A number of code validation programs, such as the OECD/NEA international standard problems (ISP) on in-vessel core degradation problems of CORA and PHEBUS and the TMI-2 benchmarking problems are performed [16,17]. However, care must be taken in using and interpreting these calculations because such codes can never be fully validated.

The stabilization and termination of severe accidents through maintaining the coolability of the degraded core is a main target of severe accident management. In KAERI, most of the studies of in-vessel phenomena are focus on this aspect. Particular focus is on the implementation of severe accident management for the Korean Optimized Power Reactor (OPR1000) and APR1400.

One element in the stabilization of a damaged core is the potential for introducing coolant into a damaged core as occurred at TMI-2. Coolant injection into a damaged core is likely to occur under certain circumstances, for example, when lost power is restored. If water is reintroduced early enough and if the fuel rods have maintained their original geometry, cooling of the core is reasonably assumed. However, reintroduction of coolant at later times can result in unforeseen circumstances. The in-vessel melt and debris configuration for coolability depends on the melt progression and the recovery time of the accident. As part of the development of accident management strategies, the in-vessel accident sequence analysis with and without water injection into the vessel have been performed for OPR1000 and APR1400 using SCDAP/RELAP5 [18,19].

Another severe accident management measure is IVR of core debris through ERVC. By flooding the reactor cavity that surrounds the vessel, significant energy can be removed from relocated corium material through the vessel wall via nucleate boiling on the vessel outer surface. Theofanous et al. (1995) found that, for the AP600 design with a uniform corium pool, there was sufficient margin between the critical heat flux on the water side and the incident heat flux from the corium pool [4]. The margin of safety, however, may decrease in case there is a metal layer present on top of the oxidic corium pool. For the higher power reactors, the safety margin is significantly reduced due to higher thermal loads on the reactor vessel wall from the molten pool. Extensive studies have been performed to evaluate the IVR-ERVC concept for APR1400 as an accident management strategy. Borne out from those studies, which investigated scaled experiment of the natural

circulation loop, improvements in the insulator design outside the vessel were achieved, that provided for an effective ERVC flow path [18,19].

2.1 Detailed Analysis of In-vessel Melt Progression of APR1400 and OPR1000

Accurate prediction of in-vessel melt progression is essential for determining effective ways to terminate core degradation (as an accident management strategy) and also to estimate the final state of the molten core at the vessel breach as an initial condition of the ex-vessel melt progression in the containment. Detailed analysis of the in-vessel melt progression of APR1400 has been performed using the SCDAP/RELAP5 code [20]. It has allowed for

predictions of the timing of major stages in accident progression and end states of core degradation. According to the Level 1 probabilistic safety assessment (PSA) results of APR1400, dominant severe accident sequences include total loss of feed water (LOFW), a small break loss of coolant accident (SBLOCA) without safety injections (SI), a station blackout (SBO), a medium break LOCA without SI, and a large break LOCA without SI. Tables 1 and 2 summarize the timing of the major events in in-vessel melt progression for high and low pressure sequences, respectively.

For high pressure sequences, pressurizer surge-line failures from hot steam emanating from the core have been considered as a separate sequence that induces

Table 1. Significant Events of the High Pressure Sequences in the APR1400 (unit: sec)

vents	Total LOFW		SBO
	Without RCS Depressurization	With RCS Depressurization	
Transient Initiated	0	0	0
Actuation of the POSRV	-	1,109	-
Core Uncovery	2,568	1,672	5,574
Actuation of the SITs	(4,730)*	1,924	(8,304)*
Core Melting	3,856	7,942	7,073
Surge Line Failure	(4,605)*	-	(8,180)*
First Relocation of Molten Core Material to the Lower Plenum	7,685 (10,387)*	8,958	11,899 (14,267)*
Reactor Vessel Failure	9,109 (11,500)*	10,305	13,242 (15,500)*
Total Oxidation Fraction (%)	54.8 (55.5)*	38.5	56.0 (56.1)*

Note) () *: Surge Line Failure Case

Table 2. Significant Events of the Low Pressure Sequences in the APR1400 (unit: sec)

Events	Small Break LOCA (0.0009 m ²)	Medium Break LOCA (0.0093 m ²)	Large Break LOCA (0.0465 m ²)
Transient Initiated	0	0	0
Core Uncovery	4,622	748	1,808
Actuation of the SITs	8,580	996	190
End of SIT Actuation	-	5,850	988
Core Melting	5,817	7,120	2,675
First Relocation of Molten Core Material to the Lower Plenum	17,875	8,170	3,306
Reactor Vessel Failure	24,400	9,550	4,910
Total Oxidation Fraction (%)	49.1	32.9	9.3

depressurization of the primary coolant system. In the LOCA sequences, the safety injection (SI) system is not actuated, but it is assumed that the safety injection tank (SIT) is working depending on the pressure variation of the system because of its passive feature. In all the sequences, approximately 80-90% of the core material was melted and relocated to the lower plenum of the reactor vessel at the time of the reactor vessel failure. The timing of the vessel breach was influenced by depressurization and the availability of safety injection of SIT depending on the sequences.

For the development of the in-vessel strategies in the severe accident management guidance (SAMG) for OPR1000, in-vessel melt progression has been analyzed using SCDAP/RELAP5/Mod3.3 [21]. Best estimate calculations of the small break LOCAs of 3.429 cm and 5.08 cm, the medium break LOCAs of 7.62 cm and 10.871 cm, and a large break LOCA of 24.892 cm have been performed without SI up to the time of the reactor vessel failure. Table 3 summarizes the occurrence time of the major events predicted by SCDAP/RELAP5. In the

small and large break LOCAs, the reactor vessel failed at a relatively early time of approximately 70~110 minutes after the transient. On the other hand, in the medium break LOCA, the reactor vessel failed at a later time of approximately 200~400 minutes after the transient because coolant in the SIT was used more effectively to cool down the degraded core.

Related to safety injection of the coolant during core degradation, depressurization of the reactor coolant system (RCS) is a prerequisite accident management strategy, especially for high pressure accident sequences. Also, it is essential to prevent the DCH which provokes the early containment failure. Depressurization can be achieved by the secondary feed and bleed operation and also directly by using a safety depressurization system (SDS). In OPR1000, eight condenser dump valves (CDV) and four atmosphere dump valves (ADV) for the secondary depressurization are installed in the main steam line. Two safety depressurization system valves for the direct RCS depressurization are installed at the top of the pressurizer. In the SAMG of OPR1000, the entry condition of severe

Table 3. Timing of Significant Events of LOCA Sequences in OPR1000 (unit: sec)

Events	3.428 cm	5.08 cm	7.62 cm	10.871 cm	24.384 cm
Transient Initiated	0	0	0	0	0
Core Uncovery	4,208	2,165	937	466	1,928
Actuation of SITs	-	3,546	1,546	750	150
End of SIT Actuation	-	-	23,644	-	3,238
Core Melting	5,525	3,225	1,768	10,538	3,102
First Relocation of Corium to the Lower Plenum	6,425	3,623	23,650	12,088	3,657
Reactor Vessel Failure	6,700	4,336	24,075	12,580	4,700
Total Oxidation Fraction (%)	53.2	45.7	53.9	45.2	20.1

Table 4. RCS Pressure at a Reactor Vessel Failure for the Small Break LOCA without a SI

Case	SIT Actuation Time (s)	Reactor Vessel Failure Time (s)	RCS Pressure at Reactor Vessel Failure (MPa)
Base	-	6,330	6.72
CDV1- 1 minutes	5,032	33,500	3.65
CDV1- 6 minutes	5,328	34,030	3.34
CDV8 + ADV4- 6 minutes	5,050	44,780	3.41
CDV1- 7 minutes	6,494	6,775	4.2
CDV8 + ADV4- 7 minutes	6,580	7,525	3.04
CDV1- 6 minutes, SDS1-32500	5,328	33,095	2.25 < (2.9 MPa)

accident mitigation is when the core exit temperature reaches 650 °C. Table 4 shows the SCDAP/RELAP5 results of reactor vessel failure time and the RCS pressure at that time for the small break LOCA of 3.429 cm with different depressurizations, where operation of the safety injection system was not assumed [22]. The actuation timing of the depressurization systems was changed from the entrance time of SAMG, and the depressurization capacity was changed from 1 to 8 CDVs and 1 to 4 ADVs. In the table, *CDV1-1 minute* refers to the condition of one condenser dump valve being opened 1 minute after the SAMG is initiated. The SCDAP/RELAP5 results indicate that the secondary feed and bleed operation can depressurize the RCS, but it cannot depressurize the RCS pressure sufficiently. For this reason, direct RCS depressurization using the SDS is necessary for a small break LOCA without SI. Additionally, proper timing and capacity of depressurization is crucial to delay the reactor vessel failure.

To terminate in-vessel melt progression and cool-down of the degraded core, coolant should be supplied into the reactor vessel as an accident management strategy. This can be achieved by the safety injection system via RCS depressurization. Detailed analysis of coolant injection into reactor vessels with depressurization for OPR1000 has been performed by using the SCDAP/RELAP5/Mod3 [23]. The OPR1000 safety injection system is composed of two high- and two low-pressure pumps connected to the four cold legs. As a base case, a high pressure sequence of a small break LOCA without SI was selected. Table 5 shows the SCDAP/RELAP5 predictions on the timing of

significant events for the SBLOCA for the case of one CDV opening after 6 minutes and SI injection with different actuation times after the transient. One train operation of the high pressure safety injection (HPSI) at 30,696 seconds with RCS depressurization by using one CDV at 6 minutes after the entrance condition of SAMG, prevents the reactor vessel failure, in spite of the fuel melting at 30,410 seconds after transient. Three train operations of HPSI at 31,296 seconds prevented reactor vessel failure- 31,996 seconds was too late. The timing and capacity of safety injection combined with depressurization is critical for the success of the prevention of reactor vessel failures and the stabilization of the in-vessel core degradation during severe accidents.

2.2 Experiment of Natural Circulation Flow in the Reactor Cavity for IVR-ERVC

The APR1400 design incorporates ex-vessel flooding as a strategy for preventing vessel failure. Plant designs initially considered a small gap between the vessel and its insulation. To ensure critical heat flux outside the reactor vessel (for successful ERVC), the design was modified to have a minimum gap size of ~15 cm. IVR evaluations were performed by the Korea Hydro & Nuclear Power Company (KHNP). They focused on identifying possible end states of relocated material considering variations in composition, geometry, and physical state that can impact the magnitude and distribution of heat flux to the lower head [24]. To resolve the IVR issue, more specific issues on ERVC should first be considered.

Table 5. Effect of the HPSI Actuation Time and Capacity on Significant Events

Case	SIT Actuation Time, HPSI Actuation Time (sec)	First Relocation Time to the LP (sec)	Large Relocation to the LP (sec)	RV Failure Time (sec)	RCS Pressure at RV Failure (MPa)
Base	n/a - n/a	6,222	6,222	6,330	6.72
CDV1-6 min, LPSI3	5,328 - n/a	38,531	39,967	40,200	3.30
CDV1-6 min, HPSI1-30000 sec	5,328 - 30,000	Fuel Melting Time = 30,359 seconds, No RV Failure			
CDV1-6 min, HPSI1-30696 sec	5,328 - 30,696	Fuel Melting Time = 30,410 seconds, No RV Failure			
CDV1-6 min, HPSI1-31296 sec	5,328 - 31,296	32,454	32,454	32,610	7.56
CDV1-6 min, HPSI3-30000 sec	5,328 - 30,000	No Fuel Melting, No RV Failure			
CDV1-6 min, HPSI3-31296 sec	5,328 - 31,296	Fuel Melting Time = 30,397 seconds, No RV Failure			
CDV1-6 min, HPSI3-31996 sec	5,328 - 31,996	33,177	33,177	33,330	5.38

These include insulation performance, vessel-instrument tube penetration, availability of sufficient flood water, potential for inadvertent flooding, and impact of flooding on other plant equipment.

As a part of a study on the thermal hydraulic behavior in the reactor cavity under the ERVC in APR1400, a large-scale air injection experiment (Hydraulic Evaluation of Reactor cooling Mechanism by External Self-induced flow-HALF scale or *HERMES-HALF*) was performed to characterize the two-phase natural circulation flow through the annular gap between the reactor vessel and the insulation material [25]. Based on the previous forced convective experiment of BALI, an increase of the mass flow rate of the coolant in the annular gap can lead to an increase in the CHF at the lower outer reactor vessel wall. The experimental results have been analyzed using the thermal hydraulic analysis code of RELAP5/MOD3.

The schematic diagram of the HERMES-HALF experimental facility is shown in Fig. 1. The facility consists of 3 parts: main test section, air supply system, and water recirculation system. The test section is a half scaled-down reactor vessel with insulation part, utilizing the scaling law proposed by Cheung. Fig. 2 presents the experimental results of the water recirculation mass flow rate with a change of the water inlet area from 0.0043 to 0.15 m² and comparisons with the RELAP5 results. Increases in the water inlet area and the water level in the reactor cavity led to increases in the water circulation mass flow rate. The effects of the air injection mass flow rate and water

outlet area on the water circulation mass flow rate were dependent on the water inlet area in experiment. As the water outlet moved to a lower position, the water circulation mass flow rate slowly increased. For the real accident condition in full scale, however, the water circulation mass flow rate is estimated with the RELAP5 calculations for APR1400.

Fig. 2. RELAP5/MOD3 Results for the Water Circulation Mass Flow Rate as a Function of the Water Inlet Area

Fig. 1. Schematic Diagram of the HERMES-HALF Experimental Facility

2.3 Remaining Issues and Issue Resolution

A fundamental problem in understanding core melt progression is that it is extremely difficult to perform experiments necessary to fully understand the relevant phenomena. Over the years, much research has investigated in-vessel melt progression. Contrary to late phases, early phases of in-vessel accident progression are relatively well-understood. Recently, more generic investigations have been pursued in the European Union (EU) project in which major uncertainties were addressed. The EU work program of *Severe Accident Research Priorities* (SARP) identified major in-vessel considerations which took into account both risk potential and severe accident issues where large uncertainties still persist, which includes [26]:

- Hydrogen generation during reflood or melt relocation into water,
- Core coolability during reflood and thermal hydraulics within particulate debris,
- Corium coolability in lower head and external corium catcher device,
- Integrity of RPV due to external reactor vessel cooling,
- Integrity of RCS,
- Corium release following vessel failure.

Currently, integral efforts in understanding in-vessel phenomena and developing analysis codes are underway through international collaborations such as those of the OECD/NEA program and the EU framework. In addition, for the development of in-vessel accident management strategies for domestic plants, the plant specific accident analysis using the severe accident analysis codes, such as SCDAP/RELAP5, is required to investigate various accident scenarios.

As the APR1400 advanced reactor is now under construction as the Shin-Kori 3&4 unit, the IVR-ERVC issue should be clarified to make a role of severe accident mitigation measure. The recent experiments in the OECD sponsored RASPLAV and MASCA projects highlighted more complicated melt pool compositions and configurations such as the three-layer molten pool. Much knowledge has been gained, but it is still argued that there are some gaps and considerable uncertainties. With the external flooding in ERVC, a strategy of simultaneous flooding of the metal layer with water inside the vessel could be a solution in APR1400 to remove sufficient heat from the upper surface of the metal layer. It also could reduce the focused heat flux to value less than the critical heat flux (CHF), which also need to be verified.

3. STEAM EXPLOSION EXPERIMENTS USING TROI FACILITY

Adoption of In-Vessel Retention strategy as a severe accident management strategy in the Korean nuclear power plant [9] posed a safety issue of the risk of steam explosion,

since it was indicated that the ex-vessel steam explosion could be strong enough to threaten the structural integrity of the containment or results in a failure of containment penetrations [10].

This led to a launch of an experimental research program at KAERI in 2000 entitled TROI [11]. A series of experiments using a prototypic mixture of UO_2 , ZrO_2 , Zr, and stainless steel was performed in the TROI facility. More than 50 experiments using a molten corium in the range of 5~20 kg were performed. Here, a brief description of the test facility, configuration of experiments, major results, and significance of the experimental results and direction for further research will be discussed.

3.1 Description of test

3.1.1 The Cold Crucible Melting Technique and Delivery of a Melt Jet

A cold crucible technique was adapted for the melting of the corium and delivery of a melt jet into a water pool. A detailed description of the technique is described in Hong et al. [27]. A typical charging pattern for corium is shown in Fig. 3. A mixture of UO_2 pellets and ZrO_2 powder was used to fill the cold crucible in several layers. The heated zirconium ring was used as an initiator.

Fig. 4 shows the melting performance for the TROI-28 test typical of other corium tests. The curves show the melt temperature measured by a two-color pyrometer, electric power input to the generator, the power absorbed into the melt, and Q-factor, which is defined as ratio of stored energy in the condenser to the input power. A low

Fig. 3. A Typical Charging Pattern

Fig. 4. Melting Performance Curve for TROI-28

Fig. 5. Shape of the Crucible

Fig. 6. Bottom of the Crucible

Q-factor represents efficient melt-electromagnetic field coupling. The curve indicates that the input power was not efficiently absorbed into the melt during the first 1,800 seconds. During the period of 1,800~2,500 seconds, absorption efficiency increased substantially. This suggests that the amount of melt in the crucible increased enough to absorb the electrical energy effectively. The efficiency was about 70%. After 2,500 seconds, the increased input power resulted in an increase in melt temperature and heat losses due to radiation and conduction from the molten corium pool to the surface of the cold crucible, which was maintained at room temperature.

An IRCON two-color pyrometer was used to measure the melt temperature. It was first detected once a certain amount of melt had formed, which was indicated by a decrease in the Q-factor. Then, the measured melt temperature rapidly increased due to oxidation of the zirconium ring. As zirconia powder is placed near the zirconium ring, the temperature reaches the melting point of zirconia. There was then a slight decrease in the melt temperature.

While power input and absorbed power increased, the melting zone expanded and the heat loss increased. Depending on the balance between the power input and heat loss, the melt temperature reached a quasi-steady state. As the power input increased, the thickness of the bottom crust decreased, while the melt temperature reached a quasi-steady state. It was necessary to minimize the thickness of the bottom crust for easy punching and melt delivery.

A typical shape of the cold crucible after the delivery of the melt is shown in Figs. 5 and 6. In the former, a small amount of molten material erupted through the gas discharge hole during the melting period. Below the upper crust, which was made of sintered zirconia powder, a pool of molten corium formed. Between the molten corium and cold crucible, a shell made of crust formed. At the bottom of the cold crucible, a plug closed the hole during the melting period. After melting completed, the plug was removed. Then the shell-like crust held the pool of molten corium. The puncher was actuated to break the crust (Fig. 6). The equivalent diameter of the hole was about 5 cm. The thickness of the bottom crust was less than 1 mm.

3.1.2 A Typical Configuration of the Test

Steam explosion tests were performed in the TROI facility. Fig. 7 shows a typical configuration of the test. The tests were performed by pouring the molten core mixture of UO_2 , ZrO_2 , Zr , and stainless steel at temperature near 3,000 K into a pool of water at room temperature and atmospheric pressure. As it is quite similar to the configuration of ex-vessel steam explosions, the results can be directly used to evaluate the risk of steam explosions in reactors. An external trigger using the pentaerythritol tetranitrate (PETN) chemical explosive was used.

Fig. 7. A Typical Configuration of Steam Explosion Test in TROI Facility

A rigid cylindrical stainless steel test section was used in the test. The height of the test section was 150 cm, and the inner diameter was 60 cm. The thickness of the wall was 2 cm, and the thickness of the bottom was 3 cm. At the bottom of the test vessel, a load cell was installed to measure the dynamic impulse. The depth of the water pool could vary. When the depth was 130 cm, the water mass was 367 kg.

An optical two-color pyrometer (3R-35C15-0-0-0-1, 1500–3500 °C) was used to measure the melt temperature in the cold crucible during melting through a quartz window from the top. An argon gas supply system was used to prevent optical noise by some participating media (aerosols) in the optical path. Another IRCON optical two-color pyrometer directed the melt delivery path just below the quick-opening valve to measure the melt temperature at the time of a melt delivery through a quartz window.

A number of K-type (Chromel-Alumel) thermocouples

were installed in the interaction vessel to measure the water temperature (IVT101~IVT104) and to measure the melt velocity (sacrificial thermocouples: IVT201~IVT209). They were also installed in the pressure vessel to measure the ambient temperatures (PVT001~PVT005) and to measure the water temperatures in the inlet and outlet cooling lines of the induction coil.

Piezoelectric static pressure transducers (Keller PA23S, < 1.0 MPa) were installed to measure the static pressure during the melt/water interaction in the furnace vessel (FVSP001) and in the pressure vessel (PVSP002 and PVSP003). Two types of piezoelectric dynamic pressure sensors were installed to measure the dynamic pressure in the interaction vessel (PCB piezotronics inc. model 112A03 < 69 MPa, PIVDP101~PIVDP104, KISTLER model 6005 < 100 MPa, KIVDP101~KIVDP104). Dynamic pressure sensors were installed in the pressure vessel to measure the ambient dynamic pressure (KISTLER model

6061B < 25 MPa, PVDP004 and PVDP005).

Various physical and chemical analyses of the ingots from the melting experiment and the post test debris from the steam explosion experiment were performed [28]. A scanning electron microscope (SEM) was used to examine the morphology of the debris. The wet analysis using inductively coupled plasma-atomic emission spectrometry (ICP-AES) and an elementary analysis using an electron probe micro analyzer (EPMA) were performed to identify the chemical composition of the debris and ingot. x-ray diffraction (XRD) patterns were measured for phase characterization.

3.2 Major Findings from the Results of Experiments

There have been more than 50 experiments in the TROI facility using prototypic materials, where the material composition, water pool depth and width, amount of sub-cooling for the water, ambient pressure, and amount of melt super heat were changed. The main focus of the steam explosion experiments was to identify major parameters and physics affecting the energetics of prototypical steam explosions in reactors. Therefore, it was very important to represent the prototypic situation in the TROI experiment.

Since many factors affect the energetics of steam explosions, we needed analytical tools that able to extrapolate experimental results to real-reactor configurations. Noting that there have been extensive efforts for the development of analytical tools that have proven useful in simulating prototypical steam explosions [29-34], we can assume that issue of scaling of the physics of steam explosions to be well-supported.

The TROI test was designed to model molten corium being poured into a pool of water-prototypic of in-vessel and ex-vessel steam explosions. The amount of melt mass and equivalent melt jet diameter were important scaling parameters. The melt jet diameter and mixing zone size are relevant to the selection of the width of the test section containing the water.

In the TROI test the initial melt jet diameter was designed to be in the range of 2~10 cm, and the width of the test section was either 30 cm or 60 cm. It turned out that, for this range of melt jet diameter, melt pools 60 cm in diameter are more favorable in terms of increasing the strength of steam explosions [35]. The required amount of melt mass depends on the depth of the water pool, as the length of melt jet should be bigger than the water pool depth. The depth of water pool was in the range of 60~130 cm. The amount of melt mass was in the range of 5~30 kg. This scale was believed sufficient to preserve major steam explosion physics in reactors.

3.2.1 Observation of Spontaneous Steam Explosion in Case of Prototypic Material

All tests using pure oxidic corium conducted thus far did not lead to energetic steam explosions, while tests using

stimulant material, such as tin or alumina, resulted in energetic spontaneous steam explosions [38,39]. Energetic steam explosions only occurred in the presence of external triggers, when prototypic material was used. As there is no clear explanation for these phenomena yet, the energetics of corium-water interactions at low pressure remain a generic issue for reactor safety.

A series of tests using a mixture of UO_2/ZrO_2 were performed in the TROI facility to revisit the issue. Among initial 5 tests, energetic spontaneous steam explosions occurred in three cases. The initial conditions and results of the five experiments are shown in Table 6.

The dynamic pressure and impulse were measured in TROI-13 test. Fig. 8 shows the dynamic pressures in the

Fig. 8. Dynamic Pressure of TROI-13

Fig. 9. Dynamic Impulse of TROI-13

Table 6. Initial Conditions and Results of Corium Experiments

TROI test number		Unit	9	10	11	12	13
		Date	6/27/01	9/6/01	9/20/01	10/11/01	10/22/01
Melt Test Section	Charge Composition UO ₂ /ZrO ₂	[w/o]	70/30	70/30	70/30	70/30	70/30
	Charged/Released mass	[kg]	14.2/4.3	13.7/8.7	13.7/9.2	13.7/8.4	13.7/7.7
	Initial Height	[cm]	90	67	67	67	67
	Initial Temperature	[K]	296	298	296	293	292
Pressure Vessel	Initial pressure(air)	[MPa]	0.1	0.117	0.111	0.11	0.108
	Free volume	[m ³]	8.032	8.032	8.032	8.032	8.032
Results	Maximum PV pressurization	[MPa]	0.01	0.012	0.01	0.008	0.016
	Time to reach peak	[sec]	18	2.5	3	2	<1
	Maximum PV heat-up	[K]	70	17	50	14	23
	Time to stabilize	[sec]	15	15	20	<7	<1
	Maximum water heat-up	[K]	6	48	20	17	24
	Time to reach peak	[sec]	18	3	12	<1	<1
	Steam explosion		NO	SE	NO	SE	SE
	Dynamic pressure peak	[MPa]	-	?	-	1.0	7.0
	Duration	μsec	-	-	-	1ms	1ms
	Impulse	KN	-	-	-	210	250
	Duration	μsec	-	-	-	16ms	15ms
Debris	Total	[kg]	4.325	8.675	9.185	8.360	7.735
	>6.35mm	[kg]	0.09	0.435	1.49	0.295	0.620
	4.75mm ~ 6.35mm	[kg]	0.4	0.725	1.365	0.390	0.245
	2.0mm ~ 4.75mm	[kg]	2.6	3.39	4.8	3.385	2.675
	1.0mm ~ 2.0mm	[kg]	0.815	2.025	1.25	1.715	1.225
	0.71mm ~ 1.0mm	[kg]	0.18	0.64	0.235	0.580	0.540
	0.425mm ~ 0.71mm	[kg]	0.14	0.705	0.04	0.805	0.965
	<0.425mm	[kg]	0.1	0.755	0.05	1.190	1.465

interaction vessel; the measured peak pressure was 7.0 MPa. After the puncher, it took 1.22 seconds for onset of the energetic steam explosion. The duration of highest dynamic pressure was about 1 ms. Fig. 9 shows the dynamic load beneath the interaction vessel. The dynamic load was measured to be higher than 250 kN, the upper limit of the dynamic load cell and then cut off. The duration of the dynamic load was about 15 ms. As the impulse is a kind of integral behavior of water inside the interaction vessel, the shape of impulse and dynamic load is rather different.

It is interesting to consider the timing of melt jet water contact and initiation of steam explosion. As the hole was maintained in the upper crust, the molten pool was at atmospheric pressure. Therefore, the melt jet was delivered

by gravity alone. Then the time to hit the water surface was calculated to be 1,128 ms using high speed video information. If we assume the melt not to be decelerated after entry into the water pool, the time to hit the bottom is 1,206 ms. Fig. 7 indicates that the steam explosion occurred at about 1,220 ms. This indicates that it took about 0.1 second to form a pre-mixture and initiation of steam explosion after the melt entry into the water. Though it is not certain, it is possible that the steam explosion might have occurred before the melt hit the bottom.

These findings on spontaneous steam explosions are significant and have far-reaching implications. Before these findings, arguments on the non-explosive nature of the prototypic material for the FARO and KROTOS tests were two folds. First, there was assumed to be no difference

between the alumina/water and corium/water systems, as they are basically volatile liquid/liquid systems where one liquid has very high thermal energy compared to the other. This argument ignores other material properties, and it recommends the use of conservative estimates based on the alumina tests at KROTOS for the prediction of steam explosion risks in reactors. The second argument considered the existence of *unknown effects* such as the generation of excessive hydrogen in the case of corium during corium water interactions [7]. However, there has been no clear evidence supporting either of these arguments.

Therefore, observation of spontaneous steam explosions in the case of corium-water interactions opened a new question on the energetics of steam explosions. Thus, the following series of tests were designed to investigate the energetics of steam explosions.

3.2.2 Strong Effect of Material Composition

As of 2007, more than 50 experiments were performed at the TROI facility. There, effects of material composition, water pool depth and width, amount of sub-cooling for the water, ambient pressure, and amount of melt super heat were investigated.

Variation of these parameters probed changes in the condition of the molten corium-water pre-mixture. These pre-mixture conditions served as initial and boundary conditions for investigating the aforementioned explosions. It was expected that the selected parameters would have a strong effect on the strength of the steam explosion since the amount of corium melt and geometric configuration of these tests were largely unchanged. However, rather surprisingly, the major finding from these series of experiments was that there is a strong effect of material composition on the strength of the steam explosion (Fig. 10).

Fig. 10 shows the conversion ratio, which is the ratio between the kinetic energy of the water slug in the test

section and the initial thermal energy of the melt jet submerged in the water down to the bottom of the test section. It represents the strength of the steam explosion. The abscissa corresponds to melt composition in terms of UO_2 . The number indicates the test number. Tests 15, 24, 45, and 57 are used pure zirconia. There are other groups of tests using a $\text{UO}_2\text{:ZrO}_2$ mixture at 70:30 weight percent. This composition is close to the eutectic composition. Another group of tests was performed at $\text{UO}_2\text{:ZrO}_2$ (80:20) corresponding to the composition when a typical fuel in a pressurized water reactor is completely oxidized. FARO [36,37] and KROTOS tests were performed using this composition [18,19]. Further tests used $\text{UO}_2\text{:ZrO}_2$ ratios of 50:50 (test 29) and 90:10 (test 32).

An order of magnitude difference in the strength of the steam explosions between tests using pure zirconia, eutectic corium (70:30), and 80:20 corium were observed. Note that the tests included changes in the diameter of the test section. Additionally, some tests were triggered and were not. Though those parameters were thought to have a strong effect on the strength of the steam explosion, the effect of material composition dominated. Some tests used partially oxidized corium. When Zr metal was not present, the strength of the steam explosion was insignificant, while the tests with Zr metal present resulted in a strong steam explosions. We will discuss this in more detail in the next section.

3.2.3 Partially Oxidized Corium and Effect of Layer Inversion

The case with partially oxidized corium represents more realistic reactor conditions. As discussed before, when Zr metal was not present, the strength of steam explosion was negligible. This was also observed in the COLTELS test [40].

Recently the existence of layer inversion was reported in MASCA [41], whereby a miscibility gap is supposed to exist such that the mixture of metallic uranium and other metals is located at the bottom of oxidic corium. Thereby, for breaches at the bottom of the reactor vessel, molten metal can be discharged, first interacting with water. This situation is quite different from that of the oxidic corium and water interaction. As indicated in the ZEREX [42] experiment, metal species has a strong effect on the energetics of steam explosions. Therefore, this situation was investigated in the late series of tests.

Experiments probing melting-solidifying and fuel and coolant interaction experiments using a partially oxidized corium were performed in the TROI facility to simulate prototypic steam explosions in reactors. We discuss typical results below.

A melting and solidification test using a cold crucible was performed first. The crucible was charged with a mixture of UO_2 65 wt%, ZrO_2 11 wt%, Zr 12 wt%, and SS 12 wt%. The charged weight was 15.2 kg. The composition was chosen to be similar than that of MA3 or

Fig. 10. Estimated Conversion Ratio of the TROI Experiments

Fig. 11. The Structure of Ingot from the Cold Crucible

Table 7. The Composition of the Layers in the Solid Ingot [%]

	Zr	U	Fe	Cr	Ni	Total
Top(TL)	67.62	1.14	3.42	0.70	0.41	73.3
Middle(CM)	22.85	61.44	0.82	0.25	0.10	85.5
Bottom(CL)	20.53	33.73	30.97	6.20	2.89	94.3
MA3 metallic liquid	17.7	43.9	29.4	5.5	3.5	100
MA4 metallic liquid	15.9	32.4	37	9.3	5.4	100

MA4 of MASCA [21] such that the extraction of oxygen from the uranium oxide would result in a separation of the metallic liquid and oxidic liquid. The melting atmosphere was argon gas. The focus of the test was to confirm the layer inversion phenomena on a bigger scale and to properly define the boundary conditions for melt delivery in the case of a partially oxidized corium. The maximum melt temperature measured by the IRCON pyrometer was about 3,360 K, which was later recalibrated to be 3,092 K after considering the effect of glass.

A schematic diagram of the structure of the ingot after solidification is shown in Fig. 11. The solidified melt was separated by two layers below the top crust, which was anchored to the top portion of the cold crucible. The lower

part was silver in color with a density of 9.25 g/cm³ and mass of 2.365 kg. The upper part was black with a density of 8.57 g/cm³ and mass of 8.21 kg.

The compositions of the various parts of the ingot were determined by ICP-AES analysis and compared with the compositions observed in MASCA (Table 7). The summation of the metallic components in the bottom part is close to 100%, while those of the top and middle parts are less than 100%. Therefore, the bottom part of the lump mainly consisted of metallic components, while the middle and top parts consisted of ZrO₂ and UO₂ oxides. It was estimated that 13.34% of the total charged mass of UO₂ was converted to uranium metal.

XRD of lump samples revealed that the samples CT,

Fig. 12. Dynamic Pressures in the Water Pool

Fig. 13. Dynamic Pressures in the Water Pool

CM, and CS mainly contained ZrO_2 , UO_2 , while the sample from the bottom contained Fe_2U and ZrO_2 . These results are consistent with the observations discussed above.

The layer inversion discussed above (MASCA) was thus confirmed. The composition of TROI-49 was used in the subsequent fuel and coolant interaction tests of TROI-51 to simulate layer inversion in reactors. A test was carried out to investigate pre-mixing behavior and energetics of steam explosions for partially oxidized corium.

The molten corium inside the cold crucible can have two separate liquid layers before it is poured into the test section: metallic liquid at the bottom and oxidic liquid at the top. However, it is possible that there could be some mixing due to both electromagnetically induced convection due to the cold crucible and vortices formed during pouring of the melt though the exit of the cold crucible. The tests were performed by pouring the molten corium into a pool of water at room temperature and atmospheric pressure, which is prototypic of ex-vessel steam explosions.

Fig. 14. Morphologies of Debris for TROI-51

In TROI-51, the mass of UO_2 , ZrO_2 , Zr , and stainless steel were 8.6, 1.9, 1.7 and 1.5 kg, respectively. This led to a total charged mass of 13.7 kg. 6.3 kg of molten corium was poured into the 294 K water pool at an ambient pressure of 0.115 MPa. An energetic steam explosion with a propagation of the dynamic pressure wave was observed. The melt temperature during the melt delivery was measured from the side window located in the lower vessel using an IRCON pyrometer. Since a normal glass window was used, the measured melt temperature had to be recalibrated. The melt temperature was higher than 3,000 K during the melt delivery period.

Figs. 12 and 13 show the dynamic pressure measured on the wall of the test section at different times. As the sequence of the pressure wave in Fig. 12 is in the same direction from the external trigger to these sensors, these peaks were caused by the external trigger. The maximum pressure exerted by external triggers was 21 MPa at IVDP101. The second signal in Fig. 13 propagates in the direction of IVDP105, IVDP106, IVDP104, IVDP103, IVDP102, and IVDP101. Therefore, the steam explosion was triggered near IVDP105. The recorded maximum pressure resulting from the steam explosions was 32 MPa at IVDP103. At the time of the trigger at 1.33 s the melt passed thermocouple IVT206. Therefore, the external triggering occurred well before the melt contacted the bottom.

Morphology of the debris is shown in Fig. 14. A significant fraction of the particles are irregularly shaped. This is in contrast to the morphology observed in previous tests using oxidic corium, where the majority of the particles were spherical. The existence of a highly conductive metallic component in the particles might be responsible for this difference in the solidification behavior of an oxidic particle from that of a partially oxidized corium. The debris of the metallic material was highly irregular [43].

4. HYDROGEN COMBUSTION CONTROL

Hydrogen combustion has been known to be one of the major factors compromising safety at NPPs because the large amount of hydrogen generated under severe accidents can cause explosive reactions. This can result in direct loads to the containment building, internal structures, and equipment. (From here-on we refer to *containment building as containment* unless otherwise stated.)

At KAERI, two research activities have investigated hydrogen control in Korea advanced nuclear power plants. The first estimated hydrogen concentration using a three-dimensional (3D) numerical code for severe accidents and evaluated the possibility of DDT in the containment vessel. The second is to design countermeasures that allow operators to avoid flame acceleration and DDT.

4.1 Hydrogen Concentration Analysis

4.1.1 Background for the 3D Analysis of the Hydrogen Safety in the APR1400

For years, lumped-parameter (LP) codes such as MAAP and CONTAIN were used to study hydrogen safety in NPP containment. The LP method is very efficient for long-term simulations of accidents because it is fast and effective for establishing averaged hydrogen concentrations in each compartment of the containment. However, the local mixture quality and flow field cannot be resolved sufficiently, which is important for the assessment of flame acceleration and DDT. Recently, 3D computational fluid dynamics (CFD) codes have been used to obtain more detailed information on hydrogen dynamics in NPP containment vessels. Royle, et al. [44] used the GASFLOW [45] CFD code to analyze the hydrogen-steam distributions in the Konvoi-type NPP. Houkema et al. [46] compared the results from the LP code and CFX, a commercial CFD code and suggested that a 3D analysis is necessary to predict non-uniform hydrogen distributions.

In order to simulate hydrogen distributions during a severe accident in the APR1400 containment, the GASFLOW code was used. GASFLOW is a finite-volume code that solves time-dependent compressible Navier-Stokes equations with multiple gas species in a 3D computational domain. The code can model the two-phase effects of condensation and vaporization in the fluid mixture regions within the assumptions of the homogeneous equilibrium model (HEM). It also calculates two-phase heat transfer to and from structures by convection and mass diffusion. A main phenomenon in the NPP containment during severe accidents is convective mixing with jet flows induced by the hydrogen-steam source release augmented by a buoyancy forces and steam condensation and vaporization. Buoyancy is fully implemented in the GASFLOW code using a compressible gas formulation. The mechanical and thermal mixing in the containment is enhanced by a

turbulent flow. GASFLOW has algebraic and 2-equation k-epsilon turbulent models to simulate turbulence.

It is possible that copious amount of super-heated steam will be released into the containment during severe accidents. This released steam reduces the concentration of the hydrogen distributed in the containment, and it also acts as a thermal source. This thermal source can produce thermal loads on the structural components not only by condensation and convective heat transfer but additionally by thermal radiation. If there exists a temperature difference between the steam and containment wall, the steam is easily absorbs and emits thermal radiation because of its large opacity. Thermal radiation significantly influences the temperature and pressure distributions in the containment via increased heat transfer between the steam and the structures. In this study, radiative heat transfer was considered by solving a radiation equation with a P-1 spherical harmonic method, which was implemented in the GASFLOW code.

Generally, simulations of hydrogen dynamics in containments include the mitigation effect of passive automatic recombiners (PARs) and igniters in the 3D concentration fields of hydrogen, oxygen and steam. The recombiner and igniter models are currently available in the GASFLOW code. In the former, the hydrogen consumption rate and the heat release via chemical reactions involving hydrogen, which are generalized as the hydrogen recombination rate, are calculated from the hydrogen concentration at the recombiner inlet using performance data available elsewhere [47]. The APR1400 has 26 PARs and 10 glow-type igniters in its design specification to control or reduce hydrogen concentrations in the containment during design-based or hypothetical severe accidents [48].

4.1.2 Modeling of the APR1400 Containment for the GASFLOW Analysis

In order to analyze hydrogen dynamics in a containment using the GASFLOW code, it is necessary to model the geometry of the target containment and steam-hydrogen source discharged into the containment by an accident.

The APR1400 containment is, for the most part, cylindrical. It has a free volume of approximately 94,000 m³. Its height above the reactor cavity floor to the apex of the containment dome is 79.4 m, and its outer containment wall inner radius is 22.86 m. Although the primary shield wall and the refueling pool are nearly hexahedral, the secondary shield wall is also cylindrical. For the GASFLOW analysis of the APR1400 containment, a cylindrical coordinate system was selected to reduce the block-off computational cells and stair-shaped walls. In the radial direction 19 nodes were generated with the first node on the axis and the last point on the containment wall. To define the reactor cavity annulus accurately, the second and third node points were located on the reactor vessel outer wall and inner wall of the reactor annular cavity,

respectively. The location of the second node from the last was adjusted to fit the computational cell size to the 30.48 cm (1 foot) annular ventilation gap between the operating deck and the containment outer wall. The other nodes from $i=3$ to $i=17$ were distributed at equal distances. The nodes in the azimuthal (ϕ) direction were uniformly distributed for geometrical modeling ease. The

total number of nodes generated was 61 in 6 degree intervals. The axial or z-directional nodes were generated from the reactor cavity floor to the top of the containment dome. 53 nodes were used for the axial direction with a uniform distance. In order to consider the complexities of the flow structure and the geometry below the operating deck where the in-vessel refueling water storage tank (IRWST), annular compartment, and some valve rooms are located the axial cell distances below the operating deck are half of the ones above the operating deck. The APR1400 containment was modeled using a single block grid, and the total number of mesh cells generated was 66,960 including the boundary ghost cells (Fig. 15).

Fig. 15. Perspective View of the Surface Mesh Generated for the APR1400 NPP Containment.

4.1.3 The Results of the GASFLOW Analysis

4.1.3.1 SBLOCA Analysis

The selected accident scenario to be simulated using GASFLOW was small break LOCA with failure of safety injection, which is one of the most probable severe accidents. Initially, pressure and temperature of the atmosphere in the containment were 1 bar and 300 K, respectively. The air contained vaporized water up to 67% relative humidity at these conditions. It is stationary before water, steam, and hydrogen blow out from the cold-leg break which act as mass, momentum, and energy sources of flow field. The source can be obtained from a lumped-parameter analysis code for severe accidents such as MELCOR and MAAP. In this study, the SBLOCA was simulated using MAAP. MAAP analyzes in-vessel dynamics after an accident has begun, and it calculates blow-down rate of water, steam, and hydrogen, which is used as source data for GASFLOW calculations. Fig. 16 gives the water, steam, and hydrogen sources during 9,000 s of SBLOCA calculated from MAAP analysis. During the first 4,000 s saturated water and steam were discharged, and then steam and hydrogen were blown out thereafter. The total

Fig. 16. Mass Rates of the Water, Steam and Hydrogen Released from a Cold-leg Break

Fig. 17. Steam and Hydrogen Distributions, (a) Steam Cloud with 45 vol% at 4000 s, (b) Steam Cloud with 40 vol% and Hydrogen Clouds with 10 and 5 vol% at 6200 s

amount of water, steam, and hydrogen discharged into the containment during 9,000 s were 165,319 kg, 97,294 kg, and 637.6 kg, respectively. The temperature of the source term was modeled using time-varying enthalpies obtained from MAAP data. The discharged steam and water in the first 4,000 s are saturated and the saturation temperature was obtained from steam table using in-vessel pressure during the accident. After 4,000 s, superheated steam was discharged with hydrogen whose temperature was assumed to be equal to that of the steam.

Because temperature of the jet was very high compared to the atmosphere in the containment, buoyancy forces became large. They resulted in a strong upward flow from the cold-leg break. 4,000 s after the accident began, a steam cloud of concentration in excess of 45 volume% developed at the dome region. At 6,000 s, hydrogen started to blow out with steam. The temperature of this source jet was also high; therefore, there was very little spreading from the discharge region. The upward jet which arrived at top of the dome expelled a steam cloud down to the cylindrical part of the containment. The hydrogen concentration of the jet was approximately 10 volume% (Fig. 17).

4.1.3.2 The Effect of the PARs

The performance of the PARs for the SBLOCA is depicted in Fig. 18. Usually large-size Siemens recombiner FR90 1-1500 recombines 4 to 5 g of hydrogen per seconds. For this study, all 26 PARs were chosen as a large size

Fig. 18. Recombined Hydrogen Mass of each PAR until 9000 s

because their size is not yet defined in the design specification of APR1400. Recombiners numbered 7 to 10 were installed in the two steam generator rooms, and 21 to 24 were installed at the polar crane. The total amount of recombined hydrogen after 9,000 s was approximately 103 kg, and the inventory, which is the amount of hydrogen left in the containment, was about 534 kg. As follows from the GASFLOW analysis, the hydrogen released from the cold-leg break formed a strong upward jet in the

Fig. 19. Effect of Hydrogen Igniters, (a) Hydrogen Distribution in the Steam Generator Compartment at 6200 s when the Igniters Turned on, (b) Steam Cloud Generated by Hydrogen Combustion.

Fig. 20. Sigma Cloud Developed in the Containment, (a) without Spray, (b) with Spray

steam generator room that went directly to the upper dome through the open top of the steam generator room.

4.1.3.3 The Effect of the Igniters.

APR1400 uses a dual strategy for hydrogen mitigation:

passive PARs and active igniters. The current APR1400 design specification has 10 igniters installed in the containment. For the SBLOCA scenario chosen in this study, it is thought that igniters can effectively consume the hydrogen released from cold-leg break. Before verifying

this postulate, the GASFLOW calculation was restarted from 5,000 s with the igniters turned on. Fig. 19a shows the hydrogen concentration in the steam generator room. As shown in Fig. 19b, a standing flame was produced around the reactor coolant pump by the igniter installed in the steam generator room. The standing flame burned most of the hydrogen blown-out, and the average hydrogen concentration in the containment was below 2 volume%. Hydrogen was effectively reduced by the igniters during the SBLOCA.

4.1.3.4 The Effect of the Spray System

A spray system in a containment of a NPP is required in order for depressurizing the containment atmosphere and for removing the fission products during an accident. It is known that spray systems can affect the behavior of hydrogen in the containment during severe accidents. Therefore, spray systems in NPP containment must be considered from a hydrogen safety perspective.

In order to visualize the characteristics of the hydrogen mixture in the containment, a sigma cloud was used in this study. The sigma cloud is a volume of the hydrogen mixture with a sigma index larger than one, where the sigma index is defined as an expansion ratio of a flammable mixture by a combustion divided by its critical value which is obtained from experimental data [49]. Potential for flame acceleration exists for sigma indices larger than one. So, the size of the sigma cloud mean the volume with flame acceleration. If the size is increased, a flame, which could occur due to combustion of the hydrogen distributed in the containment, is fully accelerated, finally undergoing a DDT. In order to reduce the possibility of DDT, it is better to minimize the size of the sigma cloud when compared to a detonation cell size of the mixture cloud [50]. Fig. 20 shows that sigma clouds developed in the containment at 6,275 s when the release rate of the hydrogen was maximum. For the case without spray activation, a sigma cloud develop and spread around the dome and the steam generator compartment where the cold-leg break occurred. When the spray system is activated during hydrogen release, hydrogen mixing is augmented by the down-flow of the droplets and the flows induced by the steam condensation. These mixing flows reduce the local concentration of the hydrogen and finally the sigma-cloud in the dome region (Fig. 20.b). From the GASFLOW analyses of the proposed cases, it was found that hydrogen dynamics are heavily influenced by spray activation.

4.2 Flame Quenching Test

Combustion in subcompartments in severe accidents is uncertain and depends on geometry, hydrogen, and steam concentration, as well as on the availability of ignition sources. Depending on the geometry and gas distributions, DDT cannot be excluded in severe accidents. DDT can largely influence the maximum loads from

hydrogen combustion sequences and the consequential structural damage. Thus, the ultimate goal in hydrogen mitigation is to design countermeasures that allow operators to avoid flame acceleration and DDT. To apply the flame arrester concept for hydrogen control in severe accidents in NPPs, quenching meshes installed between compartments was suggested. The performance of the quenching mesh was performed on the flame speed and the pressure which are major parameters to determine the performance of the mesh. More detail principles of the quenching mesh can be found elsewhere [51].

4.2.1 Flame Quenching Facility

Fig. 21 shows the test facility, which consisted of the visualization system, the combustion chamber, the data acquisition system, and the electric spark igniter system.

Fig. 22 shows the combustion chamber. Three compartments $300 \times 300 \times 300$ mm³ in size were connected

Fig. 21. Schematic of the Experimental Apparatus

Fig. 22. Combustion Chamber

Fig. 23. Quenching Mesh

in parallel. Quartz windows of $180 \times 180 \text{ mm}^2$ were installed on both sides of the compartments for optical transparency. A static (SIEMENS, 7MF4032) pressure sensor was installed at Ch3-3 to control the initial hydrogen concentration and to measure the pressure during burning. Temperature sensors (K-type) were also installed at Ch 1-2 and 2-2 to measure the gas temperature and predict whether the image is a flame or not. An air-supply valve was also installed at Ch 3-1 to supply an atmosphere for setting-up the initial air-hydrogen mixture. Fig. 23 shows the quenching mesh with a 0.3 mm quenching distance determined from the quenching distance measurement.

Hydrogen concentration was set based on the partial pressure. If we want to set a 10% hydrogen concentration at 1.0 bar, hydrogen is added until the pressure at the combustion chamber reaches 1.111 bar. Initial hydrogen concentration was determined based on the partial pressure.

(a) without mesh

(b) with mesh

Fig. 24. Pressures in the Chamber without and with Mesh

(a) without mesh

(b) with mesh

Fig. 25. Temperatures in the Chamber without and with Mesh

Fig. 26. Flame Images for the Tests without and with the Mesh
[$P=1.1$ bar (left four), $P=1.4$ bar (right four), $H_2=10\%$]

Fig. 27. Flame Velocities at Different H_2 conc. and Pressures

Fig. 28. Pressures in the Chamber with the Mesh

4.2.2 The Results of Quenching Tests

The tests were carried out by increasing the initial pressure from 1.1 bar to 1.4 bar in increments of 0.1 bar [52]. The hydrogen concentration was changed from approximately 8% to 10% volume percentage. Tests were performed with and without meshes. The mesh was installed perpendicular to the direction of the flame propagation. The dimensions of the mesh were 280×280 mm².

Fig. 24 shows the pressures in the chamber with and without the mesh. Pressure rises in the combustion chamber for the case without the mesh increased with hydrogen concentration. Meanwhile, the pressures for the case with the mesh increased rapidly for the first one second and then stabilized. They were lower than the pressures of the tests without the mesh because burning occurred only in the first compartment.

Fig. 25 shows temperatures in the chamber with and without the mesh. The peak gas temperature in the combustion chamber for the case without the mesh was approximately 450 °C, which is almost the same at an equal hydrogen concentration. The temperatures at the second

compartment for the case with the mesh were much lower than that at the first compartment because the flame did not propagate beyond the mesh boundary.

Fig. 26 shows images of the flame taken at the window of the second compartment at 1.1 bar and 1.4 bar for the 10% hydrogen concentration without the mesh. The flame that started at the first compartment easily propagated to other compartments. However, the flame did not pass through the as-implemented mesh boundary.

Fig. 27 shows the flame velocities for the tests performed at various initial pressures and hydrogen concentrations. The flame velocity was estimated using the images of the flame taken by high-speed photography. The flame velocity increased with initial hydrogen concentration. The flame velocity was approximately 1.45 m/s for the 10% hydrogen concentration.

Pressure effects on the performance of the mesh were measured. Fig. 28 shows the pressure rise in the combustion when the initial air pressure increased. The pressures increased rapidly for the first one second and then stabilized

Fig. 29. Combustion Chamber to Induce High Flame Velocity

Fig. 30. Pressures and Temperatures in the Chamber

below 1.8 bar. In the tests, the limit of the pressure gauge was 5 bar. The pressure rise was constant from 2 s to 5 s for 2.0 and 2.2 bar, respectively. The mesh could not contribute to the flame quenching over 2.0 bar.

Flame velocity effect test of the mesh was performed. The combustion chamber, in which five compartments were connected in parallel, was installed to induce the high flame velocity (Fig. 29). An expansion vessel was provided at the end of the last compartment. The dimensions of the expansion vessel were 60 cm in length and diameter.

Quenching mesh effectiveness as a function of flame speed was tested at an atmospheric pressure at around 9% hydrogen mole percent with a dry air. Flames moving approximately 30 m/s were not quenched when one sheet was used to arrest the flame. However, the flame was quenched when two sheets was used. Fig. 30 shows the pressure and temperature when two sheets were used.

4.2.3 Analysis with Quenching Model

Quenching distance models were derived to apply them to square type mesh geometries and to evaluate various heat transfer modes from flame to gas and mesh wall [53]. Quenching distance depends on the heat transfer mode from flame to gas and mesh wall as borne out from quenching distance model calculations. Therefore, a simplified model of quenching distance was derived to apply the experimental data easily. Flame quenching criteria was proposed using the experimental data with the simplified quenching distance. If $\rho_w/(\rho_b \cdot P \cdot V_f)$ is greater than 10^{-5} at combustion conditions, the flame is quenched. Less than this, the flame propagates. This criterion will be useful in evaluating whether or not a flame can be quenched or propagated.

5. DEVELOPMENT OF INTEGRATED COMPUTER CODE, MIDAS

When the mid- and long-term R&D programs started, necessary was the development of in-house simulation codes to evaluate specific issues associated with domestic power plants and to analyze test data produced from the experimental program. Considering the output production schedule and application and the applicability to various configurations of plants, MELCOR 1.8.4 [54] was selected as a reference code.

Although that reference code has state-of-the-art models for severe accident analysis, its complex data transfer system using pointer variables made it difficult to implement new models into the current structure. Therefore, MIDAS was developed focusing on the following three areas: restructuring of the data transfer scheme, model updates, and development of user friendly input/output graphic user interfaces (GUI).

Fig. 31. Comparison of Data Structure before and after Restructuring

5.1 Restructuring of Data Transfer Scheme

5.1.1 Data Transfer in MELCOR

The MELCOR code is composed of 3 parts. MELGEN checks the input data and makes the code restart needed for the main calculation. MELCOR calculates with time from the restart file and makes a log/plot file. Finally PLOT-related programs control data plotting. Additionally, MELCOR is composed of a number of different packages each of which models a different portion of the accident phenomenology or program control [54]. MELCOR 1.8.4 is programmed in FORTRAN 77 language. Since FORTRAN77 does not provide dynamic storage allocation mechanisms, MELCOR uses pointer variables to handle data outside the packages. Though this technique was popular and efficient to manage the data when MELCOR was first developed, its complex data transfer scheme deteriorated readability, maintainability, and portability.

MELCOR reserves and uses four data types for data storage and efficient data transfer (Fig. 31): real, integer, logical, and character. It allocates fixed-size storage for each data type and manages the variable-sized data within it. Data in MELCOR are transferred in two steps. First, the starting location and size of each package for each data type are conveyed to the next level subroutine in the data transfer hierarchy as database common blocks. Pointer variables indicating the information used within each

package are then transferred through arguments, and they are used as local variables thereafter. To understand the function of these pointer variables, the comments and arguments within the related subroutines should be analyzed.

5.1.2 Code Restructuring

MIDAS adopts new features of FORTRAN 90 for core structure modernization. The data transfer structures were modularized from a single array into a derived data type for ease of use and understanding (Fig. 31). Additionally, the variables used in the restructured packages were transformed into a modularized type, and their efficiency was increased through a dynamic allocation using TYPE, USE, MODULE, and ALLOCATE statements [55]. This kind of restructuring made it possible to dynamically allocate storage and to use user-defined data types. This led to efficient memory usage and tractability.

Prior to performing code restructuring, the reference program language was first converted from FORTRAN 77 into FORTRAN 90. Automatic conversion programs were separately developed [56] and used to minimize manual restructuring, which often introduces human errors. Typical results of temperature and pressure were compared and found to be the same before and after the conversion. A restructuring template for simple packages (like SPR)

was then developed and tested and later expanded into the entire MELCOR code. During this process, hundreds of subroutines were analyzed per package, and dozens of subroutines which share data with other packages were analyzed.

5.1.3 Verification of MIDAS

In order to verify the MIDAS results, a three-step process was implemented. First, a simple language conversion process from FORTRAN 77 to FORTRAN 90 was checked by comparing major variables in all the

packages before and after the conversion. The next step was to verify the package-wise restructured results using the module and the derived type variables. An interface program was developed for the data communication between the restructured package and the original packages. Finally, MIDAS results integrated with all restructured packages were compared with those from MELCOR90, which was the simple language-converted MELCOR program. Figs. 32 to 36 show comparisons between MIDAS and MELCOR90 for a station blackout scenario in APR1400.

Fig. 32. Decay Heat Comparison before/after Restructuring

Fig. 34. Volumetric Flow Rates through SRV Comparison before/after Restructuring

Fig. 33. Core Upper Plenum Pressure Comparison before/after Restructuring

Fig. 35. Fuel Temperature (Cell 103) Comparison before/after Restructuring

Fig. 36. Total Mass of Deposited Fission Product Comparison before/after Restructuring

Fig. 37. Total Power Variation in Core

They predict the same behavior for decay power, system pressure, pressurizer flow rate, mass of fission product deposited on heat slabs, and fuel temperature [55].

5.2 Model Improvements

The capability of MIDAS has been extended from MELCOR 1.8.4 by adding new models and improving the current models. Representative improvements are described here.

The current MELCOR code treats fission power as user input. In order to simulate the power during an *anticipated transient without scram* (ATWS), a simple BWR-specific Chexal-Layman correlation based on system pressure and downcomer water level was included. As this correlation was not appropriate for PWRs, a point kinetics module was implemented into MIDAS to calculate the core power when the reactor fails to trip (ATWS). Thermal-hydraulic variables like the temperature, pressure, and water level were coupled with core variables to facilitate estimations of reactivity feedback effects. The ATWS initiated by a *total loss of feedwater* transient was analyzed and compared with MELCOR results. While MELCOR crashed at around 201 seconds due to a material property error caused by the primary peak pressure being above the designed pressure, MIDAS can simulate the accident up to 600 seconds when core water mass is depleted. Figs. 37 to 40 show comparisons between two codes [57]. This capability is expected to be useful to decide whether the events defined in the event tree will succeed or not for the initiating event of ATWS.

The coolability of hot debris in the lower head filled with water during severe accidents is an important issue concerning plant safety. KAERI has launched the

Fig. 38. Water Mass in Core

SONATA-IV experimental program to resolve the in-vessel issue [62]. The *Lower-plenum Arrested Vessel Attack* (LAVA) series of experiments has been performed to study the existence of initial gaps and their effects on the cooling of hot debris. A model for coolant penetration into the gap based on counter-current flow limitation has been developed and implemented into MIDAS. The rapid cooling temperature profile on the outer surface and the average heat transfer rate through the gap were confirmed by LAVA [58] and the LMP200-1 experiments [59] (Figs. 41 and 42, respectively). Though the model

Fig. 39. Pressure in Pressurizer

Fig. 41. Comparison of Outer Wall Surface Temperatures at LAVA-4 Experiment

Fig. 40. Fuel Temperature in Fuel Cell 212

Fig. 42. Comparison of Outer Wall Surface Temperatures at LMP200-1 Experiment

captures the fundamental phenomena for rapid cooling, more sensitivity studies and model improvements are needed.

The debris dispersal process during high pressure melt ejection depends on cavity geometry. An empirical Kim correlation was formulated from a number of low temperature simulant experiments employing mock-up of reactor cavities as a function of explicit geometric parameters [60]. This correlation was coupled to MIDAS to determine the amount of debris that interacts with containment atmosphere directly during direct containment

heating. The cavity geometry and initial conditions were used as parameters of the correlation.

A steam explosion module, STX, has been developed to estimate the dynamic load in the reactor cavity. It has been implemented in MIDAS [61]. The pre-calculated results using the mechanistic steam explosion analysis code, TEXAS-V, were generated for a range of coolant and fuel conditions thought possible. The TEXAS-V results were summarized in the format of a look-up table. Linear interpolation was used to calculate mid-point values in the data table. Through the stand-alone and

integrated tests, it was confirmed that the STX module with MIDAS worked as expected in the data range investigated.

5.3 Input/Output Graphical User Interface (GUI)

The MIDAS GUI system has been developed to enhance MIDAS I/O capabilities under the Microsoft Windows operating system.

5.3.1 IEDIT Input Management System

The MIDAS input management system, IEDIT, was developed to systematically manage MIDAS input files. It provides menu-based GUI environments and graphic tools for an input design. Currently, IEDIT controls inputs for five packages: CVH, HS, FL, COR, and CF.

Input information of the selected package appears on a display panel. Since all information items are displayed

Fig. 43. CVH Information Display Panel from IEDIT

Fig. 44. Display of Plot Variables on a Graph Window from IPLOT

at once, it is possible to check the whole control volume or heat structure status with ease. Moreover, reviewing the input variables becomes much easier by printing the display panels. Fig. 43 shows the display panel designed for the CVH inputs. By clicking the package selection button on the top of the panel, one can change the display panel to the selected package.

Automatic input generation capability is one of the main features of IEDIT. With this function, IEDIT can generate new input lines as soon as the menu items of the display panel are changed. Since input lines are generated automatically, the most frequent input errors such as typing errors and complicated control function logic errors can be prevented.

5.3.2 IPLOT Variable Plotting System

IPLOT was developed to plot graphs. During execution, MIDAS records time history of meaningful variables as "plot variables". While MELCOR handles the plot file using HISPLT with a text-formatted input, IPLOT is a menu-driven program to read and write the binary data. This binary access mode enables IPLOT to search and to draw the variable parameters very quickly. In order to read the binary plot file, IPLOT employs file opening, variable nodes generation, volume number lists generation, and time data reading in order. IPLOT will be extended to treat the MAAP data as well as the user-defined graph data source.

Fig. 44 shows the plot file graph window of IPLOT. The plot file location and title are placed at the top of the window, and all the plot variables are listed in the left search window in the form of a tree view style. The identities of selected plot variables are listed in the middle-top window, and the selected plot variables are listed in the next window. The selected variables can be added or deleted by using the command buttons on the right. It is possible to change the range of the graphs by entering the x- and y-axis boundary values. Moreover several functions such as zooming, grid line setting, title editing, legend setting, and printing are available. The graph appearing in Fig. 44 can be saved for later usage. IPLOT can duplicate the graph window with another set of plot variables for ease of comparing variables.

6. CONCLUSIONS

The role of nuclear power in Korea is becoming more important in terms of enhancing its national energy. This follows from the high price of fossil fuels and international climate agreements drawn up to mitigate greenhouse gas emissions. The Korean nuclear industry thus continues to operate and develop nuclear power plants to meet energy demands. The construction of Shin-Kori NPP unit 3&4, which are the first 1,400 MWe nuclear power plants in Korea, has already begun. In developing nuclear power

plants, public safety from severe reactor accidents is the first issue to consider.

To ensure public safety against severe accidents, maintaining reactor vessel integrity and containment is key. Under the long-term nuclear R&D program, the severe accident research program at KAERI is directed to investigate, through the analysis and experimentation, unresolved severe accident issues, such as core debris coolability, steam explosions, and hydrogen combustion.

The Korean Advanced Reactor APR1400 has adopted an IVR-ERVC strategy for severe accident management aiming to preserve reactor vessel integrity. The margin of safety, however, may decrease, when for example, metal layers are present on top of the oxidic corium pool. For the higher power APR1400 reactor, the safety margin is even more reduced due to higher thermal loads on the reactor vessel wall from the molten pool. The molten pool configuration and composition of melt also affect heat flux distribution on the reactor vessel wall. Recent experimental results of the OECD sponsored MASCA projects highlighted more complicated situations of the melt pool composition and configuration such as the three layer molten pool [5].

For APR1400, a strategy of simultaneous flooding of the metal layer with the external reactor vessel cooling could be a solution to remove sufficient heat from the upper surface of the metal layer and to reduce the focused heat flux to value less than the critical heat flux, which needs to be verified. To evaluate the IVR-ERVC concept for APR1400 as a severe accident management strategy, extensive studies have been performed at KAERI, and an improvement of the insulator design outside the vessel was investigated to provide the effective flow path for ERVC by performing scaled experiments of the natural circulation loop. However, there are still some gaps in understanding molten pool behavior. Considerable uncertainties must be resolved for the IVR-ERVC SAM strategy.

A reliable quantification of the risk of steam explosions in reactors is the main focus of steam explosion research, be it experimental or analytical. Conclusions from the recent international research on steam explosions (OECD/SERENA Phase 1) was that they had to tune the parameters of the computer codes, which was tuned to the KROTOS alumina tests, to properly simulate the reduced energetics of the experimental results of FARO and TROI using corium [10]. It was concluded that the major uncertainties on explosion are related to limited knowledge on pre-mixing dynamics as already revealed by the scatter of results obtained in pre-mixing experiment simulations and of melt behavior during propagation of the explosion (material effect).

These issues could be resolved by recently launched OECD/SERENA phase 2 project. The project is formulated to resolve uncertainties on void and material effects by performing a limited number of well-designed tests with

advanced instrumentation reflecting a large spectrum of ex-vessel conditions. Complementary analytical work to bring the code capabilities to a sufficient level for use in reactor case analyses are also a goal. These goals are to be achieved by using the complementary features of KROTOS (CEA) and TROI (KAERI) corium facilities. KROTOS is more suited for investigating intrinsic FCI characteristics in one-dimensional geometries. TROI is more suited for testing the FCI behavior of these materials in reactor-like conditions by having more mass and multi-dimensional melt water interaction geometries.

Hydrogen combustion has been known to be one of the major factors that threaten containment because the large amount of hydrogen generated under severe accidents can cause explosive reactions. Considerable effort has been directed to evaluate the hydrogen concentration distribution and to control the destructive hydrogen burn.

To investigate the hydrogen concentration distribution, GASFLOW analyses have been conducted for the SBLOCA accident in the APR1400 containment; the effects of the PAR, igniter, and spray system were evaluated. PAR is known as a very effective system to remove hydrogen. However, from this study, it was found that the hydrogen concentration reaches more than 10 vol% for a short period of time in some compartments such as steam generator room and dome region when the hydrogen release rate is too large. It is thought that PAR must be considered as a long-term hydrogen removal system. When the igniter is activated, a standing flame is produced around the upper part of the reactor coolant pump. The standing flame burns most of the hydrogen blown-out. The hydrogen concentration in the containment then drops to below 2 vol%. Igniters are very effective for removing fast released hydrogen in the steam generator room. In view of the thermal load on the containment and equipment survivability from the standing flame, however, it is necessary to control the igniter operation. From the analyses of the SBLOCA where the spray is activated, it was found that the size of the sigma clouds could be reduced by spray activation. It is thought that the down-flowing of the spray droplets via gravity helps the mixing of the hydrogen in the containment, but it must be noted that the condensation of the steam distributed in the containment by the spray droplets can cause an increase in the local concentration of hydrogen.

Performance tests of a quenching mesh with a 0.3 mm gap distance were performed at elevated pressures to check their effectiveness in nuclear power plants. Experiments show that when the quenching mesh is installed between the compartments, the flame cannot pass through the mesh at about 10% hydrogen concentration with the initial pressure of below 1.8 bars. Based on the test results, the flame quenching criteria was derived.

Along with the phenomenological investigation to resolve severe accident issues, the integrated severe accident analysis code, MIDAS, has been developed

based on MELCOR 1.8.4. It has a modernized data structure using FORTRAN 90, new models, and a user friendly input/output graphical user interfaces. MIDAS is expected to be used by current users and new-comers alike, as they can prepare the input and analyze the output easily. Additionally, the restructured code should facilitate model improvements and enhance user friendliness (accessibility and readability).

The suggested new resolutions to the severe accident issues of corium retention in-vessel, steam explosion, and hydrogen mitigation can be implemented to the integrated code in a way, so as to extend their application to the actual nuclear power plants. In addition to the continuous research efforts for the model upgrade and verification, MIDAS can be a role for these issues as a reservoir with the domestic analytic and experimental data.

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