1.4 Overview of Severe Accident Research at KAERI

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ABSTRACT

The severe accident research program at Korea Atomic Energy Research Institute, within the framework of governmental 10 year long-term nuclear R&D program, aims at the development of assessment techniques and accident management strategies for the prevention and mitigation of potential risk. The research program includes experimental efforts, development of phenomena specific models and development of an integrated computer code. The results of research program is intended to be utilized for the design of the advanced light water reactor and development of accident management strategies for the operating reactors. The main focused areas of recent investigation at KAERI are experiments on in-vessel core debris retention (SONATA-IV) and fuel coolant interaction (TROI) along with the development of models and integrated computer code (MIDAS).

Keywords: severe accident, in-vessel debris retention, fuel coolant interaction, integrated computer code, and research program

1.Introduction

To enhance technological capability and self-sufficiency in nuclear energy supply, the Korean government has established a long-term nuclear R&D program. This program is aimed at developing a wide range of technologies needed for peaceful uses of nuclear energy and also building a foundation for national energy self-sufficiency. In the frame of this long-term program authorized by Ministry of Science and Technology (MOST), severe accident research program has been carried out.

In the phase-II stage (1997-2001), severe accidents research program is directed to 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(Fig. 1). Two experimental programs have been launched and are in progress. Those are SONATA-IV (Simulation Of Naturally Arrested Thermal Attack - In Vessel) program with the aim of investigating the possibility of in-vessel debris cooling and reactor vessel integrity, and CONVEX (CONtainment integrity Verification EXperiment) program, a corium-water interaction and steam explosion experiment with the aim of quantifying the impact of such events on containment integrity. Along with the experimental program, development of integral severe accident analysis computer code is being carried out by combining the assessment of currently available integral analysis code and developing new models for unresolved issues. The code is also aimed to be used to assess severe accident sequences for the future reactor, SMART (System integrated Modular Advanced ReacTor), being developed at KAERI.

2. SONATA-IV Program

SONATA-IV experimental program, which is to investigate the possibility of gap formation between the debris and the reactor vessel wall and in-vessel debris cooling through the narrow gap, has been launched. A series of experiments called LAVA (Lower-plenum Arrested Vessel Attack) and CHF Experiments in a Hemispherical Gap, are in progress as the first phase of SONATA-IV.

2.1 LAVA Experiments

In order to investigate the possibility of in-vessel debris cooling through a narrow gap, a series of experiments called LAVA (Lower-plenum Arrested Vessel Attack) are in progress as SONATA-IV phase-I study(Table 1). A $1/8^{th}$ linear scaled mockup of a lower head vessel was used with Al_2O_3/Fe thermite melt (or Al_2O_3 only) as a corium simulant. The tests were performed varying the initial conditions such as water subcooling and height, also the material compositions of the melt simulants. In these tests, the lower head vessel experienced deformation and a thin gap formed around the interface between the solidified debris and the vessel due mainly to the thermal load from the termite melt. The results show that there was a significantly rapid temperature reduction of the vessel in the Al_2O_3 thermite melt experiments compared with the Al_2O_3/Fe thermite melt experiments even though a gap was formed in both cases. It is postulated that in the Al_2O_3/Fe thermite melt experiments, the iron melt layer is so dense that water ingression into the gap is difficult due to the high pressure of escaping steam. On the other hand, in the porosity of an Al_2O_3 melt layer could enhance water ingression into the gap by giving the flow path of the evaporated steam through the porous media. The water height and subcooling could affect the melt pool formation and the initial thermal attack to the vessel. For clear confirmation of these effects, the tests will be performed with various initial conditions, especially at the saturated and the lower subcooled (approximately 100K) water conditions. In parallel with experimental work, development of computer code, LILAC (Lower Head IntegraL Analysis Code), to simulate overall phenomena for the in-vessel corium retention is on progress.

2.2 CHF Experiments in a Hemispherical Gap

A series of experimental investigation on the cooling mechanism in hemispherical narrow gaps has been carried out. A visualization experiment, VISU-II, was done as the first step to get a visual observation of the flow behavior inside a hemispherical gap and to understand the mechanism inducing global dryout. It was observed that the counter-current flow limitation (CCFL) phenomenon prevented water from wetting the heater surface and induced dryout.

CHFG (Critical Heat Flux in Gap) using R-113 has been performed to measure the critical power and to investigate the inherent cooling mechanism in hemispherical narrow gaps. The test results have shown that increases in the gap thickness and the system pressure lead to increase in critical power. Temperature measurements over the heater surface show that the two-phase flow behavior inside the gaps is quite different from the other usual CHF experiments with small-scale horizontal plates and vertical annulus. The measured critical power using R-113 in hemispherical narrow gaps are 60% lower than that using water due to the lower boiling point, which is different from the pool boiling condition.

3. Fuel Coolant Interaction Experiments

Steam explosions have been of concern in the nuclear industry since in hypothetical severe accidents which involve fuel melt down, the extremely fast thermal interactions, and possibly some chemical reactions, between the molten fuel and coolant may cause steam explosion and subsequent containment failure.

As part of the CONVEX program, KAERI launched a intermediate scale fuel coolant interaction (FCI) experiment called as TROI(Test for Real cOrium Interaction with water) using reactor material to investigate the physical mechanism of FCI in the area of explosion and mixing steam explosion experiments. Up to 20kg of the reactor material including UO_2, ZrO_2, Zr and SS will be poured into water pool inside a pressure vessel. For the melting of reactor material, the cold crucible technology that uses high frequency induction heating in a water-cooled cage will be used. As it offers greater purity and flexibility when working with reactive materials, it is beneficial in terms of composition control compared to the previous technologies. The test facility and relevant instrumentation will be designed and constructed by March 2000. In corporation with university of Wisconsin, three-dimensional FCI computer codes, TEXAS-3D, will be also developed by employing current computational fluid dynamics technology.

4.Code development and Assessments

Along with the experimental program, development of integral severe accident analysis computer code MIDAS (Multi-purposed IntegrateD Assessment code for Severe accidents) is being carried out both by assessing currently available integral analysis codes and by developing new models for unresolved issues. The MELCOR computer code has been selected as a reference code thanks to its flexibility in plant configuration simulation, consideration of momentum conservation and its many updated models for severe accident phenomena. MIDAS will be developed into a two-step approach. MIDAS/TH, which simulates the thermal hydraulic phenomena, is an output in the first step. Then, the behavior of the fission products is modeled and integrated into MIDAS.

The models based upon plant configuration parameters will replace a user specified debris dispersal in the containment during the high-pressure melt ejection sequences. Regarding the molten core-concrete interaction models, a user flexibility is provided for the heat transfer between the molten debris pool and the overlying water. Melt spreadibility in the cavity concrete is also being studied. One of the most important in-vessel retention issues is to identify gap cooling phenomena in the lower head. For this purpose, the SONATA-IV experiments have been simulated by MELCOR and the conceptual gap cooling model is being developed. As the current version of MELCOR has a limitation to describe reactor power history during an ATWS (Anticipated Transient without Scram) sequence, a point kinetics module has been implemented into MIDAS/PK. In addition, MIDAS will be modernized by redefining data structures for the enhanced readability, portability, and flexibility. Eventually, MIDAS is to be used to assess severe accident sequences for the KNGR (Korea Next Generation Reactor) and other future reactor like SMART (System integrated Modular Advanced ReacTor), being developed at KAERI.

5. Summary

Severe Accident Research Program at KAERI is toward to develop accident and risk management strategies through maintaining the integrity of the reactor vessel and the containment. For In-vessel debris retention issue resolution and coolability for accident management and improvement of design, SONATA-IV program has well under progression. Series of experiments LAVA, CHFG, CCFL have been performed and demonstrate the gap formation and cooling mechanism. TROI program is focused to investigate the physical mechanism of FCI phenomena using reactor material in the area of triggering and mixing. Melting method using cold crucible has been adopted and possibility of UO_2/ZrO_2 mixture melting is confirmed. For quantification of accident management strategy, MIDAS computer code is under developing.

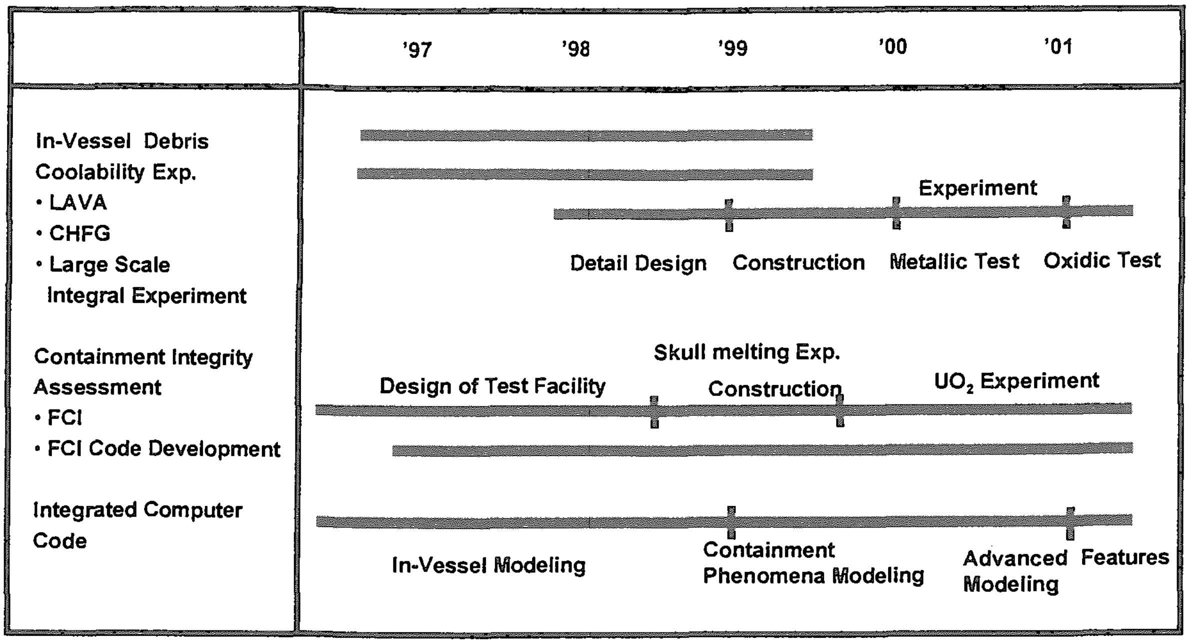

Figure 1. Phase-II of the Severe Accident Research Program

Table 1. LAVA Test Matrix

Test	Date	Melt Composition & Mass	Water Temperature & Initial Depth	Pressure Load across the LHV
LAVA_PRE	'97. 5. 23	Al_2O_3/Fe, 20 kg	423 K, 45 cm	0.0 bar
LAVA-1	'97. 7. 24	Al_2O_3/Fe, 40 kg	418 K, 50 cm	0.0 bar
LAVA-2	'97. 10.21	Al_2O_3/Fe, 40 kg	433 K, 50 cm	16.5 bar
LAVA-3	'97. 12. 9	Al_2O_3, 30 kg*	433 K, 50 cm	15.7 bar
LAVA-4	'98. 2. 17	Al_2O_3, 30 kg	427 K, 50 cm	16.9 bar
LAVA-5	'98. 9. 28	Al_2O_3, 30 kg	458 K, 50 cm	16.9 bar
LAVA-6	'98. 7. 4	Al_2O_3/Fe, 40 kg	427 K, 50 cm	16.6 bar
LAVA-7	'98. 11.18	Al_2O_3, 30 kg	448 K, 50 cm	17.4 bar
LAVA-8	'99. 2. 8	Al_2O_3, 30 kg	420 K, 25 cm	15.4 bar

*: In LAVA-3, 4, 7 tests, due to the incomplete melt separation, about 3.6 kg, 1.8 kg and 2.9 kg of Fe melt was poured into the Lower Head Vessel, respectively.

Acknowledgement

This project has been carried out under the Nuclear R&D Program by MOST

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