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Severe Accident Research Activities in Korea: In-Vessel Retention of Molten Core Material

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ABSTRACT

A cooling mechanism due to boiling in a narrow, hemispherical, irregular gap between the debris crust and the reactor vessel wall was proposed for interpreting the cooling of the vessel lower head during the TMI-2 accident. The proposed in-vessel cooling mechanism due to material creep and water ingression into the expanding gap between was found to explain the non-failure of the TMI-2 reactor vessel. With the success of the first-principle modeling (MIRA) and proof-of-principle molten material (LAVA) and thermal hydraulic (CHFG, VISU) tests, we propose to carry out a large-scale, high-pressure, prototype-material experimental program SONATA-IV. The program is factored into basic thermal hydraulic, high-temperature high-pressure melt, and material property tests in various phases of differing degrees. In addition to the natural mechanism, the program will also pioneer newly engineered concepts of in- and ex-vessel gap cooling structures for advanced reactor designs. This newly proposed cooling mechanism and structures, when experimentally and analytically validated, could be considered, counted on, and utilized as part of critical in-vessel severe accident management strategies for the current and next-generation reactors. In addition, external cooling of the reactor vessel lower head by flooding the reactor cavity is being studied to asses the feasibility as a potential severe accident management strategy.

INTRODUCTION

Because of their far-reaching consequences to the human beings and to the environment at large, severe accidents are becoming part of integrated safety analysis for reactors in their design stage ever since the Three Mile Island Unit 2 (TMI-2) accident [1]. The first priority should of course be placed on preventing the core melt accident from happening. Should the accident progress into a severe condition, however, then the next priority shall be to mitigate further development of the accident by utilizing all means of accident management available at the site.

There is thus a compelling need to develop a comprehensive accident management program that orchestrates the proven state-of-the-art technologies and evolving innovative research and engineering results in the area. The ultimate goal of the program should be to resolve the plant-specific accident management issues utilizing a coherent, consistent, pragmatic, methodical approach. In this paper we shall focus on the study of retention and coolability of molten core material within the reactor pressure vessel (RPV) lower plenum during a severe accident.

The in-vessel cooling mechanism due to material creep and water ingression into the expanding gap between the debris and the RPV wall was found to explain the nonfailure of the TMI-2 RPV lower head [2-6]. With the success of the modeling effort, a research program SONATA-IV (Simulation of Naturally Arrested Thermal Attack in Vessel) had been developed to thoroughly investigate this inherent nature of degraded core coolability inside the lower head by Suh et al. [7-9] as figuratively illustrated in Fig. 1(a). Its engineering application was recently reported by Hwang et al. [10] for the structures shown in Fig. 1(b). The feasibility study is also in progress regarding the in-vessel retention through external vessel cooling by flooding the reactor cavity.

PROBLEM STATEMENT

Internal Cooling of Reactor Vessel

A cooling mechanism due to boiling in a gap between the debris crust and the RPV wall may be theorized in terms of the following specific conditions:

- ☐ initial contact resistance gap formation between the debris and the RPV wall upon relocation to the lower plenum (see Figs. 2 and 3 for the MIRA modeling),
- ☐ relative growth of the gap between the debris and the RPV wall under mechanical and thermal loads, and
- ☐ enhanced heat removal due to coalesced bubble churn turbulent boiling in the narrow gap (see Fig. 4 from the VISU tests).

The focus of the mechanism presented here is that minimal localized damage of the RPV wall, or strain, is expected to result in cooling of the RPV lower head. The results of the proposed inherent cooling mechanism [5,6] were found to be consistent with the observations from the TMI-2 Vessel Investigation Project [1].

External Cooling of Reactor Vessel

The assessment of hydrodynamics illustrates that the energy conducted through the RPV wall can be removed by nucleate boiling process on the outer surface. For instance, it was found that heat fluxes approaching 1 MW/m^2 could be removed by boiling on the outer surface even with the RPV insulated. No indications of high wall surface temperatures were observed, and the entire transient was characterized by nucleate boiling. In a reactor system, the expected heat fluxes would be less because of a greater water leakage through the insulation and an unlimited venting of the steam through the RPV-to-insulation annulus. Consequently, the lower head would be effectively cooled to remove the energy transferred to the wall from the debris crust. This might be a very influential recovery action to both mitigate and terminate the progression of the accident by keeping the core material in the RPV lower plenum. Of particular importance is that the external cooling can retain the strength of the RPV lower head by keeping the temperature of a significant portion below 800 K.

One material concern that may arise with respect to external cooling of the RPV lower head might be the quench crack. The cracks are caused by the cool cavity water beginning to submerge the overheated RPV lower head which could be overheated with relocating mass of the molten debris material from the core. Relatively large pieces of carbon steel that are rapidly quenched may crack as a result of internal stresses. This may become a problem especially when the carbon content exceeds 0.5 wt %. For higher carbon contents, a water quench may be too severe because cracking and warping may be produced. This quench crack issue should clearly be resolved before the external cooling of the RPV could be adopted with certainty as a viable severe accident management strategy for existing and future reactors.

RESEARCH ACTIVITIES

Internal Cooling of Reactor Vessel

In order to systematically investigate the potential for in-vessel debris cooling, a series of LAVA (Fig. 5) proof-of-principle tests are in progress using the Al_2O_3/Fe thermite simulant at the Korea Atomic Energy Research Institute (KAERI). In tandem, the first-principle one- and two-dimensional MIRA analyses are being performed at the Seoul National University (SNU) [9] to interpret the LAVA test data. In these tests the influence of internal pressure load on the lower head vessel wall and the materials of the simulant melt on gap formation (Figs. 6 and 7) was investigated as well as the thermal behavior of the vessel (Fig. 8). No indication of vessel failure was observed in any of the tests performed so far. In case the internal pressure was imposed, the lower head vessel experienced deformation at elevated temperatures and a thin gap formed around the interface between the solidified debris and the carbon steel vessel. The rapid cooling of the high temperature melt manifests the existence of a cooling mechanism of water ingression through the debris-to-vessel gap and the intra-debris pores and crevices.

External Cooling of Reactor Vessel

The engineering feasibility study is in progress at the Korea Advanced Institute of Science and Technology (KAIST) [11,12], the Korea Electric Power Research Institute (KEPRI) [13] and the SNU [14,15] regarding mainly the in-vessel retention through external vessel cooling (IVR-EVC). Pool boiling critical heat flux (CHF) experiments were performed at KAIST using nearly saturated water at atmospheric pressure (Fig. 9(a)) with particular attention to varying gap sizes and inclination angles (Fig. 9(b)). Also, detailed investigations were carried out to identify the most suitable cooling methodology for cooling the molten corium in the KNGR geometry [16]. A regulatory research is in progress for the IVR-EVC concept at the Korea Institute of Nuclear Safety (KINS) along with KAIST to develop technical bases for regulation. KEPRI estimated both deterministically and probabilistically the coolability of corium. KEPRI plans to assess the scaling effect of the test vessels for the boiling mechanism and the CHF measurement, and to design the RPV flooding system for IVR. The operation of the flooding system during a severe accident will be considered in the KNGR accident management guidance (AMG).

CONCLUSION

To secure the integrity of the RPV during a severe accident considering both the mechanical and thermohydraulic aspects, it is necessary to increase the safety margin applying both the internal (Fig. 10) and external (Fig. 11) cooling strategies discussed in this paper. According to the mechanical behavior results, the RPV integrity may be jeopardized if the surface heat flux exceeds 0.4 MW/m^2. If we maximize the internal cooling capability, the amount of the heat transferred to RPV will decrease. In addition to the internal cooling, if the external cooling is realized efficiently, the integrity of the RPV may be considerably enhanced from both the mechanical and thermal points of view. The IVR-EVC may be effectuated by either flooding the reactor cavity or utilizing an advanced COASISO (Corium Attack Syndrome Immunization Structure Outer-wall) as illustrated in Fig. 12.

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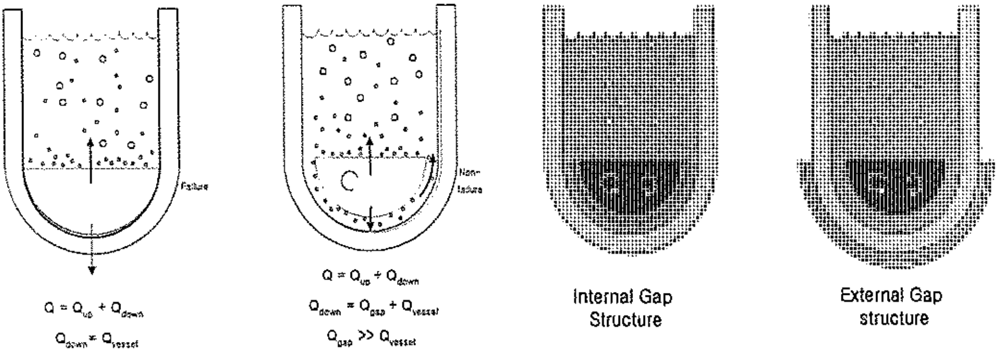


(a) natural gap (b) engineered gap

Figure 1. Gap Cooling Mechanisms

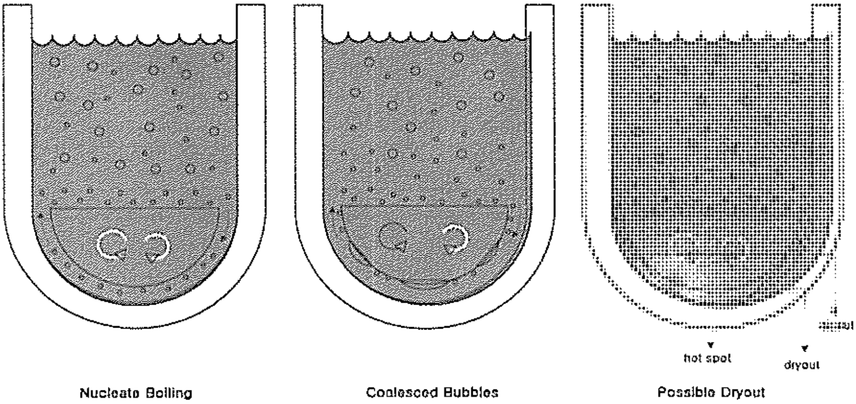


Figure 2. Inherent Negative Feedback Heat Removal Mechanism

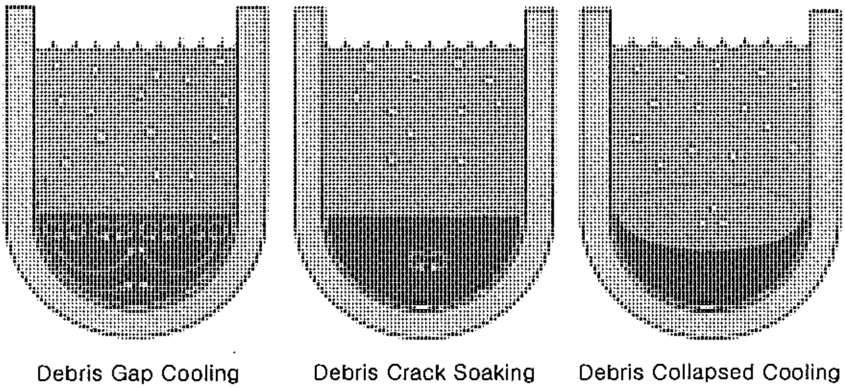
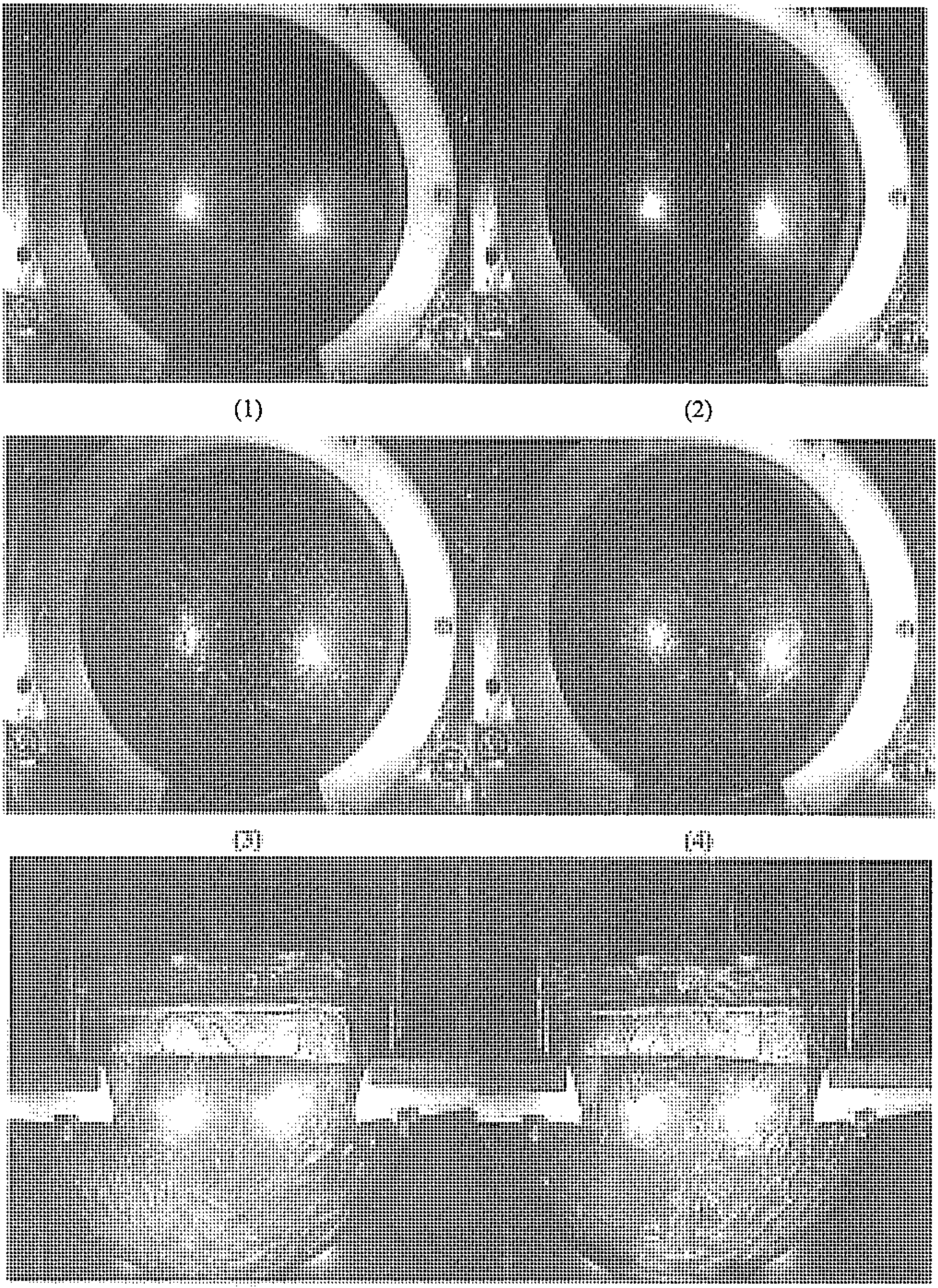


Figure 3. Enhanced Cooling of Debris inside the Vessel

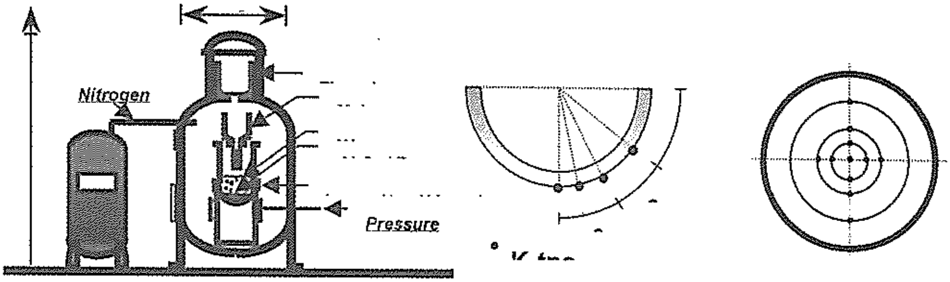


(1) (2)

(3) (4)

(5) (6)

Figure 4. Two-phase Boiling Behavior of the LHV Gap

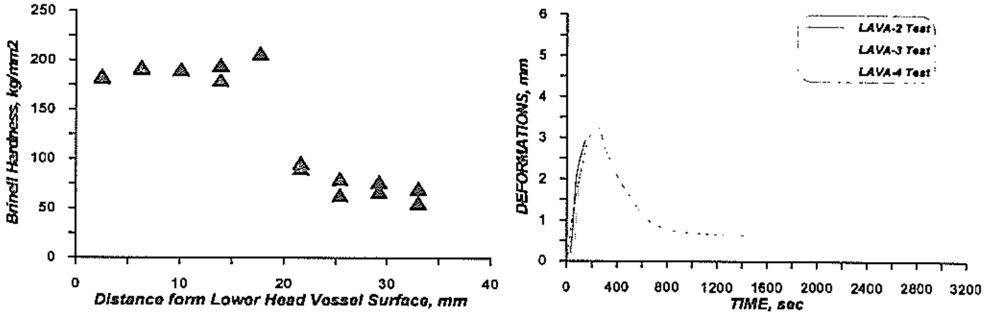


(a) schematic diagram of the LAVA test facility (b) thermocouple locations on LHV surface

Figure 5. The LAVA Facility and LHV Instrumentation

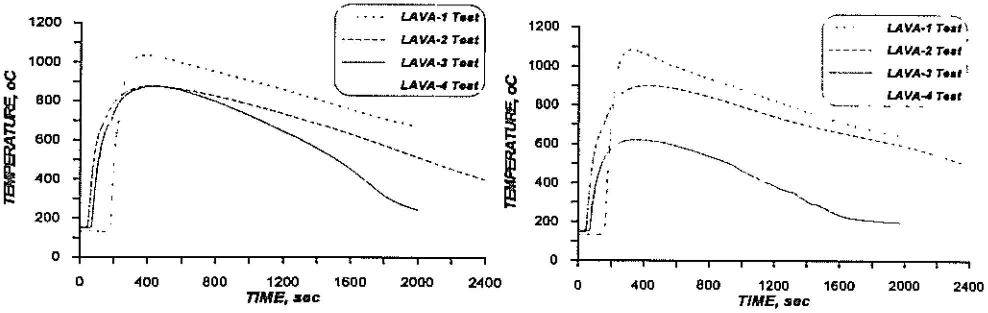
(a) LAVA-1 test (b) LAVA-2 test (c) LAVA-3 test

Figure 6. Cut-away View of the LHV and Debris



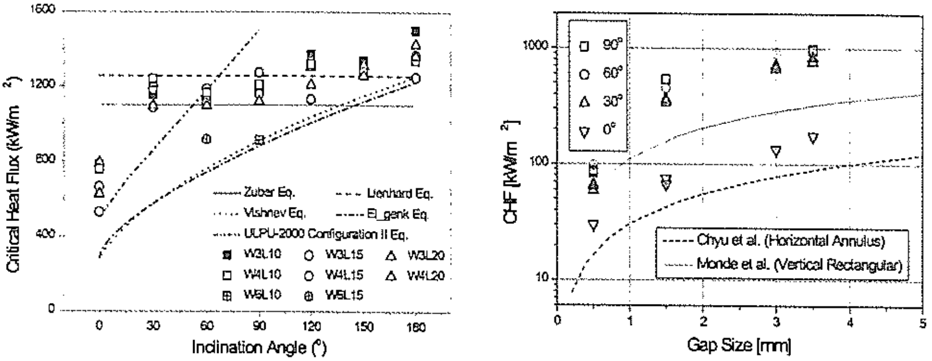
(a) mechanical property distribution (b) deformation

Figure 7. Mechanical Behavior of the LHV



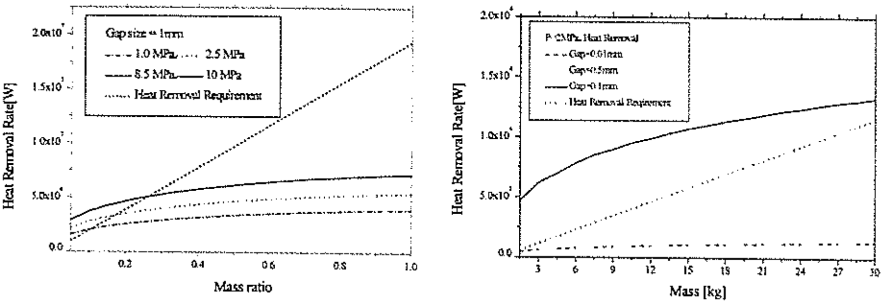
(a) 15° position (b) 30° position

Figure 8. Thermal Behavior of the LHV



(a) inclined flat plates (b) gap size and inclination angle

Figure 9. CHF Behavior of LHV Inclination and Gap Size



(a) by pressure for KNGR (b) by gap size for LAVA-4

Figure 10. Maximum Heat Removal Capability

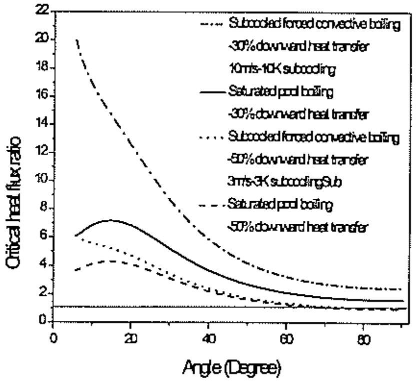


Figure 11. Calculated Thermal Margin on the LHV Outer Wall

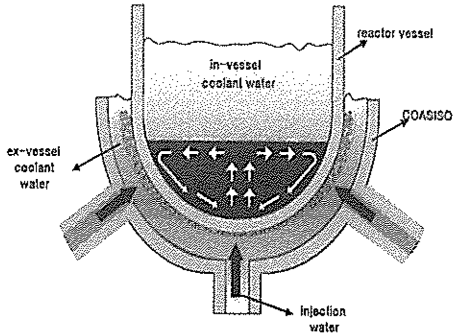


Figure 12. Conceptual Design of Advanced COASISO