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Unresolved Issues in Severe Accidents for Advanced Light Water Reactors

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ABSTRACT

Some of the outstanding issues of the severe accident are evaluated from the synthesis of the results from experimental and analytical researches including Simulation Of Naturally Arrested Thermal Attack in the vessel (SONATA), Test for Real cOrium Interaction with water (TROI), test for the performance of quenching screen. The strategy of In Vessel Retention (IVR) by Ex- Vessel Reactor Cooling (EVRC) is evaluated for the high power reactors. The favorable effect and adverse effect of the proposed strategy and the uncertainties of relevant phenomena including the gap cooling and heat removal capability by EVRC are discussed using the results of RELAP5/MOD3 analysis and SONATA. It is shown that the margin to the failure of the reactor vessel is challenged for those high power reactors. The observation of the energetic spontaneous steam explosions in TROI suggested that proper measures should be taken to avoid and/or lessen the risk of in-vessel or ex-vessel steam explosion, when we employ an ex-vessel cooling strategy and/or in vessel injection strategy. To solve the dilemma that the absence of igniters can result in potentially detonable mixtures, while multiple burns induced by the use of an igniter can pose a threat to the safety equipment, the use of a quenching screen is suggested. The results of basic experiments and laboratory scale experiments showed the feasibility of using a quenching screen to protect the essential equipment by stopping flame propagation.

KEYWORDS: Severe Accident, In Vessel Retention, Steam Explosion, Hydrogen Burn, Equipment Survivability

INTRODUCTION

The risk of severe accidents should be substantially minimized for the Advanced Light Water Reactors (ALWRs) by engineered safety features and/or a Severe Accident Management (SAM) strategy. However, there still exists some unresolved issues of severe accidents, whose phenomenology and their consequences are still uncertain. A risk informed regulation [1] is proposed, which indicates a paradigm shift in determining the risk of potential accidents in a nuclear reactor from a deterministic approach to a probabilistic approach. The highlighted phenomena of severe accidents, which are identified to have a high importance but uncertain risk [2], draw attention. Some of the outstanding issues in the severe accidents are evaluated from the synthesis of the results from the experimental research of Simulation Of Naturally Arrested Thermal Attack in the vessel (SONATA) [3], Test for Real cOrium Interaction with water (TROI) [4,5], test for the performance of quenching screen [6,7], and the results of a computer code analysis [8].

The In Vessel Retention (IVR) strategy by Ex- Vessel Reactor Cooling (EVRC) is proposed for the high power reactors, such as KSNPP (Korea Standard Nuclear Power Plant) and APR1400. The favorable effect and adverse effect of the proposed strategy and the uncertainties of the relevant phenomena including the gap cooling and heat removal capability of EVRC are discussed by using the results of the RELAP5/MOD3 analysis [8] and experiments on gap cooling [2].

The observation of the energetic spontaneous steam explosions in the fuel coolant interaction experiments using ZrO_2 [4] and corium [5] in a TROI facility suggests that proper measures to avoid and/or lessen the risk of in-vessel and/or ex-vessel steam explosion, should be taken when we employ an ex-vessel cooling strategy and/or an in vessel injection strategy.

The hydrogen burn in severe accidents was not considered for the qualification of equipment and instruments in the reactor containment, which could be utilized for the mitigation of severe accidents. So, industrial and regulatory bodies investigated the issue [9,10,11] and found that the existing equipment and instruments can perform their functions properly under single hydrogen burn. However, NUREG/4763 [11] indicated a dilemma that the absence of igniters can results in potentially detonable mixtures but multiple burns induced by the use of an igniter can pose a threat to the safety equipment. So a quenching screen, which would protect the essential equipment during hydrogen burn by stopping the propagation of the flame is suggested. The results of basic experiments [6] and a laboratory scale experiment for the proof of the principle test using a quenching screen [7] are discussed to evaluate the feasibility.

IN VESSEL RETENTION FOR HIGH POWER REACTOR

In-Vessel Retention (IVR) of molten core [12] is proposed for the Advanced Light Water Reactors (ALWRs), such as AP600. The IVR strategy is attractive, as it could substantially reduce radiological releases by confining the molten core inside the reactor vessel. The reactor cavity is flooded with cold water and reactor vessel cooling in a cavity is then performed either by a passive natural circulation or by forced cooling to implement IVR. For the success of the proposed strategy, the decay heat generated from the molten core pool should be effectively removed by the external cooling of the reactor vessel. Theofanous et. al.[12] did an extensive analysis on this subject for AP600 and concluded that lower head failure is ‘physically unreasonable. It was feasible for small power reactors, such as, AP600 and Loviisa. So, the concept is investigated for the reactors of higher core powers, such as Korea Standard Nuclear Power Plant (KSNPP), APR1400 [13,14], SWR1000 [15], and AP1000 [16].

The conceptual configuration for the ex-vessel cooling for KNGR [8] is represented in Fig. 1, where the natural circulation flow path is made along the insulation enclosure. The insulation enclosure is shown to be beneficial in terms of IVR [13], as the flow path formed between the insulation enclosure and the reactor vessel enhanced the CHF due to the impinging fluid on the bottom of the reactor vessel. The key factors for the success of the proposed strategy are the local thermal margin of the Critical Heat Flux (CHF) on the downward facing walls and the overall system behavior along the flow channel outside the reactor vessel. Previous researches [12,13] found that the CHF is lowest at the bottom and increases with an inclination angle [12,13]. By evaluating the natural convection in the molten fuel pool, Theofanous et al. [12] concluded that there is enough thermal margin at the bottom for AP600. The thermal margin to CHF was low at a higher inclination angle, in the region near the boundary of the oxidic layer and of the metallic layer.

In a two-phase natural circulation loop like external cooling of the reactor vessel, it is highly probable that the various types of two-phase flow instabilities might happen. The self-sustained flow oscillations may cause mechanical vibrations and affect the local heat transfer characteristics, which may induce a premature burnout. If the insulation structure were to fail due to mechanical vibrations in a manner that it touches the reactor vessel and/or the flow path were blocked, it would jeopardize the successful cooling of the reactor vessel.

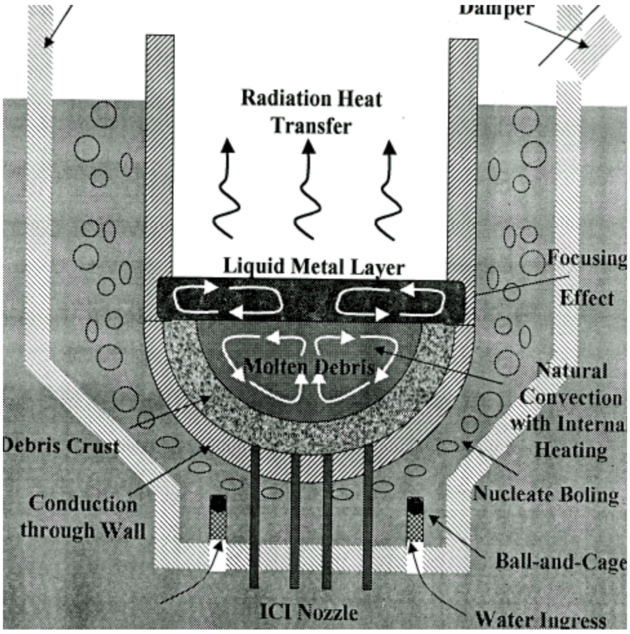

Fig. 1 Configuration of IVR

Song [8] evaluated the system behavior of ex-vessel cooling. The heat load on the reactor vessel, due to decay power and natural circulation in the molten pool, was simplified as uniform heat flux load at 600 kW/m^2 in the base case. It was shown that as the two-phase flow instability phenomena including the natural circulation oscillation and density wave oscillation affected the local thermal margins at the reactor vessel wall, the capability of natural circulation cooling was marginal at this level of heat flux. As the average heat flux for high power reactors, such as APR-1400, AP1000, and KSNPP, could exceed this range, a careful evaluation is necessary to warrant the success of the IVR strategy.

If we employ this strategy for high power reactors, there is a favorable effect that it increases the probability of the survival of the reactor pressure vessel. The adverse effect is that it increases the risk of ex-vessel steam explosion. As the dynamic load induced by a steam explosion of molten corium is still uncertain, this issue remains unresolved. Also, this strategy cannot be applied to the high-pressure scenario, such as station black out. The pressure difference between the system and containment could threaten the integrity of the reactor vessel due to thermal shock induced cracks, when the cold water contacts with the hot reactor vessel wall.

Also, it is to be noted that the IVR strategy is effective only when the reactor coolant system is fully depressurized. Otherwise, the film boiling heat transfer will occur on the surface of the reactor vessel wall. The surface temperature of the reactor vessel could be above the Leidenfrost temperature.

THE GAP COOLING

The gap cooling mechanism is suggested as an inherent safety feature to arrest the melt attack on the reactor vessel wall. As part of the experimental program named SONATA [3,17,18], the LAVA (Lower-plenum Arrested Vessel Attack) experiment was performed to prove gap formation and to investigate the gap cooling mechanisms using a simulant.

In the early series of the LAVA experiments, about 30 kg of Al_2O_3/Fe thermite melt (or Al_2O_3 only) was used as a corium simulant [17]. The melt was poured into a lower plenum of a reactor vessel of 1/8 scale. The experimental results indicate that the gap is formed between the debris and the inner surface of the lower head. The thermal response of the lower head vessel was mainly affected by the heat removal characteristics through this gap, which were determined by the amount of water ingression into the gap. The heat removal capacity through the gap in the LAVA experiments was estimated to be in the rage of 70 to 470 KW/m^2.

LMP200 experiments were performed using 200 kg of Al_2O_3/Fe thermite melt [18]. The melt was poured into a lower plenum, which is part of the 1/5 linear scaled reactor vessel, filled with water. The effect of melt mass and vessel dimension on gap formation and also the thermal behaviors of the vessel were examined. The unique characteristics of gap cooling were not clear at the bottom of the vessel in the LMP200-1 and LMP200-2 test. The gap formation was not continuous at the bottom of the vessel, where Fe melt was in contact with the inner surface of the vessel. It indicated that the increase in melt mass made it difficult for water to penetrate into the bottom of the vessel induced by the counter-current flow limits phenomena. Based on the partial cooling near the upper part of the vessel observed in the LMP200 series experiment, it is suggested that the possibility of effective cooling in the gap highly depends on the melt mass relocated into the lower plenum and the continuous gap formation.

The gap cooling is effective for high-pressure scenarios, as the heat removal capability of gap cooling increases with pressure. As the EVRC is effective for the low-pressure scenario, gap cooling does not help the EVRC. Another limitation is that as the experiment is highly transient and does not use prototypic material, direct applicability to a prototypic condition cannot avoid large uncertainties.

STEAM EXPLOSION

The energetic interaction between the molten reactor material and coolant was identified as a major risk among various risks of the severe accidents [19]. It could result in an early failure of the reactor pressure vessel and/or the containment. Experimental and analytical research [20-25] on the steam explosion has been performed widely during the last decades. Due to these efforts, it was possible to conclude that the probability of an alpha-mode failure is very low and the consequence of fuel coolant interactions (FCIs) seems to be not as catastrophic as previously thought. There are, however, a number of unresolved issues important to safety. In a broad sense, steam explosion occurring during the penetration of a large amount of core melt into a pool of water is considered as a bounding case for FCI in reactor conditions. The ex-vessel steam explosions might occur because (1) Failure of the lower head in case of core melt down cannot be ruled out for most reactors; (2) Absence of water in the cavity cannot be guaranteed at the time of lower head failure; (3) Flooding the cavity in case of expected vessel failure is already part of some SAM strategies; (4) Use of water to cool down the debris in the cavity could be generalized if it is demonstrated that energetic FCI has no consequence on the containment integrity.

The situations of concern for ex-vessel energetic FCI are unconstrained large pours of UO2-ZrO2-Zr-Steel melts in a deep pool of sub-cooled water at a low system pressure and a moderate pressure. The single pour of some tens of centimeters in diameter, which could be side pour, or the jets falling at once as a compact array are believed to be the worst case scenario. The in-vessel steam explosion has an impact on alpha-mode failure, lower head failure mode, the damage on piping and the steam generator, and debris distribution. The situations of concern for in-vessel energetic FCI are gravity pours of UO_2-ZrO_2 melt in the lower head in a ~2-m-deep pool of saturated water at a moderate system pressure.

Korea Atomic Energy Research Institute (KAERI) launched a research program on the steam explosion named "Test for Real cOrium Interaction with water (TROI)" in 1997 [4, 5]. The program is pursued not only to investigate the fundamental issue of the explosivity of reactor material but also to contribute to the development of the severe accident management strategy for the Advanced Light Water Reactors (ALWRs). For these reactors, the external cooling of the reactor vessel is considered a severe accident management strategy to achieve in-vessel-retention (IVR) of the molten core [26]. The TROI program is intended to provide adequate experimental data to enable proper evaluation of structural loadings to either the reactor pressure vessel or the containment.

Firstly, a series of experiments using ZrO_2 has been performed [4]. The inductive skull melting was used for the melting and delivery of molten material. In the tests using several kgs of ZrO_2 where the melt water interaction is made in a heated water pool at 30 ~ 95 ^oC located in a pressure vessel, either a quenching or a spontaneous steam explosion was observed. The spontaneous explosion observed in the present ZrO_2 melt/water experiments indicated the potential explosivity of corium. After preliminary tests using ZrO_2, experiments using a mixture of ZrO_2 and UO_2 [5] were performed. About 4 ~ 9 kg of corium melt jet is delivered into a sub-cooled water pool at atmospheric pressure.

Spontaneous steam explosions are observed in four tests out of six. The dynamic pressure, dynamic load, and morphology of debris is clearly indicated in the cases with steam explosion. The fact that reactor material resulted in a spontaneous explosion is a very important observation, as it deviates from the observations of previous experiments. The melt super heat, hydrogen generation, and the melt pool water interaction geometry are identified as the potential causes of this explosivity.

The experimental observations in TROI indicate that the non-explosivity argument for corium water interaction is not valid. Therefore, the steam explosion load should be properly addressed. However, the quantification of the steam explosion load is predicted with a high degree of uncertainty and with large deviations among various approaches. As a result, an international collaboration on Fuel Coolant Interaction (FCI) research called SERENA [27] is launched to obtain a convergence on the understanding of FCI processes and energetics. As part of the project, an analysis of the pre-mixing phase [28] was performed by using a TEXAS-V [29] computer code. By performing simulations for FARO L-14, L-28, and L-33 experiments, the capability and the limitations of the TEXAS-V fragmentation models for the premixing phase are investigated. The old break up model and new break up model are used for comparative simulations. As those experimental data sets cover a wide range of ambient pressures, sub-cooling of the water pool, and the melt jet diameters, the results of the simulation were beneficial in assessing the TEXAS-V code's capability. However, the prediction of the break up model was so different among the experiments that the extrapolation of the model to a prototypic condition without improving the current break up model would accompany a large uncertainty in the prediction of the pre-mixing behavior. As the initial conditions set by the pre-mixing phase determines the steam explosion load directly, the evaluation of steam explosion load might not be able to avoid large uncertainty. This situation would not be much different in case of the other computational models.

HYDROGEN BURN

The survivability of equipment is very important during the mitigation of severe accidents. The issue was addressed for existing plant [30] and new plant APR1400 [31]. As the existing equipment and instrumentations are utilized, the pressure and temperature profile during the severe accidents were compared to the curve of the Equipment Qualification (EQ), which was developed by enveloping the pressure and temperature response during the design basis accidents. It turned out that the existing equipment and instruments can be effectively utilized to execute the severe accident management strategy [30,31].

The hydrogen burn is a new phenomenon, which was not considered for the qualification of existing equipment and instruments. So, industrial and regulatory bodies investigated the issue [9,10,11]. NUREG/4763 [11] indicated that single hydrogen burns in large dry containments do not pose a serious threat to the safety equipment but a dilemma occurs, however, when considering the effect of igniter induced multiple burns. It indicated that the absence of igniters can results in potentially detonable mixtures. But multiple burns induced by the use of an igniter can pose a threat to the safety equipment.

The installation of the quenching mesh between compartments or near equipment has been suggested to prevent flame propagation among compartments and/or to maintain equipment survivability [6]. At first, the experiment to determine the quenching distance for hydrogen gas under the severe accident conditions of the nuclear power plants was performed. From this test, effects of the steam addition and initial pressure on the quenching distance were experimentally investigated. The quenching distance is a key quantity that must be characterized in order to design the quenching mesh. It showed that the quenching distance of 0.3 mm for hydrogen gas is suitable for application in nuclear power plants. Based on this quenching distance, installation characteristics of the quenching mesh were analyzed in a simplified phenomenological analysis and through experiments that were carried out in a small compartment at pressure lower than atmosphere pressure [6]. A performance test of the quenching mesh that is designed from the MESG (Maximum Experimental Safety Gap), 0.3 mm for hydrogen gas, is conducted within a large closed model compartment near atmosphere pressure for application in nuclear power plants [7]. These two test results suggested that a quenching screen could protect the essential equipment during hydrogen burn by stopping the flame propagation, though further investigations are still necessary for the practical implementation.

SUMMARY AND CONCLUSION

>From the synthesis of the results from the experimental and analytical researches including Simulation Of Naturally Arrested Thermal Attack in vessel (SONATA), Test for Real cOrium Interaction with water (TROI), and the test for the performance of quenching screen, it is concluded that

• The margin of the failure of the reactor vessel is challenged for high power reactors when we employ the strategy of In Vessel Retention (IVR) by Ex- Vessel Reactor Cooling (EVRC) due to premature dry out due to two-phase flow instability and the gap cooling alone is not sufficient to remove decay heat from the molten core

• The observation of the energetic spontaneous steam explosions in TROI suggests that proper measures should be taken to avoid and/or lessen the risk of in-vessel or ex-vessel steam explosion, when we employ ex-vessel cooling strategy and/or in vessel injection strategy for ALWRs.

• The dilemma that the absence of igniters can result in potentially detonable mixtures, while multiple burns induced by the use of an igniter can pose a threat to the safety equipment, could be solved by use of a quenching screen, as the results of basic experiments and laboratory scale experiments showed the feasibility of using a quenching screen to protect the essential equipment during hydrogen burn by stopping the flame propagation.

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