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Review of IVR-ERVC and using flooding concept for application to high power reactor

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1. Introduction

In Fukushima accident, decay heat was not adequately removed and melt-through happened. The Reactor Pressure Vessel (RPV) was failed by thermal load and large amount of radioactive material leaked into the environment. If the RPV was maintained with proper cooling, the accident could be mitigated within the pressure boundary. To protect the vessel, In-Vessel Retention through External Reactor Vessel Cooling (IVR-ERVC) concept was suggested. Decay heat from the corium is removed from boiling heat transfer on the external surface of the RPV.

It has a lot of advantages to maintain the integrity of the RPV. For environmental point of view, there is no difference between corium retained in the RPV or retained in the containment building. However, considering cleanup the corium, it is much better for corium to be retained in the RPV. Accident scope will be limited in the RPV. For example, in case of Fukushima, they have difficulties for cleanup the accident and even catching the location of the melt-through corium. Therefore, IVR-ERVC is the right strategy for mitigation of the severe accident. However, in case of high power reactors, there is a Critical Heat Flux (CHF) problem in its application to high power reactor. If CHF occurred, boiling regime changes from effective nucleate boiling to ineffective film boiling, so temperature of the RPV goes up and finally the RPV fails.

To solve the CHF problem, here have been a lot of works for IVR-ERVC. In the point of in-vessel heat transfer, Theofanous suggested risk oriented accident analysis methodology which is a combination of probabilistic and deterministic approach [1]. Esmaili introduce conditional failure probability which is a ratio for heat flux to CHF and calculate by in-house code in case of AP 1000 case [2]. A lot of experiments have been done using simulants of corium in various experimental apparatus. Their simulants were usually water due to simulate large Rayleigh number and natural circulation of corium. In ACOPO and mini-ACOPO, water and Freon-113 was used as simulants for corium [3]. COPO used water and BALI used salt added water as simulants [4, 5]. In these experiments, in-vessel heat transfer by natural circulation was studied. In case of SIMECO, KNO_3-NaNO simulants was used to simulate stratifying behavior of corium [6]. They observed bad heat transfer rate to bottom of the RPV in 3-layer case.

Unlike previous cases using simulants, RASPLAV project used real corium in relatively large scale about 200kg of corium mass [7, 8]. Because it was real corium, chemical behavior was also observed. Due to the smaller size of the experimental apparatus than real case, although Rayleigh number was not enough large like in real situation, they observed similar natural circulation behavior and it provided a basis for many experiments using simulants. They also observed 3-layer system called as layer inversion and the lowest layer was made by reaction between not completely oxidized U and molten steel. However, after oxidation of Zr was completed, iron came back to light metal layer.

In case of growth of corium pool, LIVE experiment was conducted. They used corium simulants as KNO_3-NaNO. Final heat flux profile and thickness of crust was not significantly affected by melting point lowering by eutectic phenomena. Initial temperature of corium pool was related to growth rate of the crust, and not significantly affected to steady state. If external wall of the RPV was cooled early, crust grow faster and have lower thermal conductivity.

In case of cooling at the external wall of the RPV, ULPU experiment was conducted in full scale of AP 600 and AP 1000 [10]. In this experiment, they insisted that by changing curvature of the RPV, CHF problem in AP 600 could be solved. In case of AP 1000, safety margin for CHF can be secured by surface effect like coating or painting in addition to geometry change of thermal insulator structure. Yang et al. conducted experiments on the RPV with insulation structure and observed effects of surface coating [11]. CHF was enhanced by coating, and effect of improved insulation structure enhanced CHF a lot. CHF value was varied depending on the angular position and the lowest point was near the shear key region. They also observed effect of coating by hydrated aluminum oxide on the aluminum heater and observed significant increase of local CHF limit varying depending on angular location [12]. In SULTAN experiments, sliced planed was tested with inclination angle by flow boiling [13]. Over $1MW/m^2$ heat flux could be removed by water cooling. It showed good accordance with ULPU experiments. And 3-D two phase code CATHARE was used to validate experimental results [14]. Comparing to 1-D, 3-D model was accurate when interfacial forces (drag, buoyancy, lift, etc.) is dominant. In Korea, KAIST experiment using 2-D slice test section of APR 1400 was conducted [15, 16]. It observed slightly lower CHF limit compared to ULPU experiments. It caused by smaller gap, structural material of the experimental loop, difference between forced and natural convection. There was flow fluctuation caused by counter flow, before 100kg/m^2 mass flow rate. CHF limit increased as mass flow rate increases and SULTAN correlation showed good accordance with experimental results in high mass flow rate condition over 150kg/m^2. An experiment using coolant as same as power plant, TSP and boric acid added water was conducted [17]. In case of TSP, about 50% CHF limit was enhanced and in case of boric acid, about 15% decrease, in case of both, about 35% enhanced. CHF enhancement ratio was not significantly different along angle or mass flux.

Visualization analysis was also conducted. By Chu et al., coolant behavior in real scale of AP 600 was observed and assured similarity with CYBL experiments [18]. They observed 4 stages of bubble behavior; direct contact of liquid-solid, bubble generation and growth, bubble coalescence and bubble departure. Frequency of bubble generation increased as heat flux increases and increasing rate was inversely proportional to heat flux. Bubble behavior showed more regular patterns in higher heat flux. In HERMES experiment, which has half scale of APR 1400 with air injection, chocking phenomena near shear key region were observed [19]. If air injection rate was large and exit area was small, recirculation was observed in shear key region. Therefore, shear key design needs to be revised. In CASA experiment, 1/10 scale but three dimensional apparatus, bubble behavior was also observed [20]. Heat flux and temperature was calculated before experiments using ANSYS. It showed lower CHF limit than ULPU experiments, however, in real scale, it was expected that CHF limit is higher than experiments due to linear scaling. CHF occurred at out of shear key region and propagated to shear key region. If inlet subcooling increase, porosity and quality decrease, then CHF could be enhanced.

ERVC concept was selected as severe accident management system in low power reactors up to 1000MWe and proven [21]. Even CAP 1400 which has 1400MWe power, selected ERVC and ERVC strategy for CAP 1400 is under research. Similarly, APR 1400 selected ERVC as severe accident mitigation system. However, according to Cho et al., CHF limit was violated in most of accident scenarios due to focusing effect of heat flux [22].

To solve CHF problem by mitigating focusing effect, material fin concept will be used in this study. Material was selected as gallium and dowther-rp oil by Prandtl number, which can be indicator for heat transfer mechanism. Gallium has low Prandtl number so conduction will be dominant, and oil has high Prandtl number so natural convection will be dominant. CHF will be determined by experiments and heat flux profile will be obtained by CFD analysis which has same condition with experiment.

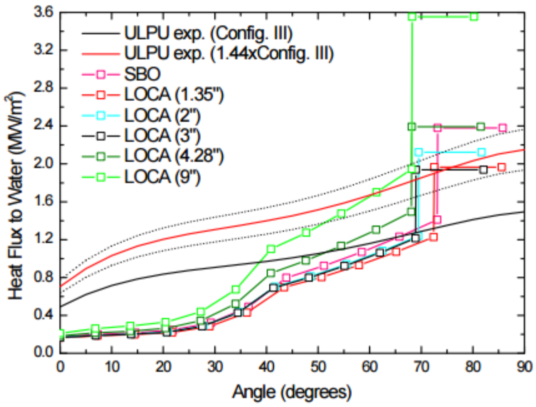


Fig. 1 Heat flux on the external surface of the RPV in various accident in APR 1400 [22]

2. Experimental and CFD analysis condition

2.1 Experimental Apparatus

The equipment is divided into three parts as boiling pool, heater and upper instruments. Boiling pool has rectangular shape and consisted of stainless steel structure and transparent glass window for visualization. Visualization was conducted through side faces and the window on the bottom face is for well-lighting. Indirect heating was selected for simulating decay heat from the corium. Heater block is made of copper and it has the shape of cylinder with hemispherical end, which is similar to the shape of the RPV. Actual heat source were cartridge heaters inside the heater block and thermocouples are sheathed in the heater. Radius of heater is about 74 mm, which is about 35 times scale down from the RPV of APR 1400. Upper instruments are upper deck and condenser. Upper deck supports the cooper heater and penetrated by power cable and thermocouple line going out from heater to outside through upper deck. Reflux coil condensers were used to condense vapor of working fluid, and final heat sink is tap water running inside the coil of the condensers. Working fluid was selected as R-123 due to lower boiling point and latent heat compared to water. So we could get CHF relatively lower power and lower temperature than water, which is original working fluid of IVR-ERVC.

2.2 Experimental Condition

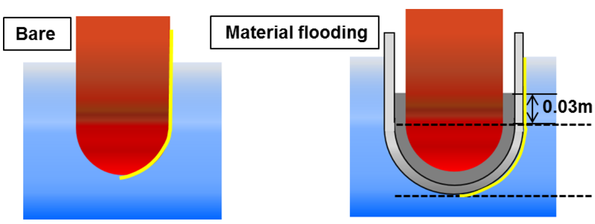


Fig. 2 Experimental condition schematic

Figure 2 is schematic diagram of bare and flooding condition. By 30mm flooding, surface enlarged by 1.79 times. Therefore, average heat flux will decrease.

2.3 CFD analysis condition

In experimental apparatus, heat source is cartridge heater sheathed in the copper heater block. Therefore, heat source in CFD was given as surface heat source interface of the cartridge heater and copper heater block. Top boundaries are exposed to the atmosphere so heat loss to the top boundaries is negligible, adiabatic condition. Boiling is simulated as constant temperature, which was the surface temperature in the experiments.

3. Results & Discussion

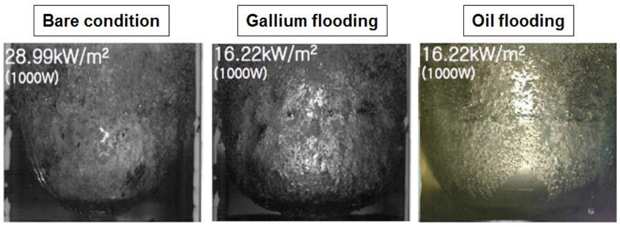
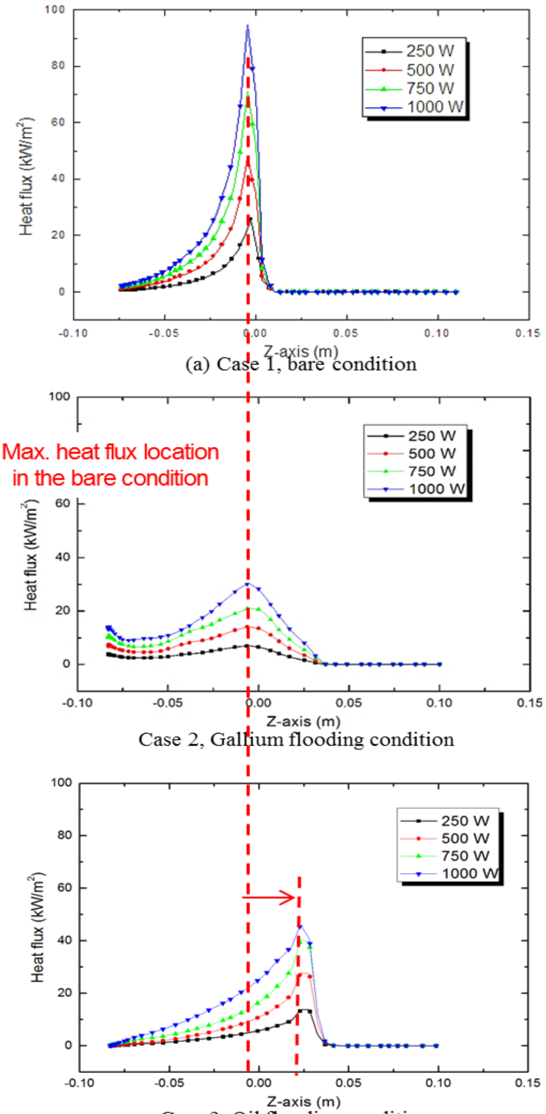


Fig. 3 Experimental condition schematic

Visualization results are shown as in figure 3. In bare condition, CHF occurred. However, in flooding condition, CHF did not occur. From this, it can be inferred that the flooding layer is effective in mitigation of the focusing effect. Compared to the gallium flooding, oil flooding showed more uneven nucleate site distribution. It can be inferred that heat flux distribution through gallium was more uniformly distributed than that of oil. Maximum heat flux in the focusing effect area will be lower in gallium than oil.

Heat flux at the external surface was obtained by CFD analysis. Bare case showed focusing effect of heat flux at the end of hemisphere region. This heat flux was achieved by internal structure and arrangement of the cartridge heaters in the heating block. In both materials, the maximum heat flux decreased commonly. From bare condition 95 kW/m^2, heat flux reduced to about 30kW/m^2 for gallium and about 46kW/m^2 for oil. Heat flux was more mitigated in gallium. However, heat flux distribution was quite different. In case of gallium, the location of the maximum heat flux was same with bare case, meaning focused heat flux was mitigated mainly by diffusion through flooding layer. On the other hand, in case of oil, the location of maximum heat flux was upwardly shifted. It means that the dominant heat transfer mechanism through oil is natural circulation. Natural circulation flow and upward shifting of heat flux distribution are illustrated in figure 5. Oil goes upward along heater surface and goes downward along the inner surface of the cap structure.



(a) Case 1, bare condition

Case 2, Gallium flooding condition

Case 3, Oil flooding condition

Fig. 4 Heat flux at the external wall with various condition

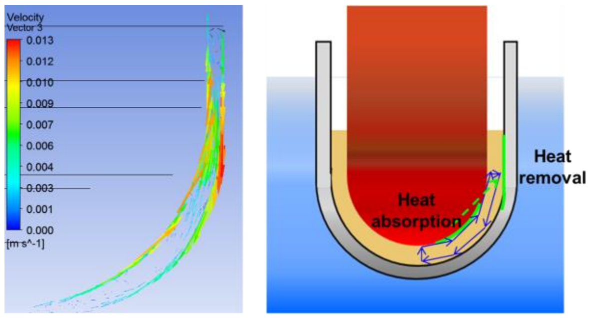
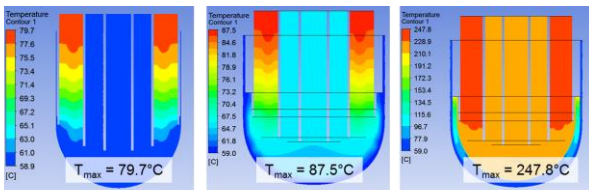


Fig. 5 Heat flux at the external wall with various condition



(a) Bare (b) Gallium flooding (c) Oil flooding

Fig. 6 Overall temperature distribution for each case

Temperature inside heater block was different for each case. In 1000W power, the maximum heater temperature of bare and gallium flooding condition was not much different. However, In case of oil, the maximum heater temperature was significantly higher than other two cases. It is caused by lower thermal conductivity of the oil. Although focusing effect could be mitigated, heater temperature was increased because oil flooding layer acted as a thermal resistance.

Comparing gallium and oil, gallium shows better overall performance of mitigating focusing effect and lowering the RPV temperature. In lower temperature, the integrity of the RPV could be preserved in long term point of view. However, in compatibility and economic aspect, oil is better. It is lighter than gallium so it applies lighter load for structures. Furthermore, oil is not corrosive as gallium with structural material, and much more inexpensive than gallium.

4. Conclusions

IVR-ERVC concept has been researched for a long time. For in-vessel heat transfer, simulants or real corium was used to get a heat flux distribution to the outer wall. And based on those results, ex-vessel cooling has been researched in various geometry to get cooling limit as CHF.

Material flooding is suggested as improvement of ERVC in APR 1400 to secure safety margin for CHF. Regardless of Prandtl number of the flooding material, the focusing effect of heat flux was mitigated; the maximum heat flux was reduced less than half of the maximum heat flux in bare condition. Therefore, material flooding is the solution for application of IVR-ERVC to high power reactor.

REFERENCES

[1] T. G. Theofanous, C. Liu, S. Additon, S. Angelini, O. Kymäläinen, T. Salmassi, In-vessel Coolability and Retention of a Core Melt, Nuclear Engineering and Design I-48, 1996.

[2] H. Esmaili and M. Khatib-Rahbar, Analysis of Likelihood of Lower Head Failure and Ex-Vessel Fuel Coolant Interaction Energetics for AP1000, Nuclear Engineering and Design, Vol.235, p.1583-1605, 2005.

[3] T. G. Theofanous, M. Maguire, S. Angelini and T. S. Salmassi, The First Results from the ACOPO Experiments, Nuclear Engineering and Design, Vol.169, pp.49-57, 1997.

[4] M. Helle, O. Kymäläinen, and H. Roumisto, Experimental data on heat flux distribution from a volumetrically heated pool with frozen boundaries, Proceedings of the Workshop on in-vessel core debris retention and Coolability, 1999.

[5] J. M. Bonnet, 'Thermal hydraulic phenomena in corium pools: the BALI experiment,' Proc. Workshop on Severe Accident Research (SARJ-98), 1999.

[6] B. R. Sehgal, Natural Convection Heat Transfer in a Stratified Melt Pool with Volumetric Heat Generation, MASCA Seminar, Aix-en-Provence, France, 2004.

[7] V. G. Asmolov, RASPLAV Project: Major Activities and Results, OECD/NEA Rasplav Seminar 2000, Munich, Germany, November 14-15, 2000.

[8] V. G. Asmolov, D. Tsurikov, MASCA Project: Major Activities and Results, OECD/NEA MASCA Seminar 2004, Aix-en-Provence, France, June 10-11, 2004.

[9] F, Kretzschmar, B. Fluhrer, Behavior of the Melt Pool in the Lower Plenum of the Reactor Pressure Vessel-Review of Experimental Programs and Background of the LIVE Program, FZKA 7382, 2008.

[10] T. N. Dinh, J. P. Tu, T. Salmassi, T. G. Theofanous, Limits of Coolability in the AP1000-Related ULPU-2400 Configuration V Facility, Proceedings of the 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-10), Seoul, Korea, 2003.

[11] J. Yang, F. B. Chung, J. L. Rempe, K. Y. Suh and S. B. Kim, Critical Heat Flux for Downward-facing Boiling on a Coated Hemispherical Vessel surrounded by an Insulation Structure, Nuclear Engineering and Technology, vol.38 No.2 Special Issue on ICAPP '05, 2006

[12] J. Yang. M. B. Dizon, F. B. Cheung, J. L. Rempe, K. Y. Suh, S. B. Kim, CHF enhancement by vessel coating for external reactor vessel cooling, Nuclear Engineering and Design 236 (2006) 1089–1098 2006.

[13] S. Rouge, SULTAN test facility for large-scale vessel coolability in natural convection at low pressure. Nuclear Engineering and Design. Vol. 169, pp. 185–195, 1997.

[14] S. Rouge, I. Dor, G. Geffraye, Reactor vessel external cooling for corium retention SULTAN experimental program and modeling with CATHARE code. In: Workshop on In-Vessel Core Debris Retention and Coolability, Organization for Economic Co-operation and Development, NEA/CSNI/R(98), Garching, Germany, 1998.

[15] Y. H. Jeong, W. P. Back, S. H. Chang, CHF Experiments for IVR-EVC using 2-D Slice Test Section, Proceedings of the Korean Nuclear Society Spring Meeting Kwangju, Korea, 2002.

[16] Y. H. Jeonng, S. H. Chang, Critical heat flux experiments on the reactor vessel wall using 2-D slice test section. Nuclear Technology, 152, 162-169, 2005.

[17] T. I. Kim, J. M. Park and S. H. Chang, CHF experiments using a 2-D curved test section with additivies for IVR-ERVC strategy, Nuclear Engineering and Design, 243, pp. 272-278, 2012.

[18] T. Y. Chu, J. H. Bentz R. B. Simpson, Observation of the boiling process from a large downward facing torispherical surface. In: Proceeding of National Heat Transfer Conference, American Nuclear Society, Portland, OR, USA, 1995

[19] K. S. Ha, F. B. Chung, R. J. Park, S. B. Kim, Evaluations of two-phase natural circulation flow induced in the reactor vessel annular gap under ERVC conditions, Nuclear Engineering and Design 253 (2012) 114– 124, 2012.

[20] S. W. Noh and K. Y. Suh, 2013, Critical heat flux for APR1400 lower head vessel during a severe accident, Nuclear Engineering and Design 258 (2013) 116– 129, 2013.

[21] O. Kymäläinen, O, H. Tuomisto, T.G. Theofanous, In-vessel retention of corium at the Loviisa plant, Nuclear Engineering and Design 169(1997)109-130, 1997.

[22] Y. J. Cho, H. S. Park, M. H. Choi, S. W. Hwang, H. K. Noh, Development of Assessment Methodologies for Corium Coolability, KINS/HR-1410, Korea Institute of Nuclear Safety, 2015.