Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 29-30, 2014

Reactor Core Failure Analysis for Feasibility Study of IVR-ERVC Strategy

Kukhee Lim [a*], Yongjin Cho [a], Taesuk Hwang[a]
[a]*Korea Institute of Nuclear Safety, P.O. Box 114, Yusong, Taejon, 305-600, Korea*
[*]*Corresponding author: limkh@kins.re.kr*

1. Introduction

An IVR-ERVC strategy is to flood the reactor cavity and submerge the reactor vessel for maintaining its integrity. This is considered as a severe accident management strategy for several recent nuclear power reactors and as severe accident mitigation strategy which is included in SAMG for operating nuclear power reactors.

The concept of IVR-ERVC is realized by circulation of coolant and the motivation is steam bubbles on outside wall of reactor vessel. The complicated physical phenomena in a reactor vessel under severe accident environments should be evaluated by effective cooling methods at the same time to satisfy thermal failure margin of the strategy. The reactor integrity by this margin criterion is guaranteed if the heat flux obtained by the analysis of heat balance equations between heat structures in a reactor vessel does not exceed the critical-heat-flux (CHF) limit for nucleate boiling on the vessel outer surface.

The representative assessment method of heat flux at a reactor vessel is a lumped-parameter method (LPM) and it is used to review IVR-ERVC design of Westinghouse AP600 and AP1000. This method assumes the layer configuration of molten pools (number of layers, thickness of layers and heat generation rate of molten corium, etc.) and evaluates representative states by a 1-dimensional heat transfer analysis. Boundary conditions of model should be well defined to increase accuracy of assessed heat flux and these are dependent on accident scenarios. Therefore, conservative assumptions or results from the analysis using system codes for accident analyses should be considered to determine boundary conditions. In this paper, in-vessel molten corium behaviors during LBLOCA which is considered as the most conservative accident scenario in IVR-ERVC design concerns are examined using MELCOR 1.8.6 to check the feasibility of APR1400 IVR-ERVC strategy.

2. Methods and Results

Several former analyses[1,2,3] which used same approach have been reviewed to determine the representative accident scenario of the analysis and LBLOCA which is conservative in terms of decay heat calculation is selected based on the reviews.

2.1 Analysis Code Description

MELCOR1.8.6 is fully integrated, engineering-level computer code that models the progression of severe accidents in light water reactor nuclear power plants. The modeling of a reactor core to simulate the IVR-ERVC strategy is treated in the MELCOR COR package. The MELCOR COR package is to calculate the thermal response of the core and lower plenum internal structures, including the portion the lower head directly below the core. The specific capabilities of the COR package are as follows;

- Heat transfer from the core to lower head
- Relocation of corium and debris of damaged core materials
- Modeling of fuel pellet & cladding, grid spacers, care baffle & formers, control rod, guide tube, molten pool and particulate debris, etc.

2.2 System Modeling

The RCS model includes the core, primary and secondary coolant systems. It also includes 2 steam generators, 4 reactor coolant pumps, and direct vessel injection from the Safety Injection System to the RCS (see Fig. 1). The 51-cell containment model consists of 32 subcompartments, 1 environment and the 18-cell IRWST with 3 axial levels in which 6 cells are azimuthally separated. And the information of accident scenario and core modeling is as follows (see Fig. 2);

- Accident scenario : cold-leg large break loss-of-coolant-accident
  (full depressurization and no safety injection, break area : 0.92 m^2)
- Reactor cavity flooding condition
  ･ Case 1 : cavity is not flooded
  ･ Case 2 : cavity is fully flooded from the initial time of the accident (t=0)
- Core modeling
  ･ Lower head : 10 temperature nodes and 8 segments (without penetrations)
  ･ Core cell : 6 radial rings

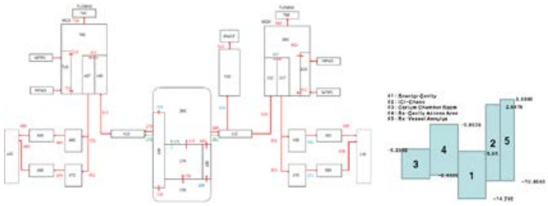

Fig. 1. APR1400 MELCOR1.8.6 nodalization

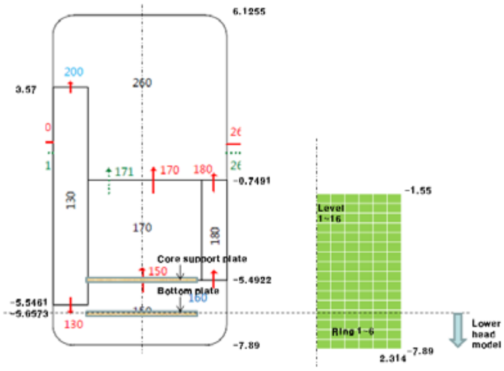

Fig. 2. APR1400 MELCOR1.8.6 core cell modeling

2.3 Results

According to the reactor cavity flooding conditions, two analysis cases are considered. The end of the analyses is determined by reactor vessel failure criteria of MELCOR code. The effect of reactor vessel cooling by cavity flooding is investigated. It is observed that little heat flux is transferred to the air in the cavity at Case 1. However, about 0.4 MW/m^2 heat flux in average in terms of radial ring of core cell model is observed at Case 2. The heat flux level at Case 2 shows that external-vessel cooling by coolant is well-modeled in the analysis.

The reactor vessel failure occurred at about 7e3 seconds because of creep rupture at the first segment of the lower head. Creep rupture failure in MELCOR COR package is predicted by Larson-Miller creep-rupture failure model which gives the time to rupture t_R(in seconds) as follows.

$$t_R = 10^{\left(\frac{P_{LM}}{T} - C_l\right)} \quad \text{(Eq. 1)}$$

where T : temperature (K)

P_{LM} : Larson-Miller parameter

The analyzed vessel failure time is 7.181e3 seconds at Case 1 and 7.662e3 seconds at Case 2. This is relatively longer than the vessel failure time (about 5e3 seconds at 9.6" LOCA) shown in the literature [3].

The relocated debris mass in the lower head at the time of vessel failure is analyzed. As shown in Fig. 5, 6 and Table I, very similar tendencies of mass increase are observed and their values are close each other. Since the time level at which creep rupture has been occurred in the analysis is relatively low comparing with that of whole ERVC process, it is considered that the effect of external vessel cooling is not fully reflected to have a remarkable differences.

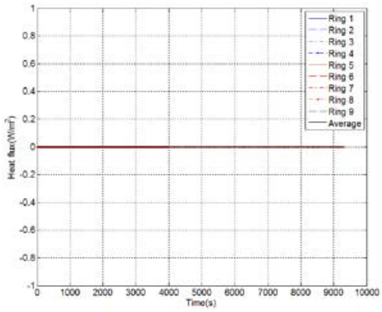

Fig. 3. Heat flux from lower head to fluid in the cavity (Case 1)

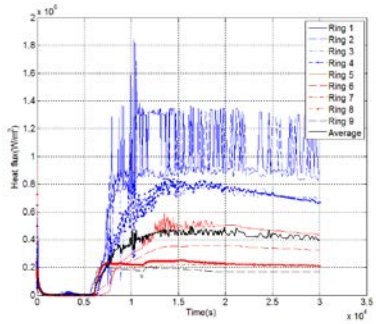

Fig. 4. Heat flux from lower head to fluid in the cavity (Case 2)

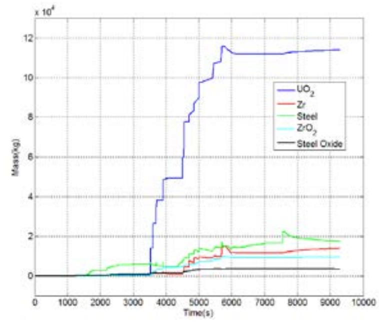

Fig. 5. APR1400 MELCOR1.8.6 core cell modeling

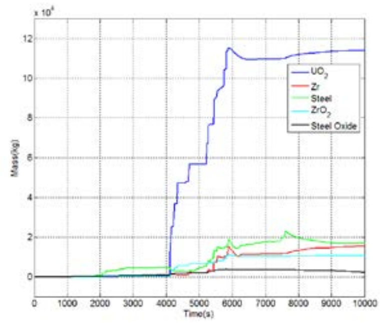

Fig. 6. APR1400 MELCOR1.8.6 core cell modeling

Table I: Comparison of relocated debris mass in the lower head (unit : kg)

	9.6" LOCA[3]	Case 1	Case 2
UO_2	1.073e5	1.178e5	1.178e5
ZrO_2	2.8e3	9.945e3	1.111e4
Steel Oxide	N/A	3.691e3	3.942e3
Zr	4.73e3	2.146e4	2.060e4
Steel	5e4	3.161e4	3.141e4

3. Conclusions

The relocated debris mass in the lower head of APR1400 reactor is analyzed using MELCOR1.8.6. This analysis is to determine the boundary conditions of the heat balance equations consisting of the lumped parameter method in order to calculate heat flux at external vessel wall surface. As a result, lower head vessel failure has been occurred at the time of about 7e3 which is very short time comparing with total period of ERVC process. Even though the effect of external vessel cooling is well-modeled, however, the differences between debris mass are relatively small. Therefore, the physical feasibility of the creep rupture model in MELCOR COR package should be verified for an adequate debris mass assessment in a reactor lower head under the ERVC condition.

REFERENCES

[1] T.G. Theofanous, C. Liu, S. Additon, S. Angelini, O. Kyma1ainen and T. Salmassi, In-Vessel Coolability and Retention of a Core Melt, DOE/ID-10460, ANL, 1995

[2] H. Esmaili and M. Khatib-Rahbar, Analysis of In-Vessel Retention and Ex-Vessel Fuel Coolant Interaction for AP1000, NUREG/CR-6849, ERI, 2004

[3] S. B. Kim and R. J. Park, etc., Optimization of the Severe Accident Management Strategy for Domestic Plants and Validation Experiments, KAERI/RR-2528, KAERI, 2004