

Numerical Assessment of Flooding Safety System Mitigation Performance During an Out-CV LOCA Scenario in a Small Modular Reactor using MELCOR code

Hyo Jun An^a, Jae Hyung Park^a, Yujin Kim^a, Sehee Kwon^a, Hyunjun Chung^a, Yuseong Go^a, JinHo Song^a, Joon Eon Yang^a, Sung Joong Kim^{a, b*}

^aDepartment of Nuclear Engineering, Hanyang University, 222 Wangsimni-ro, Seongdong-gu, Seoul 04763, Republic of Korea

^bInstitution of Nano Science and Technology, Hanyang University, 222 Wangsimni-ro, Seongdong-gu, Seoul 04763, Republic of Korea

*Corresponding author: sungkim@hanyang.ac.kr

***Keywords** : Out-CV LOCA, FSS, SMR, MELCOR, Severe accident,

1. Introduction

Safety assessment of a newly proposed reactor design is necessary for the further development and commercialization. As numerous types of small modular reactors (SMRs) are being developed and proposed, safety assessments of the SMR are being actively conducted using system codes such as RELAP5, MARS-KS, and SPACE [1-3]. Moreover, numerical investigations in terms of severe accident has been conducted [4-6] even though the risk of severe accident of the SMR is expected to be low enough.

Generally, a representative initiating event used for the safety assessment of an integrated pressurized water reactor (IPWR) type SMR is loss of coolant accident (LOCA). As the overall design of the IPWR type SMR became more simple and passive than large reactors, most of non-LOCA accidents share similar characteristics in terms of accidental behavior. Accordingly, many studies on the SMR safety analysis postulated LOCA scenario.

For IPWR type SMRs with containment vessel (CV) design, LOCA scenario can be categorized based on the break location, inside or outside of the CV. In this case, most of the studies with LOCA scenario assumes in-CV LOCA because out-CV LOCA is recognized to be unlikely to occur owing to components to prevent uncontrollable out-CV LOCA such as multiple CV isolation valves. So accident analysis on the out-CV LOCA scenario has limited literatures.

However, out-CV LOCA scenario is expected to accompany more challenging consequence due to the potential loss of coolant inventory in the reactor module (RM). IPWR type SMRs, such as NuScale US 460, use remaining inner coolant as a decay heat removal medium during an accident [7]. In other words, out-CV LOCA deteriorates the heat removal performance by releasing remaining coolant out of the RM.

Thus, this study investigated characteristics of the out-CV LOCA scenario and evaluated performance of our passive safety system using MELCOR code. For the study, we used a MELCOR input model of a typical natural circulation SMR.

2. Methodology

2.1 Reference reactor for the MELCOR analysis

NC-SMR, a IPWR-type SMR with thermal power rating of 330 MWth was selected as a reference reactor of this study. Figure 1 is a brief schematic of the NC-SMR. The NC-SMR adopted natural circulation of the primary coolant during normal operation condition. The NC-SMR has CV design, which encapsulates the reactor vessel (RV). The CV is maintained at weak vacuum condition to prevent heat loss during normal operation and potential combustion of flammable gas during an accident. Moreover, the NC-SMR adopted emergency core cooling system (ECCS), which depressurizes the RV using emergency depressurization valves (EDVs) and recirculates additional coolant in the CV into the RV through emergency recirculation valves (ERVs).

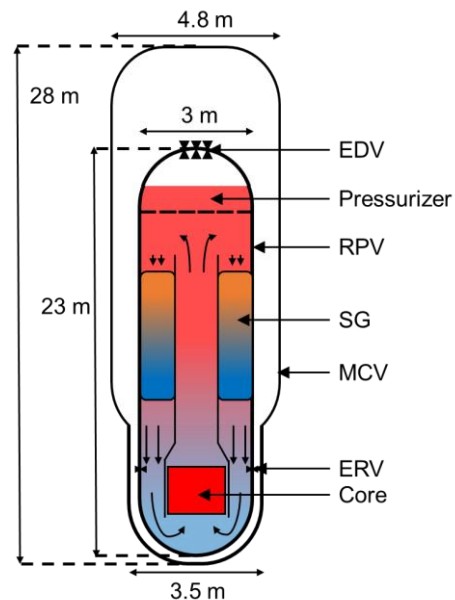


Fig 1. A conceptual schematic of NC-SMR

Flooding safety system (FSS), an innovative passive safety system shown in figure 2, is applied to the NC-SMR. The FSS provides on-demand supply of

emergency coolant into the reactor pit, where the RM is located. Accordingly, the reactor pit can be maintained at dry condition during normal operation, eliminating cons of submerged RM design such as maintenance of the RM and irradiation of the water. The FSS is composed of a large common pool and six flooding valves and six auxiliary pool. The common pool is designed with sufficient coolant inventory to flood every six reactor pits, reaching to 9,408 m³. When flooding of a reactor pit is demanded, a flooding valve automatically opens and the emergency coolant is passively flow into the reactor pit. As the flooding valves are designed with fail-safe system, the FSS can deal with station blackout (SBO) accident.

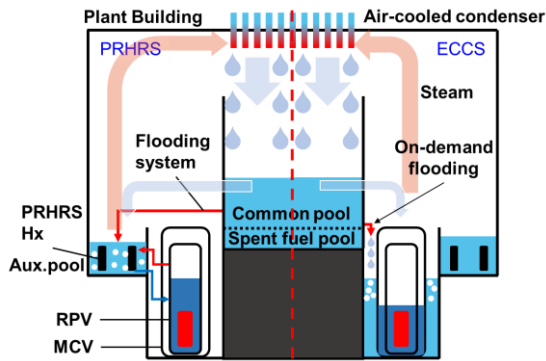


Fig 2. A conceptual schematic of FSS

In addition, auxiliary pools, containing additional emergency coolant, are installed beside each RM. In the auxiliary pool, a heat exchanger of passive residual heat removal system (PRHRS) is located. The PRHRS condensed steam in the secondary system to provide additional coolant to the steam generator (SG) during an accident. In this case, the auxiliary pool becomes a heat sink for the residual heat removal. When the heat sink is not enough and starts to deplete, coolant in the common pool can also be supplied to the auxiliary pool to sustain its coolability.

2.2. MELCOR input model of reference system.

In this study, a MELCOR input model of the reference system was used for the MELCOR calculation. Figure 3 shows the nodalization of the input model. Major components of NC-SMR such as reactor core, SG, pressurizer were modeled. ECCS was modeled with multiple valves. EDVs were set to open when pressurizer pressure decreases below 8.0 MPa while ERVs were set to open when the pressure difference between the end of the valve decreases below 0.1 MPa. CV was modeled with several axial nodes and two radial nodes which divided inner and outer channel of the CV. Cooling of the CV was calculated by heat structures between the CV and the reactor pit. The heat structures are comprised with 30 axial nodes to simulate varying heat transfer area with the water level of the CV and reactor pit.

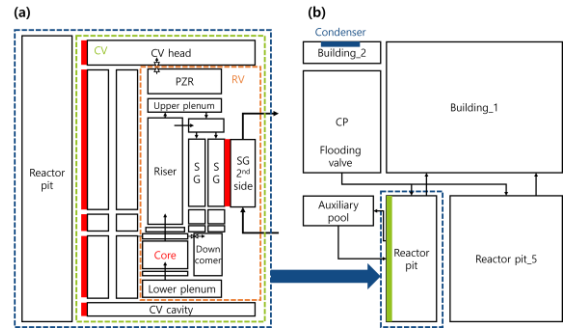


Fig 3. MELCOR nodalization (a) NC-SMR, (b) FSS

Major components of the FSS, common pool, flooding valves, and auxiliary pools, were modeled with reactor building components. The heat exchangers of the PRHRS were set to calculate heat transfer with the water in each auxiliary pool. The flooding valves PRHRS isolation valves were set to open with the reactor trip signal.

To attain reliability of the MELCOR input model, we conducted steady state calculation of the reactor. Table I shows the steady state operation parameters, and they showed a good accordance with the target operation parameters.

Table I: MELCOR input model steady-state parameters

Parameter	Steady-state value	Relative error
Core thermal power, MWt	330.06	0.02 %
Primary system pressure, MPa	15.5	0.00 %
Core inlet temperature, °C	256.53	0.79 %
Core outlet temperature, °C	329.00	0.72 %
Core inlet mass flow rate, kg/s	841.50	-1.88 %
SG steam pressure, MPa	3.7	0.18 %
Feed water inlet temperature, °C	150.03	0.02 %
SG outlet steam temperature, °C	298.98	0.76 %
Feed water flow rate, kg/s	141.63	-1.23 %

2.3. Accident scenario and MELCOR case modeling.

Target accident scenario of this study is out-CV LOCA which is initiated by a break at a connecting system such as chemical and volume control system (CVCS). As the CVCS connects the primary system with a system located out of the reactor module, some part of the CVCS pipe is expected to be exposed outer space. So we assumed break of a CVCS pipe at outer region, releasing primary coolant into the reactor pit, as shown in figure 4. Moreover, we assumed failure of isolation valves of the CVCS system.

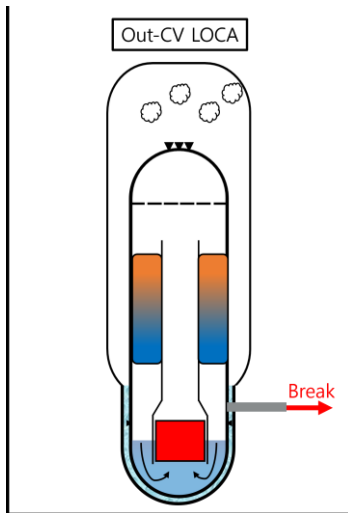


Fig 4. A schematic description of out-CV LOCA scenario

For the MELCOR case modeling, we modeled the break as a flow path that opens during normal operation. The diameter of the flow path is 2 inch and connected to the downcomer in the RV and the reactor pit.

3. Results and discussion

In this section, results of 2 MELCOR calculation results were presented and discussed to investigate characteristics and accident management performance of FSS.

3.1 Case 1

Case 1 of this study assumed no actuation of passive safety system, which means every flooding valves and PRHRS isolation valves were failed to open. Accordingly, insufficient accident management and core damage was expected. Table II shows the accident sequence of the Case 1. Figure 5 - # shows major results of the Case 1 MELCOR calculation.

TABLE II: Case 1 accident sequence

Event	Time (sec)	Time (hour)
CVCS out-CV region break	0.0	0.0
Reactor trip (low pressurizer level)	16	0.0
EDV open	535	0.149
ERV open	675	0.186
Core uncover	700	0.194
Core exit temperature > 923 K	1,130	0.311
Oxidation started	1,190	0.331
Gap release	1,307	0.363
Core support plate failure started	2,000	0.556
Core bottom dryout	3,392	0.942

Relocation to lower plenum started	4,274	1.487
Lower head failure	17,156	4.766
CV cavity dry-out	18,820	5.228

As shown figure 5, the RV pressure rapidly decreased and reached to the EDV actuation set point within about 500 seconds. After EDV open, RV-CV pressure equilibrium formed in about 150 seconds. CV water level also increased but could not reach to 0.4 m, the bottom elevation of the RV lower head. Due to the continuous coolant loss, RV water level decreased and the core was exposed at 700 seconds. As shown figure 6, core temperature increased and degradation started.

After about 1.5 hour from the accident, molten core started relocation to the lower head. Most of the core materials collapsed around 2.5 hour, heating the lower head structure. Eventually, the lower head failure occurred at about 4.8 hour after the accident.

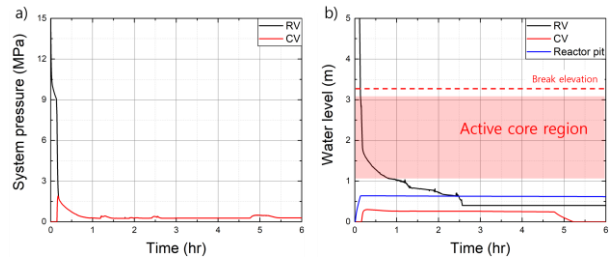


Fig 5. Case 1 MELCOR calculation results a) System pressure, b) Water level of RV, CV and reactor pit

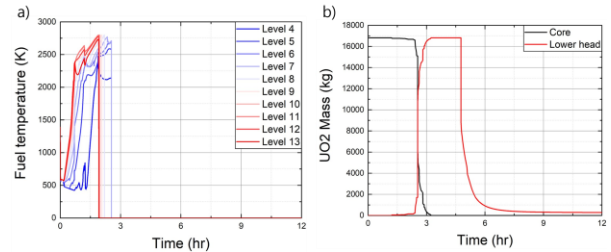


Fig 6. Case 1 MELCOR calculation results a) Core ring 1(inner most) axial fuel temperature distribution, b) UO₂ mass distribution

3.2 Case 2

In the Case 2, we assumed the FSS could successfully actuated during the accident. As soon as the reactor trip signal is generated, the flooding system for the reactor and the PRHRS isolation valves opened. Overall accident sequence of the Case 2 was shown in Table III.

TABLE III: Case 2 accident sequence

Event	Time (sec)	Time (hour)
CVCS out-CV region break	0.0	0.0

Reactor trip (low pressurizer level)	16	0.0
EDV open	375	0.104
ERV open	532	0.148
Core uncover	1,460	0.406
Core exit temperature > 923 K	1,880	0.522
Oxidation started	2,100	0.583
Gap release	2,367	0.658
Inverse flow from break	2,570	0.714
Core reflooding	3,440	0.956
Recirculation formed	11,730	3.26

Even though both accident management strategy were conducted, the reactor core was slightly damaged due to the LOCA. As shown in figure 7, the RV pressure showed stiff decrease, even faster than the Case 1 due to the cooling of the PRHRS. Water level of the RV also decreased rapidly. Accordingly, the reactor core was shortly uncovered due to the coolant loss. The core uncover increased the cladding temperature, inducing oxidation and gap release. The peak cladding temperature (PCT) reached about 1,400 K, as shown in figure 8. However, as the reactor pit flooded, supplied emergency coolant cooled the outer CV wall, depressurizing the CV and connected RV. The depressurization caused the emergency coolant to be injected into the RV.

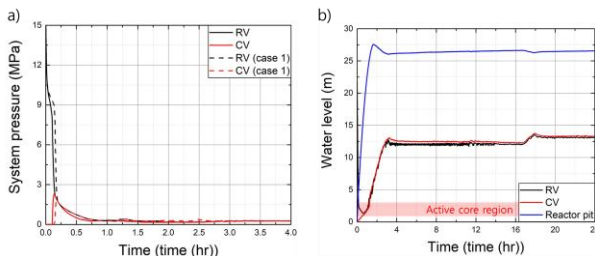


Fig 7. Case 2 MELCOR calculation results a) System pressure, b) Water level of RV, CV and reactor pit

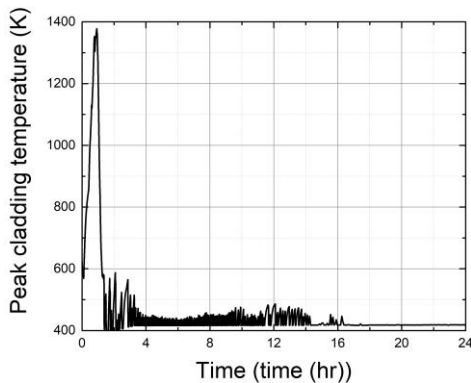


Fig 8. Case 2 MELCOR calculation results: peak cladding temperature

Consequently, a reverse flow through the CVCS pipe break was formed after 0.7 hour, as shown in figure 9. The injection of the water in the reactor pit increased the water level of the RV and the CV up to about 13 m. Even the integral mass flow through the break became negative after 2 hours, meaning the injected water was larger than the release water during the initial stage of the accident.

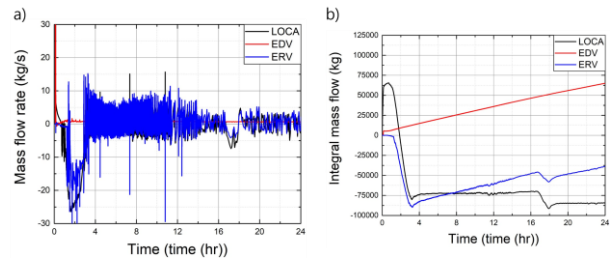


Fig 9. Case 2 MELCOR calculation results a) Mass flow rate of LOCA, EDV, and ERV b) Integral mass flow of LOCA, EDV, and ERV

Eventually, the core was reflooded and no further degradation of the core was observed. Decay heat from the core was continuously removed. Evaporated coolant was condensed on the CV inner wall surface and recirculated into the RV.

4. Conclusions

In this study, we investigated characteristics of out-CV LOCA scenario in a typical IPWR type SMR using MELCOR code. Also we conducted preliminary assessment of accident management performance of FSS, a new innovative passive safety system.

The out-CV LOCA scenario caused large release of primary coolant due to the pressure difference between the RV and reactor pit, causing extremely sharp decrease of the RV water level. The reactor core was exposed in a very short time interval, about 0.2-0.4 hour from the accident. When the passive safety systems were unavailable, the core water level could not be restored, and prompt core degradation followed, leading to the RV lower head failure.

On the other hand, when the passive safety systems were intact, the core uncover still observed but the emergency coolant in the reactor pit was injected into the RV and it reflooded the core. The core was slightly damaged but no further degradation was expected. This case should be pointed out because it implies that the out-CV LOCA scenario accompanies extremely rapid RV water decrease to cope with current passive safety systems.

To summarize, this study showed that the out-CV LOCA can accompany extremely hazardous consequence, and potential core damage even the FSS was intact and available. Moreover, there are uncertainties that the reverse flow through the break may not be formed due to the small size and long flow path

length of the CVCS pipe. Therefore, we should evaluate the risk of the out-CV LOCA scenario even though the possibility of the scenario is expected to be sufficiently low. Further studies on the uncertainties whether the break reverse flow also are needed under various condition and methodologies.

Acknowledgments

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT) and also the Innovative Small Modular Reactor Development Agency grant funded by the Korea Government (MSIT). (No. RS-2022-NR067165 and No. RS-2023-00259516)

REFERENCES

- [1] K.H. Bae, K.H. Cheol, C.M. Hee, S.S. Ku, Safety evaluation of the inherent and passive safety features of the smart design, *Annals of Nuclear Energy* 28 (2001) 333–349.
- [2] R.-J. Park, D. Son, H.S. Kang, S.M. An, K.S. Ha, Development of IVR-ERVC evaluation method and its application to the SMART, *Annals of Nuclear Energy* 161 (2021) 108463.
- [3] J. H. Park, J. Im, H. J. An, Y. Kim, J. I. Lee S. J. Kim, Development of an on-demand flooding safety system achieving long-term inexhaustible cooling of small modular reactors employing metal containment vessel. *Nuclear Engineering and Technology* 56(7) (2024) 2534-2544.
- [4] H. J. An, J. H. Park, C. H. Song, J. I. Lee, Y. Kim, S. J. Kim, Strategic analysis on sizing of flooding valve for successful accident management of small modular reactor. *Nuclear Engineering and Technology* 56(3) (2024) 949-958.
- [5] Campbell, S., Esmaili, H., & Schaperow, J. (2019). Independent MELCOR Confirmatory Analysis for NuScale Small Modular Reactor. Nuclear Regulatory Commission FSCB-18-01.
- [6] Ham, J., Kim, S. H., & Jeong, S. (2024, May). Preliminary severe accident analysis of INCV-LOCA in i-SMR using CINEMA code. In *Transactions of the Korean Nuclear Society Spring Meeting*.
- [7] M. Niemi, Simulation and safety features of NuScale small modular reactor, (2017).