Preliminary MELCOR analysis of i-SMR under hypothetical loss of coolant accident

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1. Introduction

In recent years, Small Modular Reactors (SMRs) have garnered considerable attention due to their inherent advantages over large-scale Nuclear Power Plants (NPPs), such as safety features, grid flexibility, potential for hydrogen production, and shorter construction time. These characteristics imply the higher applicability of SMRs in various countries, including developing nations considered as a promising market. Consequently, many countries worldwide are currently developing new SMRs [1–8], such as NuScale Power Module (NPM), Advanced Reactor Concepts-100 (ARC-100), GE BWRX-300, Westinghouse AP300 SMR, KHNP i-SMR, etc.

Among these, innovative SMR (i-SMR) has been developed since 2021 in Republic of Korea with strong SMR competitor [9]. With the electrical power output of 170 MWe, the i-SMR features a double containment design as shown in Figure 1. Notably, the incorporation of multiple passive safety systems, such as Passive Emergency Core Cooling System (PECCS), Passive Containment Cooling System (PCCS), and Passive Auxiliary Feedwater System (PAFS), ensures safety levels overwhelming conventional large-scale NPPs by more than 1,000 times, preventing the occurrence of severe accident. Additionally, its modular design allows for wide scalability by the deployment of multiple units to meet various energy demands, including flexible operation and hydrogen production.

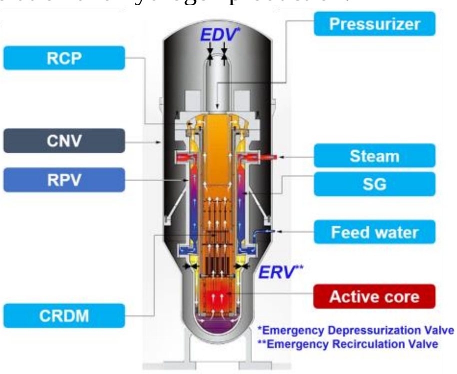

Fig 1. Schematic configuration of i-SMR with double containment geometry

The i-SMR has been developed to secure standard design approval by 2028. For this, the analysis results from the CINEMA code of comprehensive accident analysis code developed in Korea will be utilized. However, relying solely on the result by a certain code may have limitations in demonstrating the safety levels of i-SMR under accident situations. Hence, it is imperative to validate the performance of the i-SMR through a comparative analysis using multiple codes, such as MAAP, and MELCOR codes.

In this regard, this study was conducted as a preliminary accident analysis by using MELCOR code, a regulatory code utilized by the U.S.NRC. Given its widespread adoption by many regulatory and research institutions, its results are deemed sufficiently applicable for relevant licensing considerations. To perform MELCOR calculation, an i-SMR MELCOR input was developed. A hypothetical scenario with the release of coolant from the RCS into the containment interior was assumed as an initiating event for accident analysis. Additionally, the tendency of accident progression was compared to that predicted by CINEMA code calculations.

2. Methodology

2.1. i-SMR MELCOR input development

Figure 2, 3 shows the nodalization of the developed MELCOR input model and a representative safety system, respectively. As shown in Figure 1, Reactor Pressure Vessel (RPV) is surrounded by outer metal containment. For the PECCS, Emergency Depressurization Valves (EDVs) and Emergency Recirculation Valves (ERVs) are positioned at the top of pressurizer and at the upper side of active core, respectively. The EDV passively opens under accident environment to release coolant/steam into the containment, thereby depressurizing the primary system. Additionally, condensed steam from inner wall of the containment is recirculated back into the core through the ERV to prevent depletion of water in the core. Regarding the PECCS, the two heat exchangers inside the containment, connected to Emergency Cooling Tank (ECT) located in reactor building, functions to lower the temperature and pressure within the containment through natural circulation. The ECT also serves as a source for coolant injection during the operation of PAFS. In this study, it was assumed that all systems are not operational to conservatively evaluate accident progression.

Table I outlines the key design parameters obtained from steady-state calculation of the developed MELCOR input model during 2,000 s. The parameters of interest include the core output, system pressure, coolant mass flow rate, and inlet/outlet temperatures. The calculation was performed stably while maintaining a design pressure of 15 MPa and an output of 520 MW.

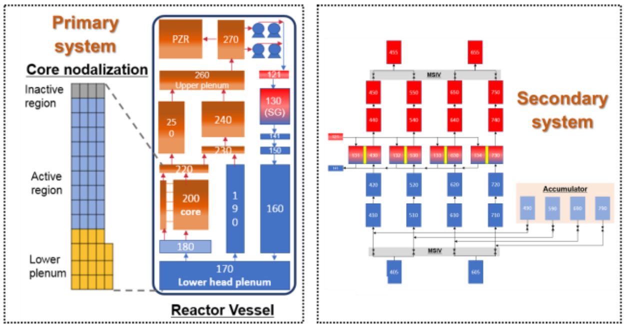

Fig 2. Nodalization of (a) primary system, (b) secondary system of i-SMR

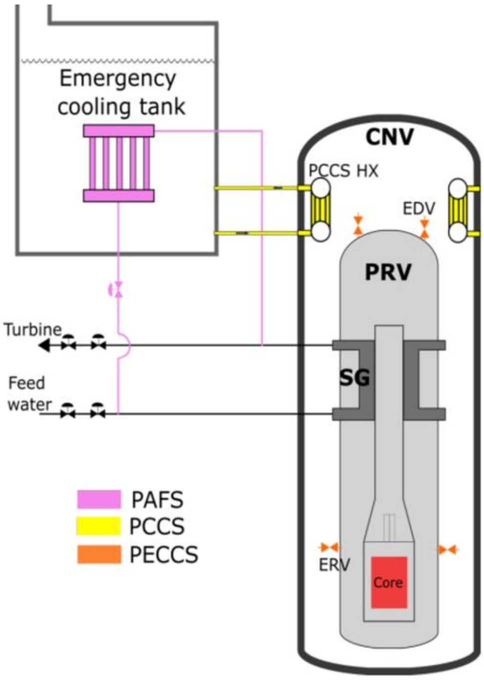

Fig 3. Passive safety systems of i-SMR

Table I: Design parameters obtained from MELCOR steady-state calculation

Design parameter	MELCOR i-SMR
Core power [MWth]	520
Primary pressure [MPa]	14.99
Primary system mass flow rate [kg/s]	1,523.4
Core inlet temperature [℃]	267.8
Core outlet temperature [℃]	305.6
Secondary system mass flow rate [kg/s]	38.92
Secondary system pressure [MPa]	4.8
SG Primary inlet temperature [℃]	310.6
SG Primary outlet temperature [℃]	267.9
SG secondary inlet temperature [℃]	217.2
SG secondary outlet temperature [℃]	256.5

2.2. Initiating event

The initiating event was assumed to be stuck open of one of the EDVs, resulting in coolant discharge from the primary system into the containment as shown in Figure 4. The size of the valve opening was assumed to be 2 inches. Under this circumstance, the ERV is opened by a low-level signal (PRZ water level < 40 %). Additionally, the PCCS was conservatively assumed to be inoperable. Figure 5 shows decay heat generation, mass flow rate, and energy discharge rate through the opened valves.

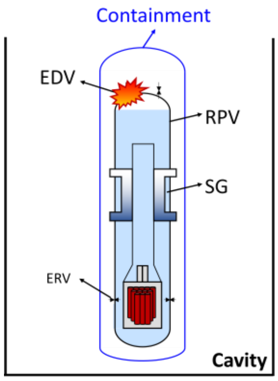

Fig 4. Release of coolant in RCS into containment under EDV stuck open accident

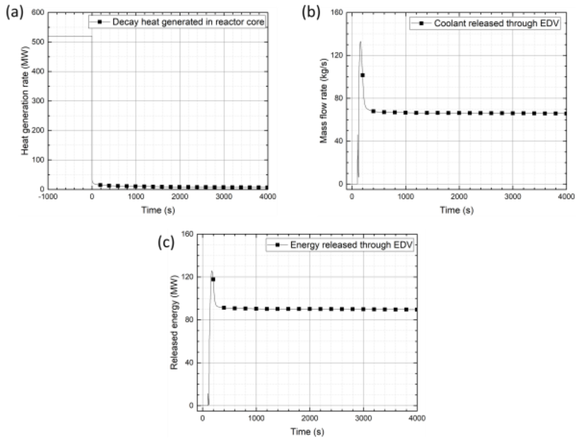

Fig 5. (a) decay heat generated within reactor core after EDV stuck open, (b) mass flow rate of coolant through EDV, and (c) energy released through EDV

3. Result and Discussion

Table II summarizes the major accident progression sequences in the event of an EDV stuck open accident. Coolant in the primary system is continuously discharged through the EDVs, resulting in a gradual decrease in primary pressure as shown Figure 6. Consequently, at 109.5 s, the reactor trip is predicted to occur due to low-pressure signal. Subsequently, as shown in Figure 7, there is a continuous decrease in the water level within the RPV, leading to the exposure of fuel rods to steam and the onset of oxidation reactions, accelerating the accident progression. For Core Exit Temperature (CET), it reaches 923 K, which is the point of severe accident entrance in domestic Pressurized Water Reactors (PWRs), at 4,795 s. Finally, RPV failure at 187,868 s is predicted so that the corium is released into the containment. The delayed RPV failure is attributed to the external wall cooling by the condensate within the containment.

Table II: Major sequence during accident progression of EDV stuck open scenario

Major sequence	Time [s]
Occurrence of EDV stuck open	0
Rx trip (low pressure signal)	109.5
Gap release start	3,952.3
Core support structure failure	5,042.1
RPV failure	187,868

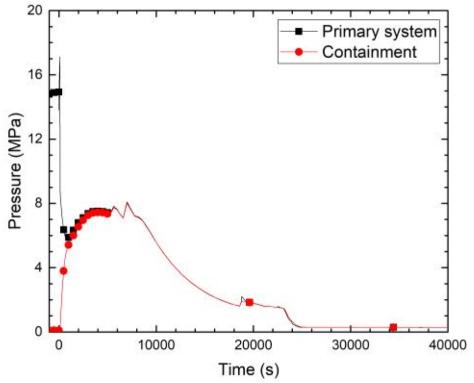

Fig 6. Primary pressure and pressure inside containment after EDV stuck open predicted by MELCOR

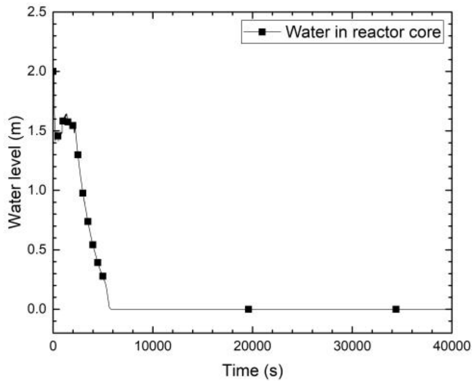

Fig 7. Water level inside reactor core after EDV stuck open predicted by MELCOR

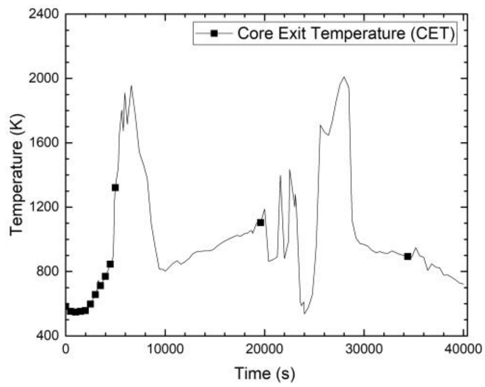

Fig 8. Core exit temperature after EDV stuck open predicted by MELCOR

Figures 9 to 11 show pressure behaviors, void fraction within the RPV, corium mass, and hydrogen generation mass predicted by CINEMA under same initiating event, respectively. In terms of pressure, the RCS and CV pressures were observed to converge around 24,000 s, maintaining a pressure of approximately 0.2 MPa, consistent with the trends predicted by the MELCOR code. However, some discrepancies were noted in the prediction of core degradation process. While MELCOR predicted core damage around 15,000 s, CINEMA indicated that core damage began around 33,000 s. This difference is attributed to variations in the heat transfer rate resulting from the cooling of the RPV outer wall by the condensate in the CV. Therefore, it is deemed necessary to modify and improve the MELCOR input model based on actual design information, specifically geometric structure of the RPV.

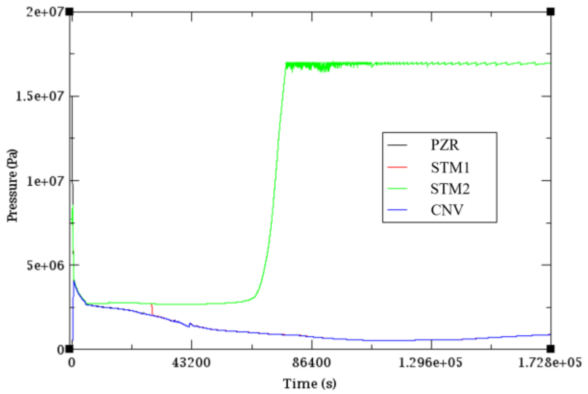

Fig 9. Pressure behavior predicted by CINEMA code

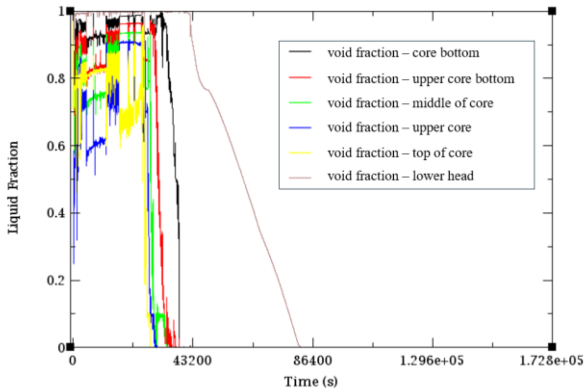

Fig 10. Void fraction predicted by CINEMA code

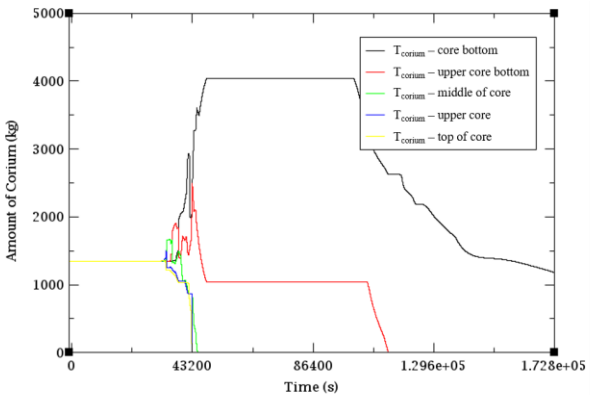

Fig 11. Amount of corium predicted by CINEMA code

Figures 12 and 13 show the mass of hydrogen generated within the reactor core predicted by MELCOR and CINEMA, respectively. Both codes predicted the production of over 200 kg of hydrogen. These trends are attributed to the smaller decay heat generation compared to large-scale NPPs, resulting in a slower decrease in water level. Consequently, it was inferred that the interaction between the fuel cladding and steam persists for a longer duration. Compared to the value predicted by CINEMA, MELCOR predicted the production of approximately 150 kg more hydrogen. The hydrogen was ultimately released into the containment space.

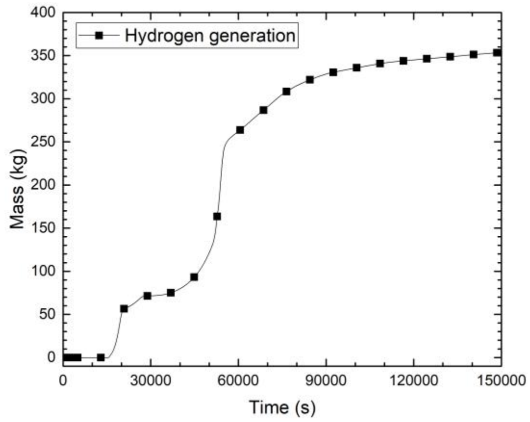

Fig 12. Mass of hydrogen generated inside reactor core predicted by MELCOR

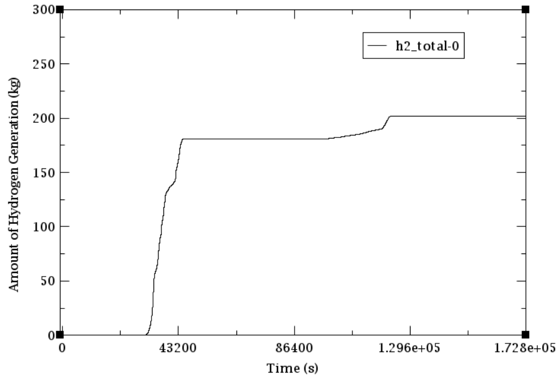

Fig 13. Mass of hydrogen generated inside reactor core predicted by CINEMA

4. Conclusions

In this study, preliminary analysis on accident progression in the i-SMR under development in Republic of Korea was conducted using the MELCOR code which is a regulatory code utilized by the U.S.NRC. To perform MELCOR calculations, an i-SMR MELCOR input model was developed. The initiating event assumed was the discharge of coolant from the primary system into the containment, with the scenario assuming the malfunction of passive safety systems. Key findings and future research directions are summarized as follows.

(1) The i-SMR was successfully modeled using the MELCOR 2.1 code, yielding reasonable results in steady-state calculation.
(2) Continuous release of coolant from the primary system into the containment led to a gradual decrease in the water level within the RPV. It resulted in the anticipation of fuel rod exposure to

steam, which accelerated the accident progression due to oxidation reaction.

(3) At approximately 15,000 s, core degradation started, and, RPV failure was predicted at 187,868 s, leading to the release of corium into the containment.

(4) MELCOR calculation predicted that over 350 kg of hydrogen would be generated. This is attributed to the characteristic of SMR accidents, where reactions between cladding materials and steam occur over a relatively prolonged period compared to large-scale NPPs.

(5) Improvements will be made to the MELCOR input model based on actual i-SMR design information to obtain more accurate analysis results. This task is expected to be one of the most crucial aspects of future research.

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