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Preliminary Single-code Analysis of Mass and Energy Release and Containment Vessel Response in i-SMR during Postulated LOCA and MSLB Using SPACE Code

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1. Introduction

In response to the growing electricity demand and global trend toward de-carbonization, Small Modular Reactors (SMRs) have emerged as a promising power solution. The Republic of Korea is currently developing the Innovative SMR (i-SMR), aiming to complete its standard design by 2025 and to obtain the standard design approval by 2028 [1]. The i-SMR employs an integral pressurizer water reactor design and features a steel containment vessel (CV) instead of a conventional concrete containment building.

The CV serves as a critical safety barrier to prevent the release of radioactive materials into the environment and is required to maintain structural integrity during design-basis accidents such as a Loss of Coolant Accident (LOCA) and a Main Steam Line Break (MSLB). The i-SMR is designed without large pipes, thereby fundamentally eliminating the possibility of a Large Break LOCA (LBLOCA). When the Passive Emergency Core Cooling System (PECCS) is actuated, a flow path is established from the reactor coolant system (RCS) to the CV. Accordingly, inadvertent operation of PECCS (IOPECCS) events are regarded as a type of LOCA. Hence, Small Break LOCA (SBLOCA) and IOPECCS events are identified as the representative LOCA scenarios for the i-SMR. The CV is designed to withstand a pressure of 5.0 MPa under LOCA and MSLB conditions [2]. To evaluate its structural integrity, integrated mass and energy (M/E) release and CV response analyses are essential.

This paper presents a preliminary study on M/E release as well as CV response during postulated LOCA and MSLB accidents in the i-SMR. The thermal-hydraulic modeling and calculations of the i-SMR's nuclear steam supply system (NSSS) and CV were performed using the Safety and Performance Analysis CodE for nuclear power plants (SPACE) [3]. In contrast to the conventional two-step approach typically applied in M/E release and containment response analyses for large scale reactor [4, 5], this study adopts a one-through, M/E release and CV response integrated analysis using the SPACE code, reflecting the design characteristics of the i-SMR's CV.

2. Analysis Methodology

To analyze the M/E release and CV response of the i-SMR, the NSSS and CV were modeled using thermal-hydraulic nodes in the SPACE code. The analysis employed a standard input deck jointly developed by participating organizations, including KEPCO Engineering & Construction Company, Inc., Korea Hydro & Nuclear Power Co., Ltd., KEPCO Nuclear Fuel Co., Ltd., and FNC Technology Co., Ltd., based on the i-SMR Design Product Level 2 (DPL-2). To maximize M/E release and CV response, this standard input deck was modified by incorporating conservative assumptions. The SPACE code applied in this study is an upgraded version of SPACE ver. 3.3, modified by the Korea Atomic Energy Research Institute specifically for the i-SMR safety analysis.

For the postulated LOCAs of the i-SMR, four representative cases were analyzed: a charging line break (CLB) and a letdown line break (LLB), classified as SBLOCAs, and an inadvertent opening of an Emergency Depressurization Valve (IOEDV) and an Emergency Recirculation Valve (IOERV), classified as IOPECCS events. A double-ended break with an area of 2,087.6 mm^2 was assumed for both the CLB and LLB. The areas of the EDV and ERV were assumed to be 2,827.4 mm^2 and 921.1 mm^2, respectively. Additionally, a postulated MSLB was analyzed by assuming a double-ended rupture of a single main steam line inside the CV, with a rupture area of 257,500 mm^2.

The key assumptions applied in the analyses are summarized in Table I. These assumptions were conservatively established to maximize the M/E release rate and to ensure conservative CV response results. The Henry-Fauske/Moody critical flow model [6, 7] with a discharge coefficient of 1.0 was applied to the breaks, EDVs, and ERVs in both LOCA and MSLB analyses. The turbine was assumed to trip at break initiation in both LOCA and MSLB scenarios so as to retain energy within the NSSS. Upon loss of alternating current (AC) power, the reactor coolant pumps (RCPs) and feedwater pumps were assumed to trip, reducing heat transfer from the primary to the secondary side. For LOCA scenarios, a simultaneous loss of AC power was assumed to minimize heat transfer from primary to secondary side and thereby maximize the M/E release rate. In contrast, for MSLB scenarios, AC power was assumed to remain available to enhance heat transfer from primary to secondary side, thereby maximizing the M/E release rate. Direct current (DC) power was assumed to be available in both LOCA and MSLB scenarios. The single failure of the passive auxiliary feedwater startup valve (PAFSV) was also considered. However, no significant impact on the results was identified due to the series installation of the PAFSVs. The initial conditions were determined using either nominal values with added uncertainty margins or conservative values.

Table I: Major assumptions for the M/E release and CV response analysis during LOCA and MSLB in the i-SMR

Parameters	Assumptions	
	LOCA	MSLB
Initiating event	CLB inside CV, LLB inside CV, IOEDV, IOERV	MSLB inside CV
Break size	Maximum	
Critical flow model	Henry-Fauske (Sub-cooled) Moody (Two phase)	
Core power	103% of full power (including instrument uncertainty)	
Decay heat	1979 ANS Standard + 20% uncertainty	
Turbine trip	At break	
AC power	Loss at break	Available
DC power	Available	
Single failure	PAFSV failure (no significant effect to results)	
Low riser level setpoint	Maximum in harsh environment	
SOPM[1)] setpoint	Nominal	

1) Spurious opening protection module

3. Analysis Results and Discussion

3.1 M/E Release Analysis during LOCA

When a LOCA occurs, reactor coolant is released into the CV, resulting in pressurization of the CV. The high containment pressure (HCP) signal initiates reactor trip, followed by the CV isolation and Passive Auxiliary Feedwater System (PAFS) actuation. As coolant continues to be released, the RCS water level decreases to the low riser level setpoint for PECCS actuation and the differential pressure between the reactor vessel (RV) and CV reaches SOPM setpoint. Then, the PECCS is actuated by opening the remaining EDVs and ERVs. Subsequently, the RCS is cooled to a safe shutdown state through continued actuation of both the PAFS and the PECCS.

Figures 1 and 2 present the total M/E release rates for the four cases of LOCA scenarios. If a CLB or a LLB occurs, reactor coolant is discharged through the breaks. Due to the lower elevation of the charging line compared to the letdown line, a higher M/E release rate is observed in the case of CLB. During an IOEDV, reactor coolant is discharged through the EDV as a two-phase mixture, since the EDVs are installed at the top of the pressurizer. In contrast, during an IOERV, the liquid coolant is discharged through the ERV, since it is connected to the RV downcomer, which is located below the normal water level of the RCS. As the accident progresses, the M/E release rate gradually decreases. However, when the PECCS is actuated, additional M/E is released due to the subsequent opening of the remaining EDVs and ERVs.

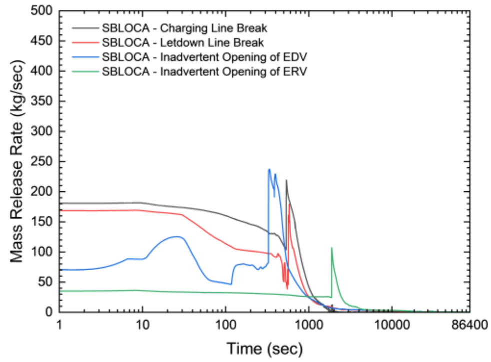

Fig. 1. Break and PECCS Mass Release Rate during LOCA

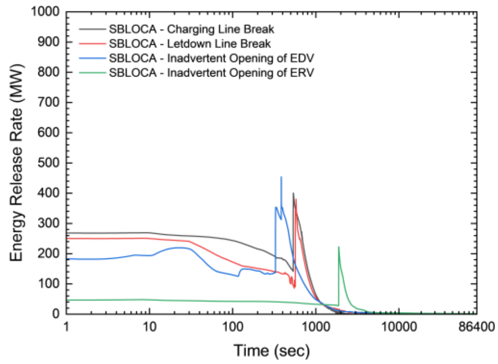

Fig. 2. Break and PECCS Energy Release Rate during LOCA

3.2 M/E Release Analysis during MSLB

Upon the occurrence of an MSLB accident, superheated steam is released into the CV, which consequently leads to an HCP reactor trip. Subsequently, the CV is isolated, and the cooling of the RCS is initiated by the PAFS actuation. Unlike the LOCA, the M/E release terminates within a short duration due to the limited inventory of the helical-coiled steam generator (HCSG). The RCS is then cooled down to a safe shutdown state solely by PAFS, without PECCS actuation.

Figures 3 and 4 show the M/E release rates during postulated MSLB accident. In the very early phase, a significant amount of superheated steam M/E is released through the break. Steam discharge occurs simultaneously from both the affected HCSG and the steam header. Break occurrence can lead to the entrainment of liquid and droplet due to depressurization of the affected HCSG. However, only a very small amount of such entrainment was observed. At 7 seconds after break occurrence, the main steam isolation valves (MSIVs) are closed as a result of the HCP reactor trip, which blocks steam release from the steam header and leads to a rapid decrease in the steam M/E release rate. The M/E release completely terminates at 11.2 seconds, when the inventory of the affected HCSG is depleted.

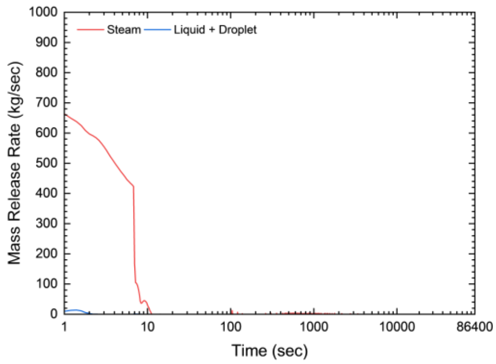

Fig. 3. Break Mass Release Rate during MSLB

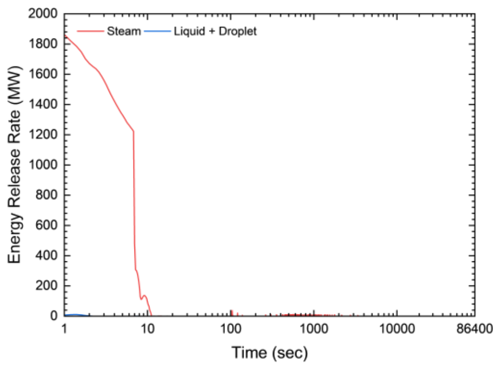

Fig. 4. Break Energy Release Rate during MSLB

3.3 CV Response Analysis during LOCA and MSLB

In the event of a LOCA or MSLB accident, the loss of vacuum in the CV occurs due to the release of coolant through the break. At this time, the Passive Containment Cooling System (PCCS) installed at the upper part of CV condensates the steam inside the CV. Additionally, condensation may occur on passive heat sinks such as the CV walls. The condensed water accumulates at the bottom of the CV, leading to an increase in the CV water level. If the PECCS is actuated, the water collected at the bottom of the CV is injected into the core through the ERVs, thereby contributing to core cooling.

Figure 5 shows the CV pressure behavior of the i-SMR during the postulated LOCAs and MSLB, respectively. In the very early phase, MSLB shows higher CV pressure than those of LOCAs due to the high M/E release rate of superheated steam. However, the M/E release in MSLB terminates early, whereas in LOCA, the continued M/E release leads to continued increase in CV pressure. In the case of LOCA scenarios, the PECCS actuation causes an increase in the rate of CV pressure rise due to additional M/E release.

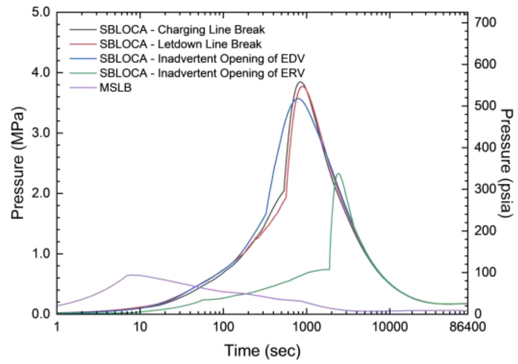

Fig. 5. CV Pressure during LOCA and MSLB

As the accident progresses into the later phase, the M/E release rate gradually decreases, and the depressurization and cooling by the PCCS operation and passive heat sinks become dominant. Consequently, the CV pressure reaches a peak and then exhibit a decreasing trend. The highest CV peak pressure was observed in the postulated CLB accident, reaching 3.85 MPa (558.40 psia).

4. Conclusion

In this study, the M/E release and CV response of i-SMR during postulated LOCA and MSLB accidents were investigated using the SPACE code. The postulated LOCAs are categorized into four cases: CLB, LLB, IOEDV, and IOERV. Upon the occurrence of a LOCA, the CV pressure initially rises due to the discharge of reactor coolant through the break, and then increase more rapidly following the actuation of PECCS, eventually reaching a peak. In case of the MSLB accident, the blowdown of the steam terminated within a short duration due to the limited inventory of the HCSG, resulting in a lower CV peak pressure compared to the LOCA.

The highest CV peak pressure was observed in the CLB, reaching 3.85 MPa. The acceptance criteria for containment functional design require a margin of at least 10 % above the design pressure. The most limiting CV peak pressure of 3.85 MPa in this study provides a margin of 23 %, thus satisfying the acceptance criteria. In other words, even though the current design meets the regulatory requirements, installing safety-related check valves on the charging line near the reactor vessel could further reduce the likelihood of a CLB and enhance the safety of the i-SMR.

Future research will focus on advancing the methodologies for M/E release and CV response analyses to support the standard design approval of the i-SMR. Current challenges, such as the reliability of non-safety grade power and the development of two-step analytical methodology, will be addressed as part of this effort.

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