Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 9-10, 2024

A Mass and Energy Release Analysis of Postulated Main Steam Line Break Accidents on APR1000 Using KIMERA Methodology

Jisu Kim*, Sung Yong Kim, Seong Ho Jee, Chan Eok Park, Min Shin Jung, Eun Ju Lee, Seok Jeong Park
Safety Analysis Department, KEPCO Engineering & Construction Company Inc.,
111, Daedeok-daero 989 Beon-Gil, Yuseong-gu, Daejeon 34057, Republic of Korea
**Corresponding author: jisukim@kepco-enc.com*

Keywords : Mass and energy release, MSLB, REALP5-ME, CONTEMPT4/MOD5, SPACE-ME

1. Introduction

The mass and energy (M/E) of coolant release analysis of postulated loss of coolant (LOCA) and main steam line break (MSLB) accidents is absolutely required for the containment response analysis of pressurized water reactor (PWR). Korea Electric Power Corporation Engineering and Construction Company Inc. (KEPCO E&C) developed KEPCO E&C Improved Mass and Energy Release Analysis (KIMERA) [1,2]. Recently, KEPCO E&C has been developing a new methodology for the M/E release analysis, known as SPACE-ME methodology. In the recent studies, preliminary research has been conducted on the M/E release analysis for the Advanced Power Reactor 1400 MWe (APR1400) using the SPACE-ME methodology [3,4,5].

The Advanced Power Reactor 1000 MWe (APR1000) is a Gen III+ PWR under development in Republic of Korea for export purposes. It has been designed with enhanced safety and performance based on the proven technology and operational experience of Optimized Power Reactor 1000 MWe (OPR1000) and APR1400. The standard design of APR1000 was certified as compliant with the European Utility Requirements (EUR) Association in March 2023. Furthermore, the application of APR1000 standard design approval will be submitted to Korea Institute of Nuclear Safety (KINS) in the near future.

This paper presents the M/E release analysis of the postulated MSLB accident in APR1000 using KIMERA methodology as basic data prior to the application of SPACE-ME methodology to APR1000. The M/E release data were calculated by RELAP5-ME code, with coupling of RELAP5K and CONTEMPT4/MOD5 codes [1,2]. The reactor containment response regarding pressure and temperature (P/T) behavior was analyzed with stand-alone CONTEMPT4/MOD5 code.

2. Methodology

APR1000 nuclear steam supply system (NSSS) modeling of RELAP5-ME code was used. It is assumed that the rupture of steam line occurs inside containment. Given the postulated accident analysis, conservative assumptions and initial conditions of APR1000 NSSS aimed at maximizing M/E released from the break were applied to the analysis. Table I describes the major assumptions employed in this study, based on those of OPR1000 and APR1400 [1,2]. Table II describes the initial conditions of APR1000 NSSS used in this study, based on the limiting conditions for operation of APR1000.

Table I: Major assumptions of the M/E release analysis for the postulated MSLB accidents of APR1000

Parameters	Assumptions
Evaluation time	30 min. from the accident initiation
Turbine trip	At the accident initiation
Offsite power	Available
Feedwater flow to steam generator (SG)	Maximum total flow only to broken side
Feedwater enthalpy	Maximum enthalpy

Table II. Initial conditions of the M/E release analysis for the postulated MSLB accidents of APR1000

Parameter	Value
Core power, %FP (% of full power, 2815 MWt)	102, 75, 50, 20, and 0
Pressurizer (PZR) pressure, MPa (psia)	16.03 (2325)
Core inlet temperature, K (°F)	573.15 (572)
RCS flow rate, %	95
PZR water level, %	60 (for 102, 75, and 50 %FP), 50 (for 20 %FP), 40 (for 0 %FP)
SG water level, %NR (% of narrow range)	95
Break type	Slot break
Break size, Cd (Discharge Coefficient)	1.0, 0.5, 0.4, 0.3, 0.2, and 0.1

The RELAP5-ME code simulated the thermal-hydraulic behavior of APR1000 NSSS to analyze the M/E release during the postulated MSLB accidents. The M/E release analysis was performed for the various initial core power conditions, such as 102%, 75%, 50%, 20%, and 0% of full power. The M/E release rates for the break size spectrum for each core power, such as discharge coefficient of 1.0, 0.5, 0.4, 0.3, 0.2, and 0.1 were calculated. The stand-alone CONTEMPT4/MOD5 code predicted the containment P/T with the obtained M/E release data.

3. Results and Discussion

3.1 Mass and Energy Release Analysis

After the accident initiation, a large amount of steam M/E is released through the rupture of steam line. It results in excessive heat removal of reactor coolant, which in turn increases the core power by negative moderator temperature coefficient, and pressurization of the containment atmosphere. The reactor shuts down by the high-containment pressure trip (HCPT) signal. The main steam line isolation valves (MSIVs) are closed by HCPT signal, isolating the intact SG. The feedwater supply is also cut off by the main feedwater isolation valves (MFIVs) closure. The M/E release continues until the affected SG inventory is depleted.

Figure 1 and 2 respectively represent the integrated amount of M/E release corresponding to the break size with the highest containment peak pressure of each core power. The M/E release continues after accident initiation until the affected SG inventory is exhausted. After that, there is no further increase in the cumulative M/E release due to the absence of feedwater supply. The largest amount of M/E release occurred for the postulated MSLB accident of initial core power of 75 %FP and break size corresponding to discharge coefficient of 0.2.

3.2 Containment Pressure and Temperature Analysis

The containment P/T increases sharply after the accident initiation. If the containment pressure reaches the setpoint of containment spray actuation (CSA) signal, CSA begins and cools down the containment P/T after a certain response time. When the cooling by the containment spray and passive heat sink becomes greater than pressurization and heating by the M/E release, the containment P/T starts to decrease. Interestingly, the containment pressure may continue to rise even after the CSA is in operation, while the containment temperature reaches its highest peak at the onset of containment spray.

Figure 3 and 4 depict the containment P/T behavior corresponding to the break size with the highest containment peak P/T of each core power, respectively. The maximum containment peak pressure is observed at 411,616 Pa (59.70 psia) from the accident of initial core power of 75 %FP and break size with discharge coefficient of 0.2. The maximum peak temperature of the containment was found to be at 409.15 K (276.78 ºF) from the accident of initial core power of 0 %FP and break size with discharge coefficient of 0.2.

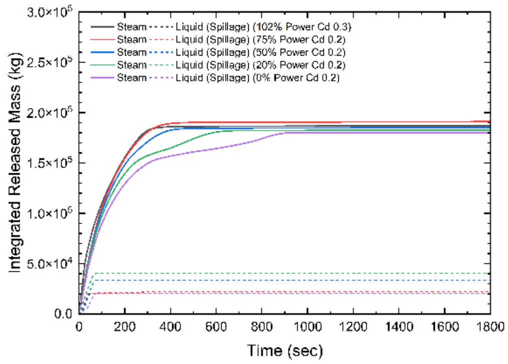

Fig. 1 Integrated released mass during the postulated MSLB accidents on APR1000 by break size with the highest containment pressure for each core power

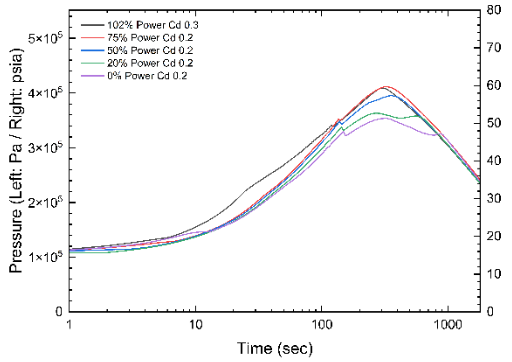

Fig. 3 Containment pressure during the postulated MSLB accidents on APR1000 by break size with the highest containment pressure for each core power

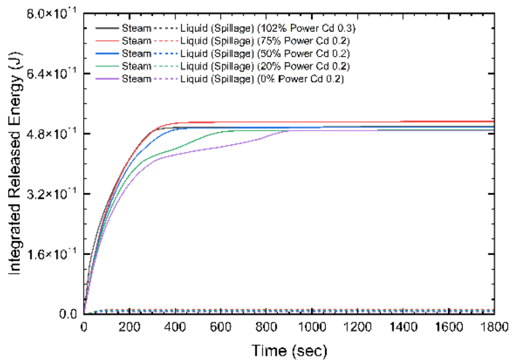

Fig. 2 Integrated released energy during the postulated MSLB accidents on APR1000 by break size with the highest containment pressure for each core power

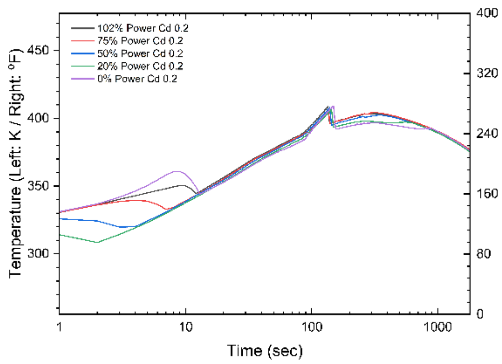

Fig. 4 Containment temperature during the postulated MSLB accidents on APR1000 by break size with the highest containment temperature for each core power

3.3 Comparison APR1000 with OPR1000

Figure 5 and 6 compare the M/E release rates of the postulated MSLB accident of 75 %FP and break size with discharge coefficient of 0.2 for APR1000 and OPR1000. The trends are similar for APR1000 and OPR1000, which share the same full power level, until the affected SG inventory is depleted. After depletion, the M/E release rates of APR1000 drop to zero due to the lack of auxiliary feedwater supply. It is because the auxiliary feedwater system of APR1000 is a passive system. In contrast, OPR1000 equipped with active auxiliary feedwater systems maintains a continuous auxiliary feedwater supply, resulting in the M/E release that converge to positive values.

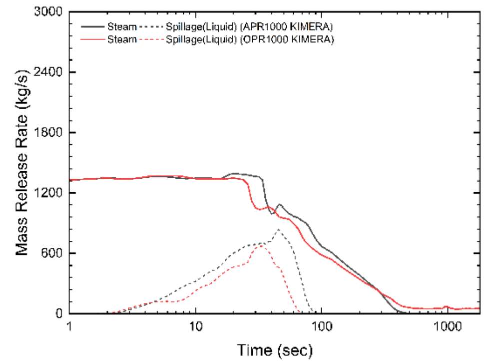

Fig. 5 Mass release rate by reactor type during the postulated MSLB accident of initial core power of 75 %FP and break size with discharge coefficient of 0.2

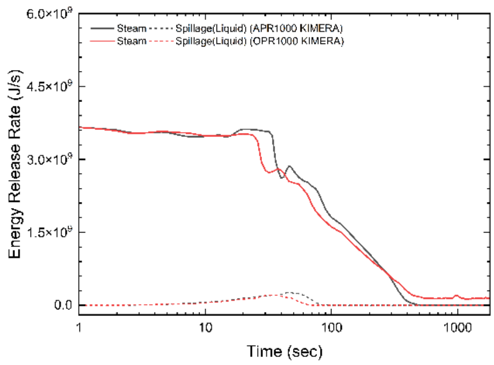

Fig. 6 Energy release rate by reactor type during the postulated MSLB accident of initial core power of 75 %FP and break size with discharge coefficient of 0.2

Table IV summarizes the highest containment peak P/T included in the final safety analysis reports (FSAR) of Shin-Kori Units 1&2 (SKN 1&2), which employ OPR1000. This study excluded the MSIV failure scenario, as the presence of two MSIVs in series on APR1000's main steam line. The analysis results revealed that the maximum containment peak P/T for APR1000 is lower compared to those of OPR1000. This is because the APR1000 containment is more than 10% larger in volume than the OPR1000, and the containment spray flow of the APR1000 has been increased, even though the M/E release rate of the APR1000 is higher than that of the OPR1000, as shown in Figure 5 and 6.

Table IV: Summary of the highest containment peak P/T, along with initial core power and break size, in the postulated MSLB accidents on APR1000 and OPR1000.

Comparison of methodology		MSIV failure		LCC	
		Pressure (psia)	Temp. (ºF)	Pressure (psia)	Temp. (ºF)
APR1000 KIMERA (This Study)	Peak	Not analyzed		**59.7 at 330 sec**	**276.8 at 147 sec**
	Power /Size			75 %FP Cd 0.2	0 %FP Cd 0.2
OPR1000 (SKN 1&2 FSAR, Rev.1)	Peak	63.24 at 332 sec	325.5 at 118 sec	**64.14 at 498 sec**	**335.1 at 125 sec**
	Power /Size	102 %FP, Double-Ended (Cd 1.0)	75 %FP, Double-Ended (Cd 1.0)	75 %FP, Double-Ended (Cd 1.0)	102 %FP, Double-Ended (Cd 1.0)

4. Conclusion

In this paper, the M/E release analysis for the postulated MSLB accidents on APR1000 was performed using KIMERA methodology. The containment P/T behavior was investigated sequentially based on the calculated M/E release data. The maximum containment P/T of APR1000 were determined to be 411,616 Pa (59.70 psia) and 409.15 K (276.78 ºF), which are less than those of OPR1000. This study will serve as basic data for the M/E release analysis on APR1000 using SPACE-ME methodology. Moreover, further studies are required for the application of APR1000 standard design approval.

REFERENCES

[1] Korea Electric Power Corporation Engineering and Construction Company Inc. "KOPEC Improved Mass and Energy Release Analysis Methodology," KOPEC/NED/TR/06-005, Rev. 0, 2007.
[2] Korea Electric Power Corporation Engineering and Construction Company Inc. "Applicability Assessment of KIMERA to APR1400 Nuclear Power Plants," KEPCO-E&C/TR/12-008, Rev. 0, 2012
[3] J. Kim, S. H. Jee, S. Y. Kim, M. S. Jung, E. J. Lee, and S. J. Park. "Preliminary Mass and Energy Release Analysis for Postulated MSLB Accidents on APR1400 Using SPACE-ME Methodology," Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, 2023.
[4] S. Y. Kim, S. H. Jee, M. S. Jung, E. J. Lee, and S. J. Park. "Preliminary SBLOCA M/E Release Analysis using SPACE-ME Methodology," Korean Nuclear Society Autumn Meeting, Gyeongju, Korea, 2023.
[5] S. H. Jee, S. Y. Kim, J. W. Cho, E. J. Lee, and S. J. Park. "Preliminary LOCA M/E Release Analysis using SPACE-ME Methodology," Korean Nuclear Society Autumn Meeting, Changwon, Korea, 2022.

Acknowledgment

This work was supported by Korea Institute of Energy Technology Evaluation and Planning (KETEP) grant funded by the Korea government (MOTIE) (20217810100020, Development of Unique Thermal Hydraulic Analysis System of Containment Building Based on SPACE for APR Nuclear Power Plant).