Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 17-18, 2018

Sensitivity Analysis for Containment Leakage Estimation under Severe Accident Conditions

Youngsuk Bang [a*], Juyub Kim [a], Seungwon Seo [b], Kuyoung Chung [b] and Kyuntae Kim [b]
[a] *FNC Technology, Co., Ltd., Giheung-gu, Yongin-si, Gyeonggi-do, 16954*
[b] *Korea Institute of Nuclear Safety, Yuseong-gu, Daejeon 34142*
*Corresponding author: ysbang00@fnctech.com

1. Introduction

In order to protect the public and the environment from an accident in nuclear power plant, the predictive approach may be the most effective way, i.e. estimate the radiological consequences, prepare the emergency response plan and take the protective actions as planned. For that, numerical simulation would be the essential tool in estimating source terms, particulates/gases dispersion, and external/internal dose regarding the complexity of related phenomena. Source term has been considered as the most influential factor because depending on magnitude, composition and timing of radioactive material releases, the results of the subsequent calculations, i.e., dispersion and dose calculations would be significantly varied.

Due to a wide spectrum of accident scenarios, various physical phenomena involved and inherent uncertainties in physical models, estimating the source term accurately is very challenging. Therefore, it is more important to understand the general relationships between plant conditions and source term characteristics and find the tendency in radioactive material releases. The emergency response plan could be established in support of the information with conservatism.

In this study, the source term due to containment leakage has been investigated. Conventionally, the radiological consequence analysis for containment leakage has been conducted with the assumption of design leakage rate, i.e., 0.1 volume percent per day. However, in case of high containment pressure, the containment leakage rate would be higher than design leakage rate, which means the larger source terms and the worsen radiological consequences [1]. Firstly, 1400MWe grade reactor has been modeled by using MELCOR code [2]. In particular, the containment leakage has been modeled considering the yield of liner plate and rebar of containment. Secondary, the radionuclide behaviors in the reactor vessel and the containment and the source term characteristics have been estimated and compared for different accident scenarios. Different initiating event and various combinations of safety system operations have been considered. Thirdly, the sensitivity analysis on the core initial inventory has been conducted by using a MELCOR sensitivity coefficient. Finally, the source characteristics have been investigated with respect to parameters representing plant conditions.

2. MELCOR Model

MELCOR model for 1400MWe grade reactor has been developed for numerical simulations based on the information in Reference 3. If necessary information was not found in publically available documents, it was assumed with reasonable justifications.

For containment leakage modeling, the simplified analysis method for concrete containment has been adopted [4]. It is assumed that the containment leakage is about 1% of the containment mass per day when the liner plate yields. This increases to 13% of containment mass per day when rebar yield and 62% when the containment global strains are 2%. The yield pressures for liner plate and rebar have been calculated and in order to consider the uncertainty, the pressure values reduced to 85% have been used in MELCOR model. It has been assumed that the leakage would occur at the equipment hatch, which is conservative because it would be a direct path to environment.

It was expected that according to the initiating event, the accident progression and the radioactive materials distribution would be differentiated. For example, in Loss of Coolant Accident, the pressure of reactor coolant system (RCS) would be low during the core degradation and a large portion of the generated radioactive gases and particulates would be released to containment atmosphere while in Station Black-Out, a large portion of the generated particulates would be deposited in RCS due to natural circulation. In order to capture those differences, following initiating events have been considered:

- Large Break Loss of Coolant Accident (LLOCA): Cold Leg Guillotine Break
- Medium Break Loss of Coolant Accident (MLOCA): Cold Leg 6 inch Break
- Small Break Loss of Coolant Accident (SLOCA): Cold Leg 2 inch Break
- Station Black Out (SBO)

In addition, the operations of systems which are influential on the plant conditions and the accident progression have been considered:

- Cavity Flooding System (CFS)
- Emergency Containment Spray Backup System (ECSBS)

3. Numerical Experiment

The accident scenarios with different combination of safety system operation have been simulated by using MELCOR code. In this chapter, the major results are briefly presented.

3.1 Operation of Cavity Flooding System

In case that the cavity flooding system (CFS) is not operated, there would be no water in the cavity; thus, after reactor vessel fails, the steam vaporization at cavity would be much less than the one in case without CFS operation. The containment would be pressurized mainly due to the gases generated by molten corium-concrete interaction (MCCI). As can be seen in Fig. 1, the rate of containment pressurization would be appreciably increased if CFS operated. The 2% strain pressure was reached within 2 days, which results in increased leakage area; thus, the containment leakage would be increased significantly as can be seen in Fig. 2.

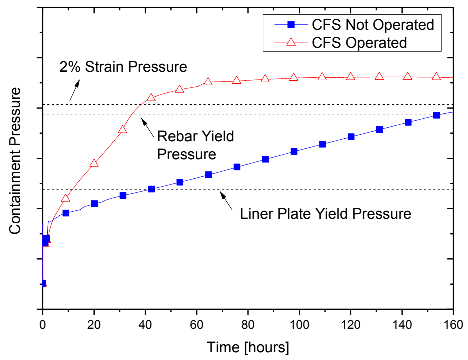

Fig. 1 Containment Pressure Comparison (LLOCA with ECSBS Operation)

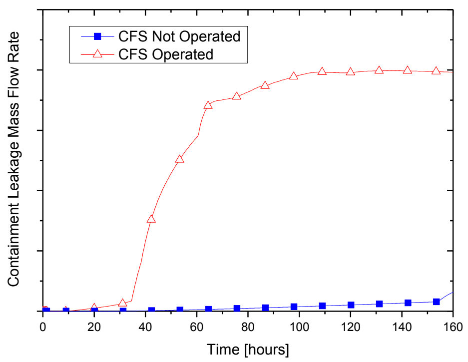

Fig. 2 Containment Leakage Rate Comparison (LLOCA without ECSBS Operation)

3.2 Operation of Emergency Containment Spray Backup System

The containment spray would condense the steam in the containment atmosphere and decrease the containment pressure instantly as can be seen in Fig. 3. The containment leakage mainly depends on the pressure difference between containment and environment. Therefore, the containment leakage rate would be decreased significantly as the containment spray initiated. In addition, the containment spray would remove the particulates and water soluble gases in the containment atmosphere effectively. Therefore, the release of those via containment leakage would be significantly decreased as can be seen in Fig. 4.

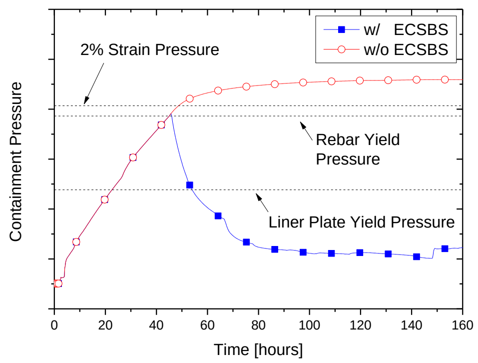

Fig. 3 Containment Pressure Comparison (SBO with CFS Operation)

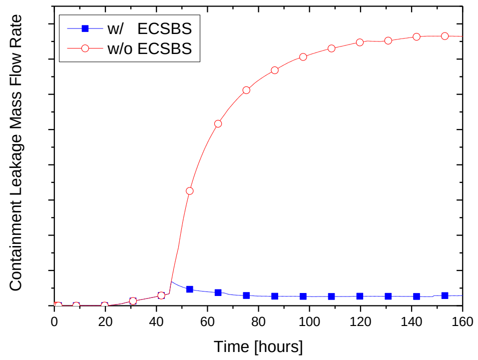

Fig. 4 Containment Leakage Rate Comparison (SBO with CFS Operation)

3.3 Initial Core Inventory Sensitivity Analysis

Initial core inventory sensitivity analysis has been conducted by using MELCOR sensitivity coefficient (SC) 3212. MELCOR has default core inventory data sets and according to specifying SC 3212, core inventory composition can be selected, i.e., zero corresponding to Beginning of Cycle (BOC) and one to End of Cycle (EOC).

Basically, the fission products would be accumulated in the core as the fuel depleted. Therefore, the larger release mass of radioactive materials can be expected. However, in this study, the core degradation rate has been found to be the important factor on release mass. In the case with EOC inventory, the core was degraded while the water by safety injection tank was being injected and relocated into the flooded cavity. Therefore, the particulates and the water soluble gases were scrubbed and, their release masses were significantly diminished. On the other hands, in case with BOC or MOC inventory, the core degradation was a little slower than EOC case. After the safety injection tank was depleted, the core was exposed until relocated into the flooded cavity and the generated particulates and the gases were released to containment atmosphere without scrubbing.

As can be seen, Fig. 5, the released mass of the non-condensable and insoluble gas was increased as the core depleted. However, in Fig. 6, the released mass of soluble particulate was significantly decreased due to scrubbing effects.

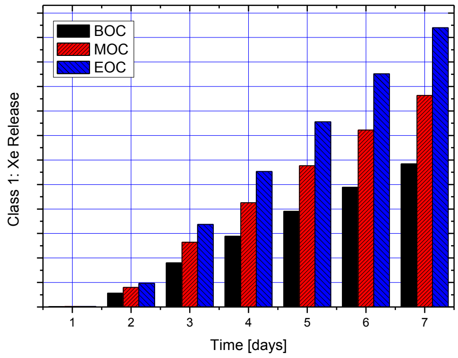

Fig. 5 Comparison of Non-condensable and Insoluble Gas released Mass due to Containment Leakage (SBO with CFS Operation)

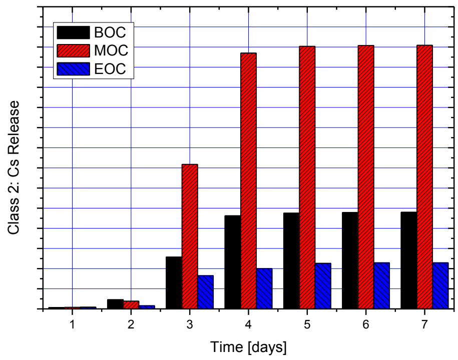

Fig. 6 Comparison of Soluble Particulates released Mass due to Containment Leakage (SBO with CFS Operation)

3.4 Comparison to Conventional Approach

Conventionally, the containment leakage has been modeled as a small hole, of which area is calculated based on nominal design leakage rate, i.e., 0.1 volume percent per day at the design internal pressure and remains unchanged. Obviously, the leakage rate with the consideration of yields of liner plate and rebar would be increased appreciably because of the enlarged leakage area as can be seen in Fig. xx and Fig. xx. However, it is important to note that the conventional assumption would not always produce the smaller source term. Due to the smaller leakage, the containment pressure would be increased rapidly and it may end up with the catastrophic containment failure due to over-pressurization as in SLOCA_CFS case in Fig. xx.

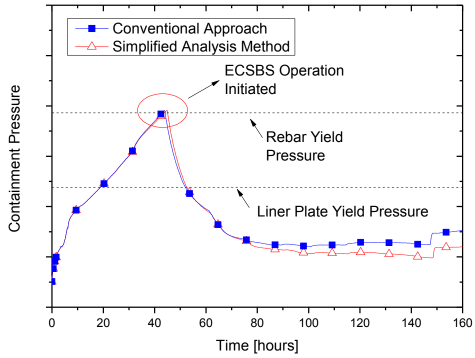

Fig. 7 Comparison of Containment Pressure (SLOCA with CFS and ECSBS Operation)

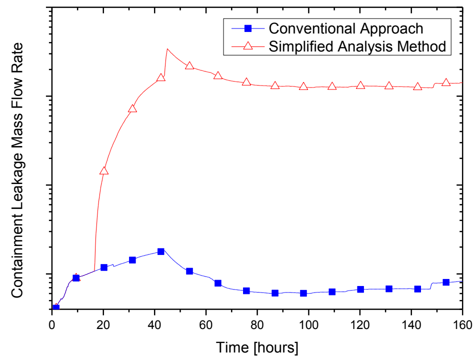

Fig. 8 Comparison of Containment Leakage Rate (SLOCA with CFS and ECSBS Operation)

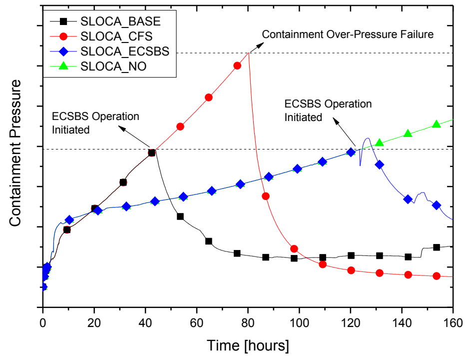

Fig. 9 Comparison of Containment Pressure (SLOCA with Conventional Assumption for Containment Leakage)

4. Conclusions

Severe Accident Analyses have been performed by using MELCOR code. Several initiating events with different safety system operations have been assumed and the simulation results have been compared. It has been found that the containment pressure is the most influential factor on leakage rate (i.e., released mass). Considering yields and failure of containment inner structure, the released mass would be significantly increased.

It is important to note that the leakage characteristics would be varied according to plant condition and safety system operations. For example, if the containment pressure is high when the reactor vessel fails, the particulate generated would be released via containment leakage without less experiencing natural removal. Flooding the cavity would reduce the soluble radioactive materials by pool scrubbing. However, it would accelerate the containment pressurization; thus, the leakage rate of non-condensable and in-soluble materials would be increased. Operation of depressurization system (e.g., ECSBS) would be critical to minimize the environmental effects. Therefore, it would be critical to identify the plant status accurately for estimating the source term and implement the emergency plan.

REFERENCES

[1] State-of-the-Art Reactor Consequence Analyses (SOARCA) Report, NUREG-1935, U.S.NRC, 2012.
[2] MELCOR Computer Code Manuals, SAND2015-6691R, U.S.NRC, 2015.
[3] APR1400 Design Control Document, APR1400-K-X-FS-14002-NP, Rev. 0, 2014.
[4] State-of-the-Art Reactor Consequence Analyses Project, Volume 2: Surry Integrated Analysis, NUREG/CR-7110, Vol. 2, U.S.NRC, 2012.