Transactions of the Korean Nuclear Society Virtual Spring Meeting
July 9-10, 2020

Assessment of Source Terms for ISLOCA Using MELCOR

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1. Introduction

Interfacing system loss of coolant accident (ISLOCA) is an accident in which the breakdown occurs at the low-pressure boundary of the reactor coolant system (RCS) connected to the outside of the containment. Under the ISLOCA condition, the fission product is released to the environment directly without going through the containment, causing a large amount of source term to be released early [1].

This characteristic of the ISLOCA has a big influence when assessing the site risk. In the past, safety analysis was limited to a single unit. Recently, however, evaluating the safety of the entire site such as multi-unit risk and site risk has become an issue. As a result of a recent study about evaluating site risk of the reference site, the risk tended to be overrated because of the overestimation of the source term for ISLOCA. The existing analysis of ISLOCA did not model the auxiliary building, and used a conservative assumption that all fission products leaving the auxiliary building are released into the environment. Therefore, realistic source term evaluation without the conservative assumption is needed [2].

In this study, the ISLOCA piping and the auxiliary building were modeled to realistically evaluate the source term. Also, the effect of pool scrubbing phenomenon and filtration function, which are the major retention mechanisms under the ISLOCA, was analyzed. Lastly, the effectiveness of the mitigation strategy using Power Operated Relief Valve (PORV) was also evaluated. If PORV is opened, the fission product could be induced to escape to the containment, and the amount of escape to the auxiliary building be relatively reduced.

2. Methods

MELCOR code version 2.2 which is a severe accident analysis code was used to analyze the behavior of fission products in the auxiliary building under the ISLOCA condition in this study. Also, a Westinghouse 2-loop pressurized water reactor was selected as a reference plant.

2.1 Reference plant Modelling

The reference plant has two loops, and there are one hot leg and one cold leg for each loop. The plant's RCS coolant inventory is 170 m^3 and its thermal output is 1,876 MW. There are two Accumulators, each with 35.4 m^3 capacity. There are two high-pressure safety injection (HPSI) pumps, and the refueling water storage tank (RWST) which is a safety injection water source has a capacity of 1,170.0 m^3 [3]. Pool scrubbing is affected by the submerged depth, and most of the inventories of RCS, accumulators, and RWST escapes to the auxiliary building during ISLOCA to submerge the break-part.

2.2 ISLOCA Piping Selection

Four pipes could have a possibility of ISLOCA in the reference plant [4].

- Piping connected to cold leg safety injection inlet
- Piping connected to RPV safety injection inlet
- Piping connected to hot leg recirculation inlet
- An inlet piping for residual heat removal system

Among them, the inlet piping of residual heat removal (RHR) system was selected as the ISLOCA break location according to the previous study [5]. Also, this piping is connected to an RHR pump located on the lowest floor of the auxiliary building. Therefore, the break location could be flooded because of the inventories of the coolant that has passed into the auxiliary building, and the effect of the pool scrubbing could be seen. This ISLOCA piping connects the hot leg to the RHR pump. There are two motor-driven valves and one pressure-relief valve in the piping. In this study, it was assumed that both motor-driven valves were ruptured and the pressure-relief valve failed to open. Also, it is assumed that a 4-in size break occurred at the place where the piping and the RHR pump meet.

2.3 ISLOCA Piping and Auxiliary Building Modelling

Figure 1 shows the nodalization of the ISLOCA piping and auxiliary building. The ISLOCA piping was modeled as one horizontal pipe (CV901) and one vertical pipe (CV902). This piping has actually a complicated structure, but for convenience of the calculation, it was modeled two control volumes. The auxiliary building was divided with six control volumes (CV921-926) and with the RHR pump room (CV911). Flow path from the vertical pipe (CV902) to the RHR pump room (CV911) was modeled as a 4in-break (FL911). The flow path from the RHR pump room (CV911) to the auxiliary building (CV921) had a waterproof door (FL912) and a drain pipe (FL941). The watertight door was assumed to open when the pressure difference between the RHR pump room and the auxiliary building exceeds 6.894 kPa. It is assumed that there was a flow path (FL926) of size 2.0 m^2 from the auxiliary building to the environment. Heat structures of floor, wall, and ceiling were modeled in both the RHR pump room and the auxiliary building.

A ventilation system was also modeled in the auxiliary building. There is a supply system that blows air into the auxiliary building and an exhaust system that draws air into the environment. A filter is present at the end of the exhaust system to prevent fission products from releasing into the environment.

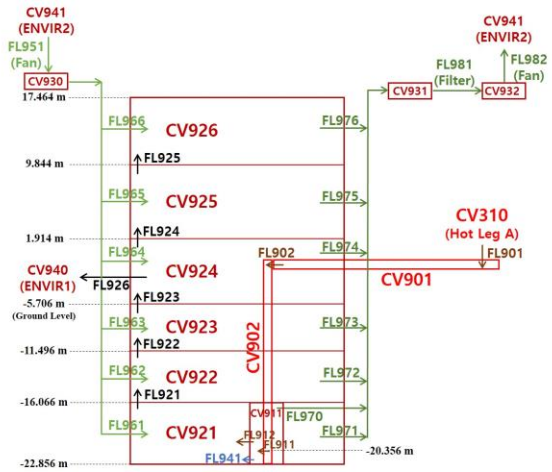

Fig. 1. The nodalization of the ISLOCA piping and the auxiliary building for the reference plant.

2.4 ISLOCA Scenario Selection

Table I: Sensitivity runs for ISLOCA

	B-Case	F-Case	V-Case	P-Case	A-Case
ISLOCA Occurs	0 sec				
Break Size	8.11E-3 m^2 (4-in break)				
HPSI Injection	Succeed				
HPSI Recirculation	Fail				
Break-part Flooded	X	O	X	X	O
Ventilation System	X	X	O	X	O
PORV Open	X	X	X	O	O

In this study, 5 cases were analyzed. Table I shows the feature of each case. B-Case assumed the following accident scenarios. The ISLOCA occurred at 0 seconds, HPSI injection succeeded, but recirculation failed. The ISLOCA pipe break part was not flooded, and the ventilation system of the auxiliary building failed to operate.

The fission products could be highly retained by the pool scrubbing due to flooding and filtration of the ventilation system [6]. Therefore, in this study, the fission product behavior was analyzed for F-Case with pool scrubbing, and V-Case with filtration by the ventilation system.

Besides, it is also possible to mitigate the amount of fission product released into the environment by opening PORVs during ISLOCA [1]. Therefore, P-Case opened a PORV 5 minutes after the accident to confirm the effectiveness of the mitigation strategy. Lastly, A-Case with all retention mechanism was also analyzed.

3. Result

3.1 Accident progression and source term analysis

Table II: The accident progression for B-Case

Event	Time (sec)	Time (hr)
ISLOCA Starts	0.0	0.0
Reactor Trip	17.6	0.0
HPSI Injection	17.6	0.0
HPSI End (RWST Exhaust)	12,931.3	3.6
SAMG Entry (CET > 650 °C)	16,738.4	4.6
Gap Release	17,102.5	4.8
FP Release to Environment	17,103.2	4.8
RPV Failure	24,973.4	6.9

Table II summarizes the timings of the major events during the accident. After ISLOCA occurs, the RCS pressure decreases rapidly. Due to the RCS low-pressure signal, the reactor trips at 17.6 seconds and HPSI begins. However, after HPSI stops at 3.6 hours, the water level drops gradually and the core is uncovered at 4.2 hours. As the core is exposed, the core temperature rises, and the cladding is damaged and a gap release begins at 4.8 hours. At 6.0 hours, most of the fission products are released out of fuel.

The pressures of the RHR pump room and the auxiliary building rise immediately and reach 111.6 kPa(a). Then, the pressure decreases to atmospheric pressure after 200 seconds. The water level in the RHR pump room rises sharply at the beginning of the accident and rises to 2.36 m from the bottom of the lowest floor, but remains at 1.46 m from the bottom after HPSI injection stops.

When the gap release begins, most of the cesium escapes to the auxiliary building through the ISLOCA piping. In Figure 2, cesium is deposited up to 3.4 % at the ISLOCA piping early in the accident. However, it resuspends over time and leaves only 0.1 % 24 hours after the accident. About 5.1 % of cesium is deposited in the RCS. In the containment, about 0.8 % of cesium is retained at 24 hours. About 80.0 % of cesium is retained in the auxiliary building. In Figure 3, cesium, which has escaped into the auxiliary building after gap release, is mostly present in the atmosphere of the auxiliary building. However, it agglomerates and settles with time. It dissolves in the pool on the bottom or is deposited on HS. Also, it is released to the environment with 10.4 % of cesium at 24 hours.

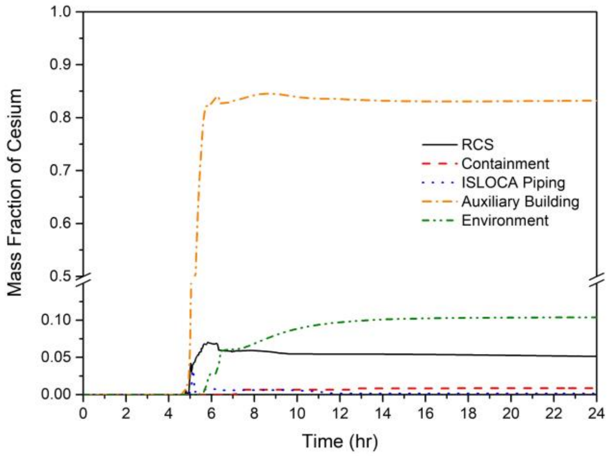

Fig. 2. The distribution of cesium for B-case

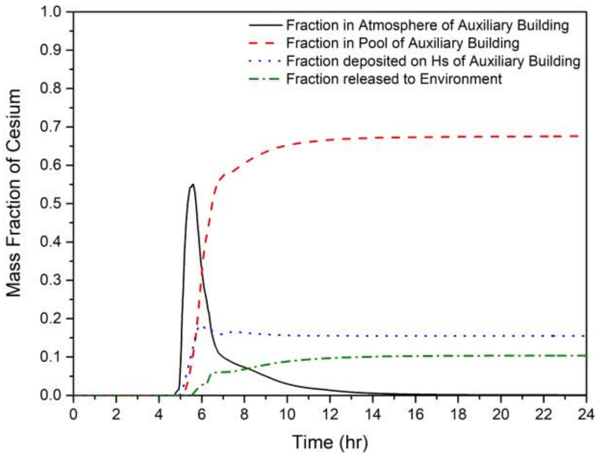

Fig. 3. The behavior of cesium in the auxiliary building for B-Case

3.2 Effect of pool scrubbing, filtration and mitigation strategy

Table III: The distribution of cesium for each case

	B-Case	F-Case	V-Case	P-Case	A-Case
RCS	5.1%	5.8%	5.3%	5.6%	6.0%
Containment	0.8%	1.6%	1.4%	5.3%	5.9%
ISLOCA pipe	0.1%	0.2%	0.1%	0.2%	0.2%
Auxilary Building	83.2%	81.9%	11.0%	78.9%	14.8%
Environment	**10.4%**	**9.8%**	**1.7%**	**9.7%**	**1.1%**
Filter	-	-	79.9%	-	71.6%

Table III shows the distribution of cesium at 24 hours after the accident occurred for each case.

Since the break-part in F-Case is flooded, some of the cesium that escapes to the auxiliary building is expected to be dissolved in the pool. Figure 4 shows the cesium mass distribution inside the auxiliary building. Comparing Figure 4 with Figure 3, the release fraction of cesium in F-Case is 9.8 %, which is only 0.6 % lower than that of B-Case. It is known that the smaller the bubble size and the deeper the submergence depth, the better the pool scrubbing. In this case, the break-size is 4 inches, which is too large, and the auxiliary building of the reference plant is submerged only about 1.5 m. Hence it is understood that the mitigation effect by the pool scrubbing is not large.

Since V-Case is operated with a ventilation system, cesium that has escaped into the atmosphere of the auxiliary building is filtered by a filter. In Figure 5, 79.9 % of cesium is filtered, and only 11.0 % will be retained in the auxiliary building. In V-Case, the release fraction of cesium to the environment is 1.7 % which is 8.7 % decrease compared to B-Case.

The significant difference of P-Case compared to B-Case is the amount of cesium retained in the containment building. Since PORV is open in P-Case, cesium begins to escape into the containment immediately after the gap release, in Figure 6. In this case, 5.3 % of initial inventory is retained in the containment, and it is about 6.6 times B-Case. The release fraction of cesium for P-Case is 9.7 %, which is only 0.7% lower than that of B-Case. Also, this mitigation strategy has side effects in which the pressure of the containment rises. In the B-case, the peak pressure of the containment is 127 kPa (a), whereas the peak pressure of the containment in P-Case rises to 142 kPa (a). However, it does not reach the design pressure (410 kPa) of the containment of the reference plant.

In A-Case, only 1.1 % of the initial inventory of cesium is released to the environment (Refer to Figure 7). This is the lowest release fraction among the sensitivity case. The containment, auxiliary building, and filter retain 5.9, 14.8, 71.6 % respectively.

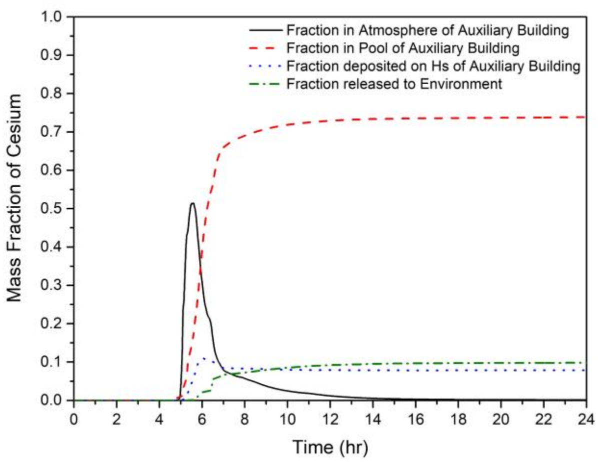

Fig. 4. The behavior of cesium in the auxiliary building for F-Case

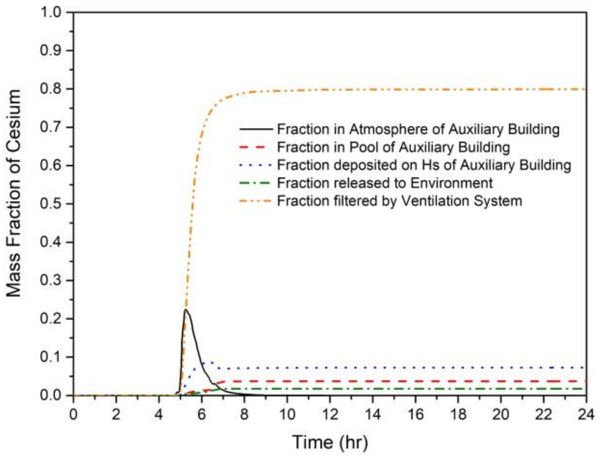

Fig. 5. The behavior of cesium in the auxiliary building for V-Case

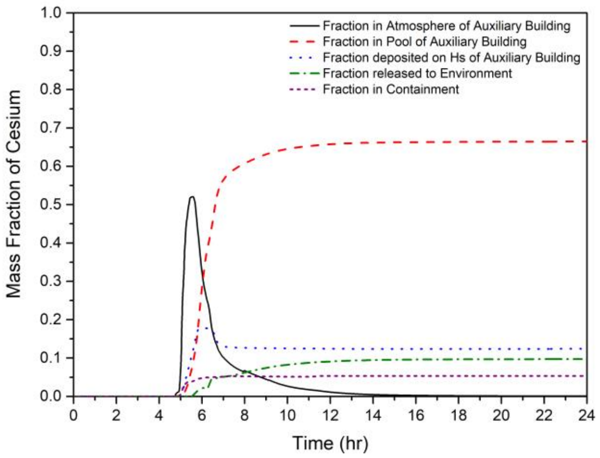

Fig. 6. The behavior of cesium for P-Case

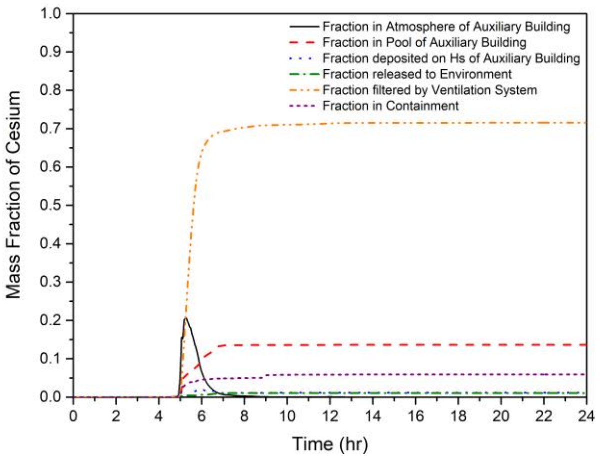

Fig. 7. The behavior of cesium for A-Case

4. Conclusions

In this study, accident progression and release fraction of cesium were analyzed under the ISLOCA condition, and the mitigation effects were evaluated. As a result, it was found that the mitigation effect due to pool scrubbing and PORV open was not large. By the way, the mitigation effect of the ventilation system was relatively large. Therefore, the ventilation system of the auxiliary building should be well managed so that it could work well even under the accident condition. Finally, the case with all retention mechanism and the mitigation strategy could reduce the release fraction of cesium up to 1.1 %. As the initial inventory of Cs-137 of the reference plant is $1.77\text{x}10^5$ TBq [7], a 1.1 % release means that about 1,947 TBq is released into the environment. This is over the safety goal of 100 TBq proposed by the Nuclear Safety and Security Commission (NSSC) in 2016. However, this safety goal has also a condition about core damage frequency (CDF), and the CDF for ISLOCA of the reference plant is $4.24\text{x}10^{-8}$ /yr [4], which is lower than the safety goal ($1.0\text{x}10^{-6}$ /yr). Besides, It should be noted that in these calculations a direct flow path(FL926) is assumed from the auxiliary building to the environment considering the confinement characteristics of the auxiliary building. Hence closer walkdown will be needed to check whether there is an open space to the environment.

ACKNOWLEDGMENTS

This work was supported by the Nuclear Safety Research Program through the Korea Foundation of Nuclear Safety (KOFONS), granted financial resource from the Multi-Unit Risk Research Group (MURRG), Republic of Korea (No. 1705001).

REFERENCES

[1] Sandia National Laboratories, State-of-the-Art Reactor Consequence Analyses Project, Volume 2: Surry Integrated Analysis, 2013, Vol. 2, NUREG/CR-7110.
[2] American Nuclear Society(ANS), Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs), ASME/ANS RA-S-1.2-2014, Jan 5, 2015.
[3] Korea Hydro & Nuclear Power Co., Ltd, Final Safety Analysis Report for Kori Unit 2, 1989.
[4] Korea Hydro & Nuclear Power Co., Ltd, Probabilistic Safety Assessment Report for Kori Unit 2: Initiating Event Analysis., 2015.
[5] Seok-Jung Han, Tae-Woon Kim, Kwang-Il Ahn, Journal of Radiation Protection and Research, 2017;42(2):106-113. https://doi.org/10.14407/jrpr.2017.42.2.106
[6] Keo-hyoung Lee, Kwang-il Ahn, and Seok-won Hwang, Analyses of Fission Product Retention under ISLOCA using MELCOR for APR1400, Transactions of the Korean Nuclear Society Spring Meeting, Korea, May 23-24, 2019.
[7] Seungwoo Kim et al., An Evaluation of Core Inventory for Reactor Types by using ORIGEN, NSTAR-19NS12-82, 2019.