Transactions of the Korean Nuclear Society Spring Meeting
Jeju, Korea, May 29-30, 2014

Environmental Source Term Analysis in A Sample LOCA Sequence of OPR1000

Tae-Woon KIM, Seok-Jung HAN and Kwang-Il AHN
[a]*Korea Atomic Energy Research Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, 305-353, Korea*
[*]*Corresponding author: twkim2@kaeri.re.kr*

Abstract

Since the accident of Fukushima, the assessment of source term effects on the environment is a key concern of the nuclear safety. As an effort to take into account the current knowledge of source term in off-site consequence analysis, the effects of the source term according to the containment response simulated by MELCOR code have been examined. In the view of the consequence, the containment response directly affects key features making a shape of plume behaviors to estimate the atmospheric dispersion, which are the release time, duration, and relevant source term features. The source term features for a large break LOCA sequence of a typical PWR plant according to the containment response (failure pressure and break size) have been investigated. In the results of the containment failure pressure, it has been observed that the release time varied 17.4 hour to 52.2 hour according to the containment failure pressure of 4.4 bar to 14.6 bar, respectively. This result potentially affects the radiological emergency strategies such as the public evacuation. Moreover, a considerable amount of the released source term is varied. This is resulted in about twice differences of the radiation exposure dose within the simulation cases. In the break size, it has been observed that the release source term is varied relatively small, but the release features to model the plume behavior are varied according to the break size.

1. Introduction

Historically, nuclear power plant is designed based on very conservative assumption of radiological source terms in accident conditions. In 1962 The U.S. Atomic Energy Commission published TID-14844, "Calculation of Distance Factors for Power and Test Reactors" which specified a release of fission products from the core to the reactor containment in the event of a postulated accident involving a "substantial meltdown of the core." This "source term," the basis for the NRC's Regulatory guides 1.3 and 1.4, has been used to determine compliance with the NRC's 10 CFR Part 100 “Reactor Site Criteria” and to evaluate other important plant performance requirements.

TID-14844 assumes that 50% of I (iodine) initial core inventory is released form the core to the containment. Regulatory Guides 1.3 and 1.4 specify that the source term within containment is assumed to be instantaneously available for release and that the iodine chemical form is assumed to be predominantly (91%) in elemental (I2) form, with 5% assumed to be particulate iodine and 4% assumed to be in organic form. These assumptions have significantly affected the design of engineered safety features.

Since the publication of TID-14844 (Ref. 1), significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. In 1995, the NRC published NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants” (Ref. 5). NUREG-1465 used this research to provide estimates of the accident source term that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative accident source term for a boiling-water reactor (BWR) and for a pressurized-water reactor (PWR). These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment.

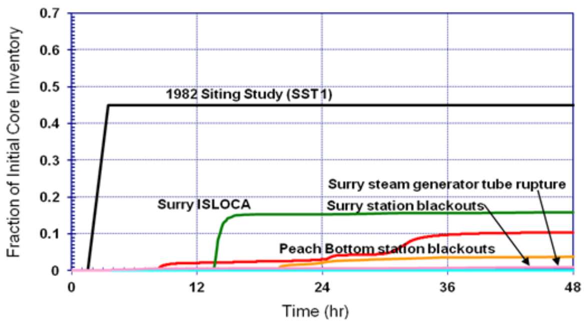
Table 3.11 Mean Values of Radionuclide Releases Into Containment for PWRs, Low RCS Pressure, High Zirconium Oxidation

Nuclide	Early In-Vessel	Ex-Vessel	Late In-vessel
N.G.	1.0	0	0
I	0.4	0.29	0.07
Cs	0.3	0.39	0.06
Te	0.15	0.29	0.025
Sr	0.03	0.12	0
Ba	0.04	0.1	0
Ru	0.008	0.004	0
La	0.002	0.015	0
Ce	0.01	0.02	0

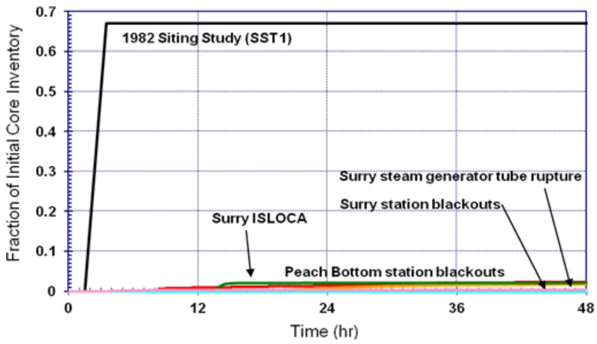
Even in NUREG-1465, which estimates a little bit less conservative than TID-14844, I (iodine) and Cs (cesium) release fractions are 40% and 30% of initial core inventories, respectively.

The amount of radioactive material release to environment in four units of the Fukushima Daiichi is one order of magnitude less than the amount of release of one unit of Chernobyl accident. In the Fukushima accident, reactor building of Unit 1 is exploded (2011.3.12 15:36) almost one day (exactly 25 hours) after the earthquake occurs (2011.3.11 14:46). The public near the site can be evacuated next day morning (2011.3.12). Hydrogen explosion at Unit 3 occurs 3 days (2011.3.14) after the earthquake and explosion occurs 4 days (2011.3.15) after the earthquake. There is enough time for the public to evacuate. The public inside 30km boundary of the site evacuated to outside of 30km from the site. Nobody died due to radiation exposure.

In 2012 US NRC published State-of-art Consequence analysis (SOARCA) report one and half year after Fukushima accident. They state that in-plant severe accident scenario and ex-plant public protection action are realistically evaluated as far as possible. The containment failure times (large amount of radiation release times) are more than 24 hours after the accident initiation in most cases even though the initiating event is earthquake induced station blackout (SBO). Therefore, there is enough time to evacuate general public. In considering the protective actions such as evacuation and sheltering, they assumed protective action scenario realistically as far as possible.



Iodine Release Fractionsin SOARCA report



Cesium Release Fractionsin SOARCA report

There is much uncertainty in source term and consequence analysis in PRA reports such as WASH-1400 and NUREG-1150. Since then, much knowledge are obtained and much technologies are improved on the source term estimation and consequence analysis fields. MELCOR and MACCS2 codes are representative codes in these fields. MELMACCS code connects between MELCOR and MACCS2. WinMACCS is a Window version of MACCS2. MAAP code can be used for in-plant source term analysis.

In lessons learned from the Fukushima accident, an improvement of knowledge and understanding of the off-site consequence analysis (CA) became a key concern of the nuclear safety. The CA is to assess an environmental effect of the radiation exposure due to the radioactive materials release to the environment during severe accidents of a nuclear facility. The CA is an integrated analysis including the assessments of radiological source term, atmospheric dispersion, dosimetry according to exposure pathways, health effects of radiation exposure. Among those parts, the radiological source term (shortly, source term)[1] as a comprehensive technical terminology covering the characteristics of radioactive materials escaped to the environment is a principal part of the CA of nuclear facilities.

Because there are a considerable limitation to provide the overall source term features needed in CA and a large degree of uncertainty in their features [3, 4], the simplified source term have been applied in the typical CAs. However, the severe accident analysis codes such as MELCOR and MAAP provide more detailed information for quantifying the source term features. The current state-of-art approaches to the source term estimation in CA are to use these codes. Recently, the US NRC SOARCA report showed an approach to utilize the detailed source term features provided by MELCOR code, of which features are to use a realistic off-site consequence analysis.

In the present study, as an effort to take into account the current knowledge of source term in CA, the source term features provided by MELCOR code have been utilized. In this work, a large break Loss-Of-Coolant-Accident (LOCA) sequence of a typical large dry containment PWR has been investigated. In a large LOCA sequences, the containment response is a key factor making a shape of the source term behaviors. In the view of the consequence, the containment response directly affects key features making a shape of plume behaviors to estimate the atmospheric dispersion, which are the release time, duration, and relevant source term features. The source term features according to the containment response (failure pressure and break size) simulated by MELCOR code have been examined by MACCS2 code for a CA.

Table 1: Typical information required in off-site consequence analysis

[1] This terminology is including the radioactive materials as constituent, radiological characteristics, physicochemical characteristics, relevant phenomenology in their transport, release pathways, amount of their release, etc.

Area	Feature	Element
Accident sequence	Scenario	state of key safety functions
	Phenomenology	progress of severe accident phenomena
	Release pathways	**containment response**
Radioactive materials inventory	Chemical features	classifications
	Radionuclide	total amount
Radioactive materials transport phenomena	Segments	transport (core/RCS/ Containment)
	Plume	**characteristics**
	Release	**release amount**
Dispersion features	Aerosol	size distribution
	Release Energy	sensible heat
	Time	**release time**
	Duration	**release duration**

2. SOURCE TERM PROJECTION APPROACHES TO CONSEQUENCE ANALYSIS

There are many features characterizing the source term, but the key features are to determine initial and boundary conditions of an atmospheric dispersion model such as (1) release amounts of source term, (2) release time, and (3) duration during a release phase. For an advanced analysis of atmospheric dispersion, the dispersion features of the source term such as aerosol size or sensible heat of plume are required.

Although a description of dispersion features depends on the atmospheric dispersion models, the typical parts of an atmospheric dispersion model consist of (1) the initial dimension of plumes, (2) plume rise characteristics, (3) deposition characteristics of radioactive materials during the dispersion. Typical information required in CA is shown in Table 1. Among these features, this study focuses on the containment response with the selected accident sequence to make the plume characteristics, release amount, release time, and release duration.

In the view of CA, the source term results provided by the severe accident codes are not directly adopted in CA because of the different modeling techniques. A process utilizing the source term results of the severe accident codes to CA is a kind of the projection technique. To derive the source term features needed in CA, it should assess the atmospheric dispersion model before characterizing the source term features. In this study, the required source term features have been derived based on MACCS2 code developed by Sandia National Laboratories (SNL) in USA. Because the atmosphere is a primary pathway of the radiological dispersion, atmospheric dispersion is a key model to CA. In MACCS2 code, a Gaussian plume model is adopted as a key model to describe the atmospheric dispersion:

$$\chi(x,y,z)=\frac{Q}{\bar{u}}\cdot\frac{e^{-y^2/2\sigma_y^2(x)}}{\sqrt{2\pi}\sigma_y(x)}\cdot\frac{1}{\sqrt{2\pi}\sigma_z(x)}\left(e^{-\frac{(z+h)^2}{2\sigma_z^2(x)}}+e^{-\frac{(z-h)^2}{2\sigma_z^2(x)}}\right)$$

$$=\frac{Q}{\bar{u}}\cdot\frac{1}{\pi\sqrt{2}\sigma_z(x)\sqrt{2}\sigma_y(x)}\cdot e^{-y^2/2\sigma_y^2(x)}\cdot\left(e^{-\frac{(z+h)^2}{2\sigma_z^2(x)}}+e^{-\frac{(z-h)^2}{2\sigma_z^2(x)}}\right)$$

Here χ is the time-integrated concentration of released radiation materials, Q is the total amount of released radiation materials, $\bar{u}$ is the wind speed, σ_y and σ_z are lateral and vertical dispersion coefficients, respectively, and h is the release height. Although the Gaussian plume is a static model, time-dependent features are treated in MACCS2 code using an hourly-based unit-time interval approach for released amounts within the limitation of four plumes. Key factors to represent a plume features using the source term results of the MELOCR code are manipulated considering the MACCS2 plume model features.

In MACCS2, plumes can be modeled up to four plumes, which are specified by a start time and duration. In the typical single plume model, short and long duration approaches are applied in CA according to case by case since a plume shape is determined by duration, of which the release concentrations are varied from high to low because of the conservation of the total amount of released source term (Fig. 1-(a)). Taking into account simulation results, the shapes of the release features could be projected in plume modeling.

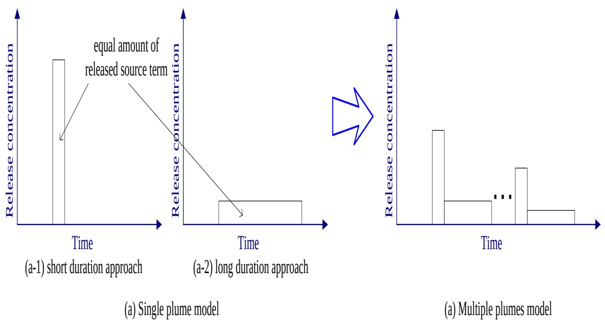
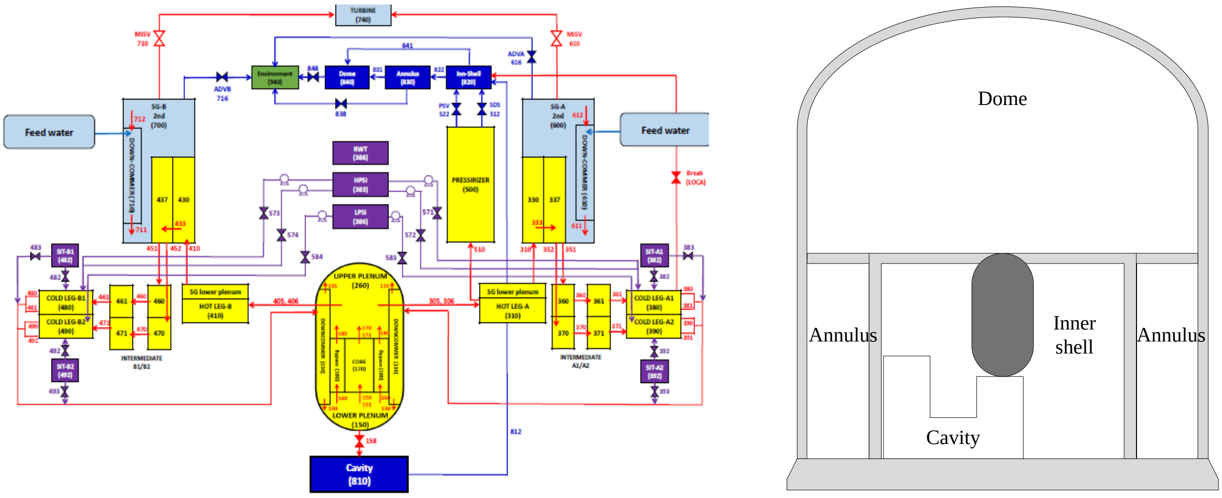


Fig. 1. Plume modeling approaches



(a) RCS model (b) containment model

Fig. 2. A nodalization diagram of the reference plant

3. SOURCE TERM AND CONSEQUENCE ANLYSIS

3.1. Plant Model in MELCOR

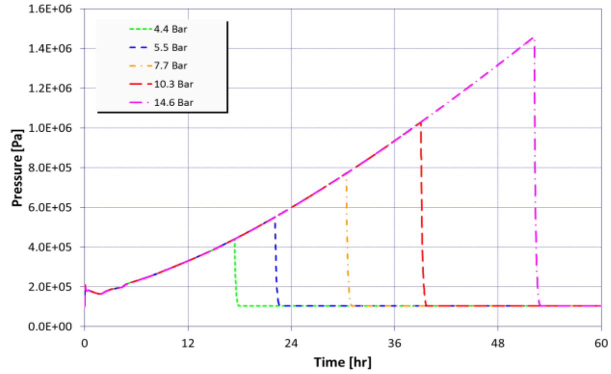
An application case, i.e., a loss-of-coolant-accident (LOCA) as a typical sequence reached to severe accident with an over-pressurization containment failure, was selected to investigate the source term behaviors on CA. The containment failure mode due to over-pressurization, although this is the most possible source term release pathway in LOCA sequence, has a large degree of uncertainty to apply the relevant parameters. Most of all, the containment failure pressure and break size are key parameters to determine containment response and the source term behaviors.

In this study, the effects of CA according to the variation of the containment failure pressure and break size have been investigated by MELCOR code (Version 1.8.6 YT). The reference plant for this work was adopted OPR-1000 type plants which are a Korean typical plant [9]. These plants are designed to two-loop (2 steam generator) type PWR with a 2815MW thermal power and housing a large dry containment. The reference plant model in MELOCR is shown in Fig. 2. The containment model adopted four control volumes such as (1) reactor cavity, (2) inner shell, (3) annulus, and (4) upper compartment dome.

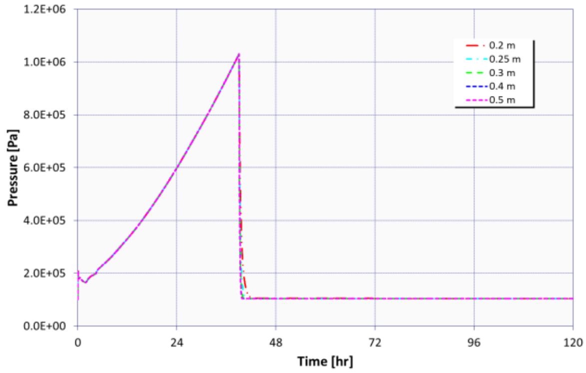
In this sequence, a dominant containment response is that the containment failure occurs by an over-pressurization over the containment design pressure. For this sequence, the cavity state is assumed as a dry state initially. The containment spray did not operate the early phase because the containment pressure did not reach to the operating condition (2.39E5 Pa) and it are assumed not working in the late phase because of the assumption of the recirculation failure. The accident progression of the given case is shown in Table 2.

During the severe accident progression initiated from a LOCA, the containment pressure is continuously increasing due to severe accident phenomena, which results in a containment failure. There is a large amount of uncertainty of the containment response. This study focused on key parameters in the containment response, i.e., the containment failure pressure and break size, of which effects on a CA were investigated.

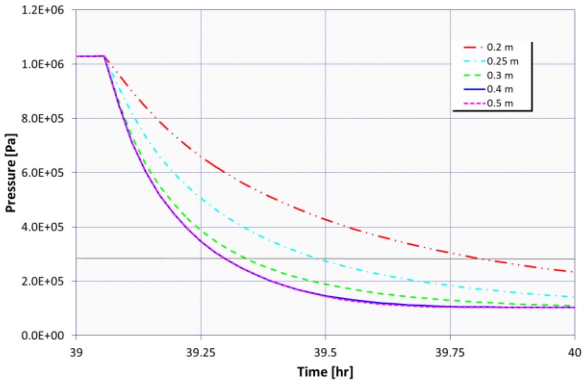
In this study, a six-inch (0.15 meter) break size (break area of 1.82E-2 square-meter) in a cold leg, which is a typical large-break LOCA sequence in the PSA [10], was taken into account. Among the sequences to reach the core damage, a sequence of the recirculation phase failure of safety injection from the containment sump after a dry-out of the water source (RWST) was adopted as a simulation case. This sequence is a highly ranked sequence among the LOCA-induced severe accident sequences.



(a) effect of containment failure pressure

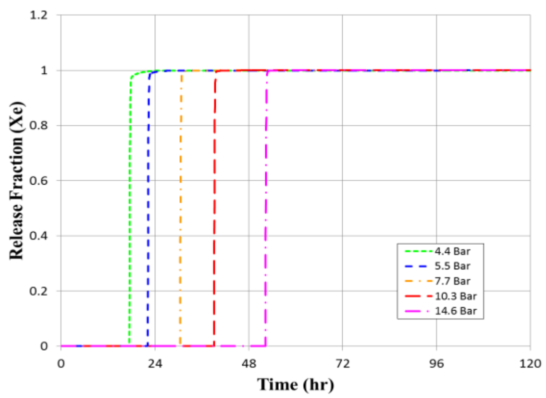


(b) effect of break size (diameter)

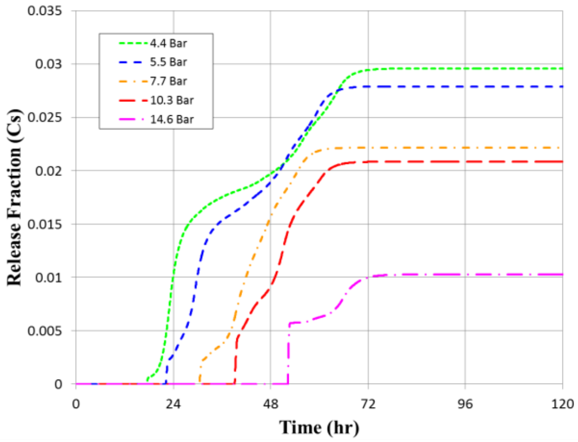


(C) containment pressure responses near failure time for the break sizes

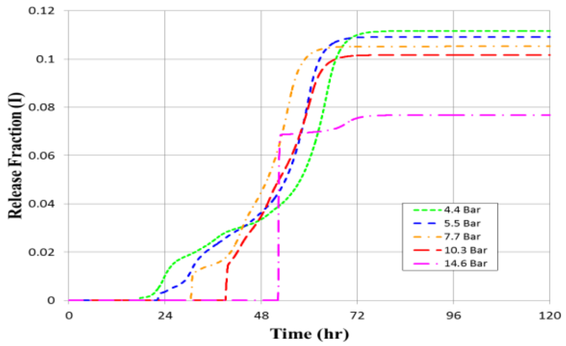
Fig. 3. Containment response according to failure pressure and break size



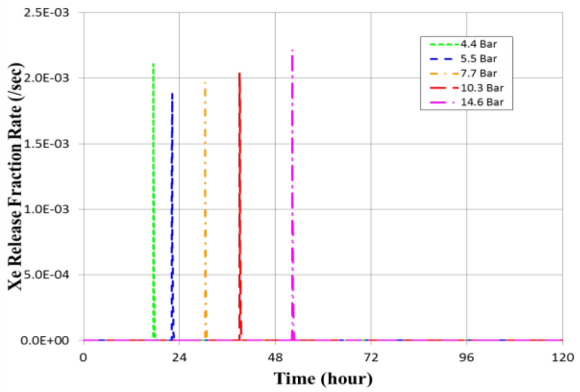
(a-1) release fraction of noble gas



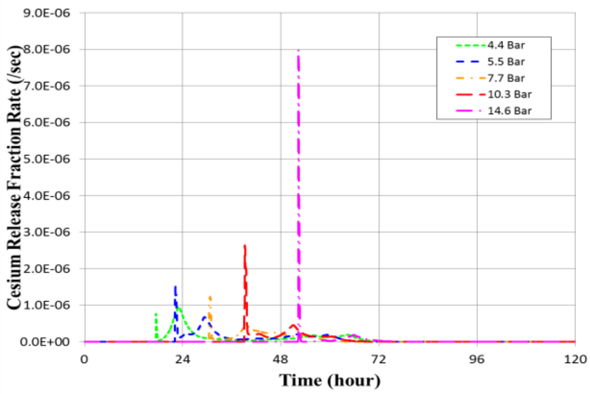
(b-1) release fraction of Cesium



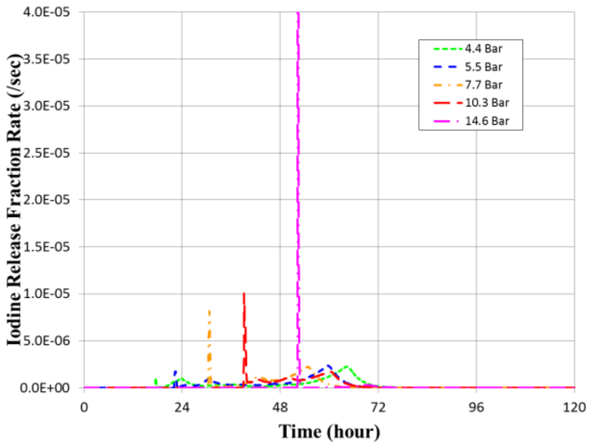
(c-1) release fraction of Iodine



(a-2) release fraction rate of noble gas



(b-2) release fraction rate of Cesium



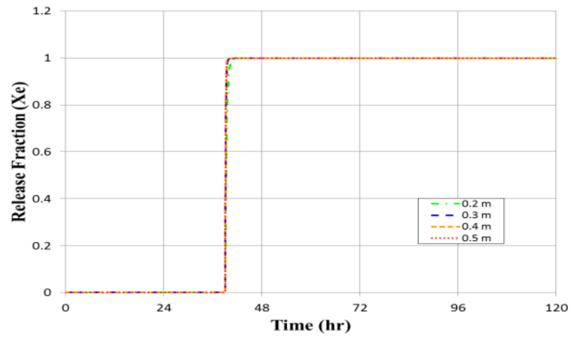
(c-2) release fraction rate of Iodine

Fig. 4. Source term behaviors according to containment failure pressure

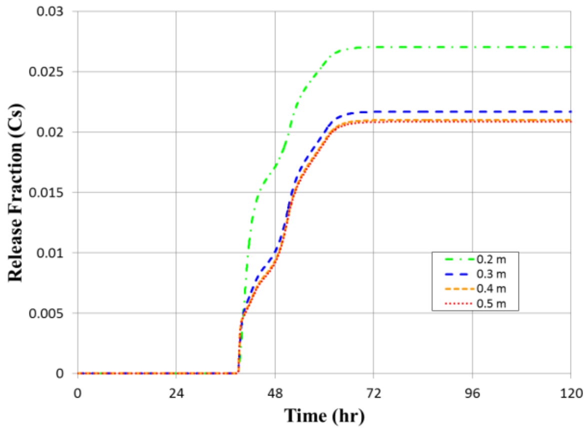
3.2. Source Term Analysis

The PSA report denoted that the range of containment failure pressure is varied from 4.4 bar (leak failure start point) to 14.6 bar (catastrophic rupture) [10]. For the containment failure pressure, five cases (4.4, 5.5, 7.7, 10.3, and 14.6 bar) were simulated (Fig.3-(a)). In this simulation, the break size of containment is assumed as 0.5 m inner diameter. Containment failure in each case occurs at about 17.4, 22.1, 30.4, 39.1, and 50 hour, respectively. It is noted that this result potentially affects the radiological emergency strategies such as the public evacuation.

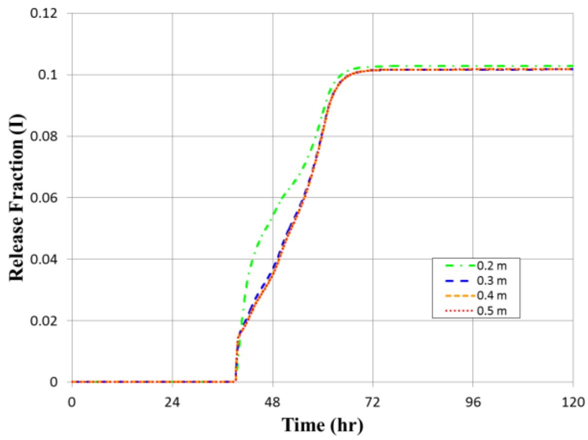
In these simulations, it was identified that the containment failure pressure affects the containment failure time and it was expected that the containment break size mainly affects the immediate source term behaviors. The source term behaviors (the release fraction and its rate) of noble gases, Cesium and Iodine according to the variation of the containment failure pressure and the containment break size are shown in the Fig.4 and Fig.5. Fig. 6 shows that the variation of the containment failure time (Fig. 6-(a)) and the release fraction of Cesium and Iodine (Fig. 6-(b)) according to the containment failure pressure. It is noted that Fig. 6-(b) delineates that a considerable amount of the release fractions according to the containment failure pressure are reduced to affect the radiological effect on environment. On the other hand, Fig. 7 reveals that the variation of the containment break size affects the source term behaviors, in particular the immediate behaviors near the containment failure time, are drastically changed.



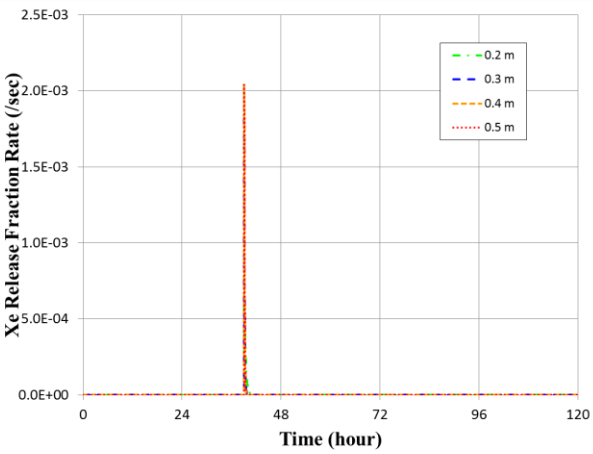
(a-1) release fraction of noble gas



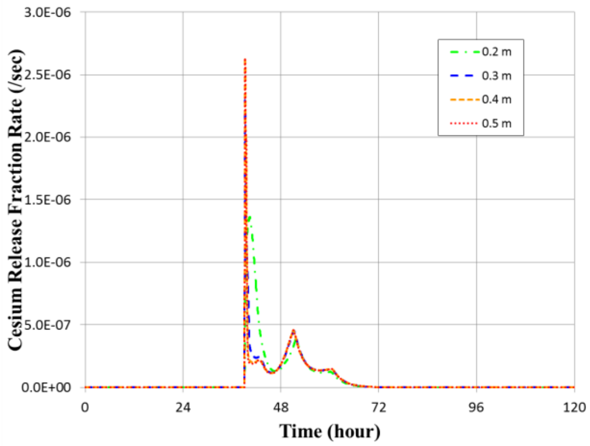
(b-1) release fraction of Cesium



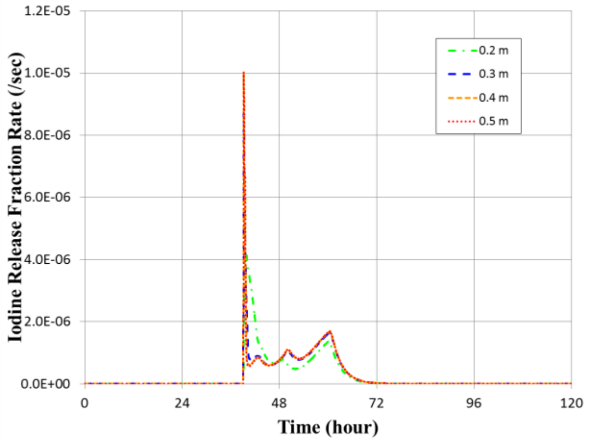
(c-1) release fraction of Iodine



(a-2) release fraction rate of noble gas

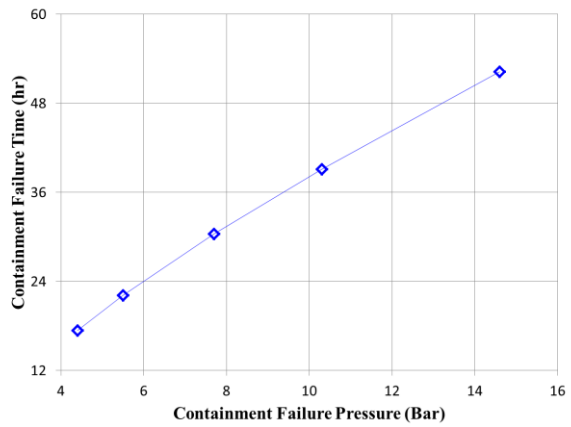


(b-2) release fraction rate of Cesium

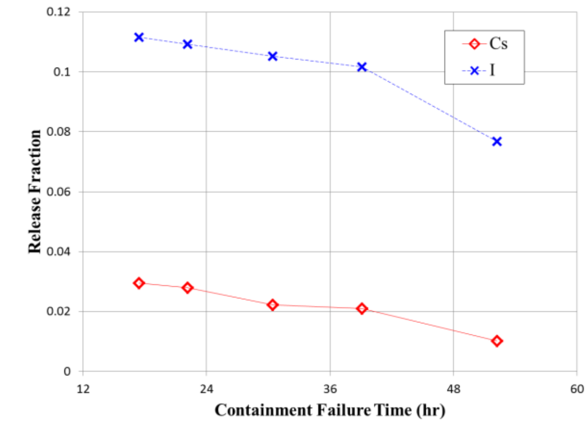


(c-2) release fraction rate of Iodine

Fig. 5. A source term behavior according to containment failure size

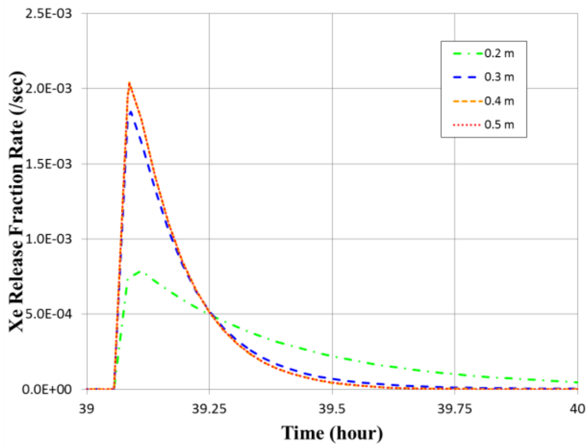


(a) containment failure time

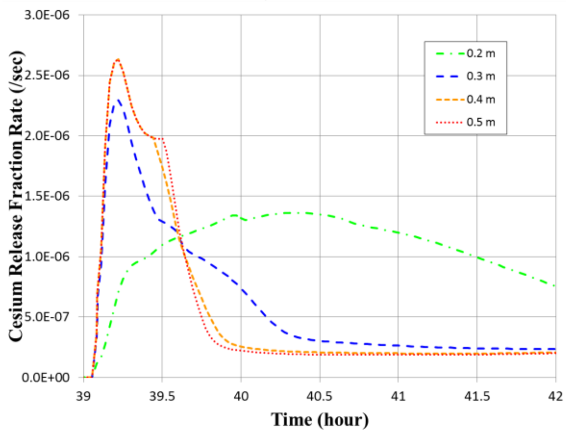


(b) release fraction vs. containment failure time

Fig. 6. Features of containment failure time and release fraction



(a) release fraction rate of noble gases



(b) release fraction rate of Cesium

Fig. 7. Variation of release fraction rate near failure time

3.3. Source Term Projection

The results of these simulations provide the basis of plume modeling for an atmospheric dispersion. Because this study focused on the effects of the source term according to the containment responses, the different plume models were adopted according to the types of containment response, i.e., containment failure pressure and break size. For the containment failure pressure, one plume model was applied in order to investigate their effects. One-hour duration was applied although a considerable amount of the residual was observed in the simulation results. As the results, Table 3 shows the characterization of this single-plume model. For each chemical group, almost all of the noble gases, maximum 3 % of Cesium and maximum 11 % of Iodine released to the environment.

3.4. Effects of the source term on the off-site consequence

The effects of the source term according to the characterization of source term aforementioned are simulated by MACCS2 code (WinMACCS Version 3.7). In this study, only three isotope groups (noble gases, Cesium, and Iodine) were considered, although nine isotope groups are treated for the radiological exposure in MACCS2 code. For assessing the specified consequence, weather condition should be fixed. In this case, the following weather condition applied:

- Wind speed: 3.2 m/s
- Atmospheric stability Class: D (neutral)
- Release height: 0 m (ground level release).

To calculate the radiation exposure dosimetry, the peak whole-body dose in the ground centerline under the plume provided by the default output of MACCS2 code were calculated and the default values of dose conversion factors (DCFs) in MACCS2 code were used. In this study, the relative peak whole-body dose comparing with the minimum calculated value was presented. The Fig. 8 shows the relative peak whole-body dose according to distance from a release point for the containment failure pressure. For the simulation cases (the containment failure pressure, 4.4 bar to 14.6 bar), maximum whole-body dose is about 50 Sv at 0.5 km distance from the plant. But it is decreased to 0.5 Sv in 10 km distance. Maximum value of the relative peak whole-body dose is about 100% larger than minimum value of them at the same distance, but it is decreased to about 50% at 10 km distance. Revealing the plumes characterization in Table 3, the whole-body dose for the lower containment failure pressure cases are sequentially highly ranked comparing with the higher containment failure pressure cases. This observation shows that higher containment failure pressure reduces the radiation exposure of the environment even except the effects of release start time.

Table 3. The plumes characterization for the containment failure pressure

Contain-ment failure pressure (Bar)	Failure Time (hr)	Release Fraction of Initial Core Inventory		
		Xe, Kr	Cs	I
4.4	17.4	0.999	0.0296	0.112
5.5	22.1	0.999	0.0279	0.109
7.7	30.4	0.999	0.0222	0.105
10.3	39.1	0.999	0.0210	0.102
14.6	52.2	1.000	0.0103	0.077

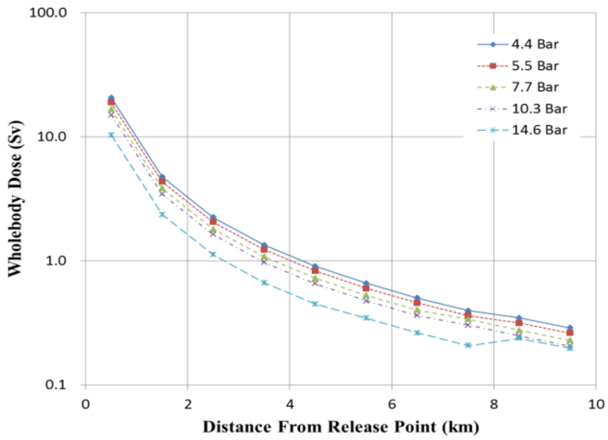
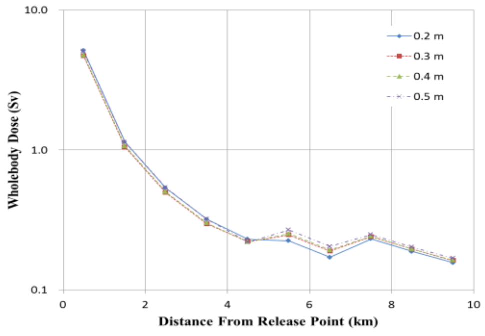
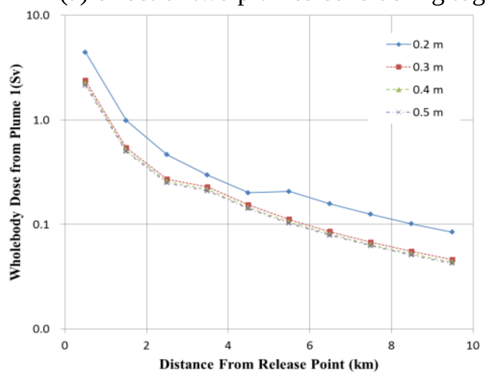


Fig. 8. The relative peak wholebody dose for the containment failure pressure (except the failure time)

Fig. 9 shows the radiation exposure dosimetry for the variation of containment break size. In this case, the primary effect is a reduction of whole-body dose due to two-plume model approach comparing with a similar case of 10.3 bar of the containment failure pressure in Fig. 9. This is due to the split of the amount of source term into two plumes. In particular, it is observed that the effects of first plume are a considerable difference (Fig.10-(b) although the effect of two plumes shows a little difference between the cases (Fig.10-(a). As the view point of radiological health effect, this result shows a meaningful effect of source term because a primary plume is a key contributor to assign an acute effect. Whole-body dose is about 0.1 Sv when only the first plume is considered, while it is about 0.3 Sv (except the case of 0.2 m diameter break) when two plumes are considered together at the distance of 6 km.



(a) effect of two plumes considering together



(b) effect of the first plume only considered

Fig. 9. Radiation exposure dosimetry according to the containment break size changes

From these examinations, it is presented that the characteristics of the containment response to affect the off-site consequence are as follows:

- Increased containment failure pressure delays the source term release time to govern the execution of the emergency plan, so roughly speaking that the better resistance of the containment against the severe accident progression may provide a margin of the execution of the emergency plan.
- In particular, a reduced source term according to the increased containment failure pressure may reduce the consequential health effects.
- A plume model approach to follow the containment response (i.e., release rate instead of cumulative measure of source term) may represent a realistic consequential effect. In the view of the off-site consequence, the conservative approaches may provide biased insights to reach a different decision making in the execution of the emergency plan.
- In this study, the accident progression and relevant severe accident phenomena has a large uncertainty and the simulation case does not provide overall aspects of these knowledge. To obtain useful insights, a more realistic approach to the accident progression and detailed assessment to show a containment response are required.

4. CONCLUDING REMARK

As an effort to take into account the current knowledge of source term in CA, the effects of the source term according to the containment response simulated by MELCOR code have been examined. The obtained results reveal that the containment response in a large LOCA may affect the off-site consequence. A realistic estimation in the off-site consequence analysis has been a long-lasting issue, due to large uncertainty in the source term estimation. In recent times, however, there were more understandings on severe accident phenomenology and progress in simulation tools such as MELCOR, making it possible to assess more realistically the off-site consequence. The present study examined a containment response focusing on the off-site consequence. Within this simulation case, the useful insights were obtained, but for making a sure insight, further study is recommended.

Acknowledgements

This work was supported by the Nuclear Research & Development Program of the National Research Foundation grant funded by the Ministry of Science, ICT and Future Planning, Korea.

References

[1] U.S. Atomic Energy Commission, TID-14844, "Calculation of Distance Factors for Power and Test Reactors", (1962)
[2] U.S. Atomic Energy Commission, Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors", (1974)
[3] U.S. Atomic Energy Commission, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors", (1974)
[4] US NRC, Accident Source Terms for Light-Water Nuclear Power Plants, NUREG-1465, (1995)
[5] US NRC, Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, (2000)
[6] Genki Katata et al, “Atmospheric discharge and dispersion of radionuclides during the Fukushima Dai-ichi Nuclear Power Plant accident. Part I: Source term estimation and local-scale atmospheric dispersion in early phase of the accident,” Journal of Environmental Radioactivity, Vol. 109, pp.103-113 (2012).
[7] US NRC, “Reactor Safety Study,” WASH-1400 (1975).

[8] US NRC, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (1990)
[9] US NRC, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," NUREG-1935, (2012).
[10] US NRC, NUREG/CR-2239, "Technical Guidance for Siting Criteria Development", Sandia National Laboratory, Albuquerque, NM, USA (1982)
[11] T. Homma et al, "Uncertainty and Sensitivity Studies with the Probabilistic Accdient Consequence Assessment Code OSCAAR," Nuclear Engineering and Technology, Vol.37, No. 3 (2005).
[12] J.C. Helton et al, "Robustness of an Uncertainty and Sensitivity Analysis of Early Exposure Results with the MACCS Reactor Accident Consequence Model," Reliability Engineering and System Safety, Vol. 48, pp.129-148 (1995).
[13] US NRC, "MELCOR Computer Code Manuals," NUREG/CR-6119 (2005).
[14] Electric Power Research Institute (EPRI), ""MAAP4, Modular Accident Analysis Program User''s Manual,"" EPRI Report prepared by Fauske & Associates, Inc. (1994).
[15] US NRC, "Code Manual for MACCS2," NUREG/CR-6613, (1997).
[16] http://www.opr1000.co.kr/