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Preliminary Study for Application of the New Safety Goal related with the Limitation of Cs-137 release

Mi Ro Seo*, Tae Young Shin
KHNP-CRI, Nuclear System Safety Laboratory, 70, 1312-gil, Yusung-Daero, Yusung-Gu, Daejeon, 305-343, Korea
**Corresponding author: mrseo9710@khnp.co.kr*

1. Introduction

After the Fukushima accident, the Korean regulatory body has considered that it should be necessary to revise the Safety Goal for domestic nuclear power plants. According to this consideration, the New Safety Goal was published in June, 2016. In the New Safety Goal, it is clearly stated that the Probabilistic Safety Assessment (PSA) should be performed with the proper technical appropriateness, the detail, and the scope in accordance with the integrated risk assessment against the accident for the nuclear power plants. Also, the newly added safety goal requires that the sum of the accident frequency that the release of the radioactive nuclide Cesium-137 to the environment exceeds the 100TBq should be less than 1.0E-6/RY.

This requirement is known to be come from the provision for preventing the long term ground contamination due to the release of radioactive material. However, there were so many concerns that this goal is so severe that the current design, even in the case of the constructing nuclear power plants, cannot meet this criterion. Especially for the operating nuclear power plants, since there were no mitigation facilities against the severe accident at the design stages, the application of this new goal is known to be much severe that the constructing nuclear power plants and it is necessary to develop the alternative methods to strengthen the safety of the operating nuclear power plants.

The purpose of this study is to review the new safety goal from the view point of severe accident analysis and probabilistic safety assessment, and to find the appropriate methods in order to meet that goal for the operating nuclear power plants.

2. Methods and Results

2.1 The amount of Cs-137 released

The mass of Cs-137 that is correspondent with the 100TBq is calculated as below. [1]

$$\alpha = \lambda \times n$$

where, α = Activity
λ = Decay Constant
n = number of atoms

According to this equation, it can be calculated that the mass of Cs-137 correspondent with the 100TBq is approximately 32g.

Generally, the mass of Cesium in the core inventory is calculated from 150kg to 380kg. And the mass of Cs-137 among the Cesium is approximately calculated from 4kg to 11kg based on the ORIGEN code calculation results. Since the containment failure events can occur when the mitigation facilities are not available during the severe accident, it is expected that the large amounts of fission products should be released to the environment. So, it is also expected that the large amount of Cs-137 exceeding the 100TBq can be released to the environment.

2.2 Source Term Analysis using MAAP code

In order to predict the actual amount of Cs-137 released during the severe accident, the preliminary assessment was performed for the representative source term category (STC) of OPR1000 type nuclear power plants using MAAP (Modular Accident Analysis Program) severe accident code. The major accident sequence for STC8 that was the STC group due to late containment failure (leak) was selected for the representative STC. The initiating event for STC8 is the Station Blackout. The core damage had been progressed due to the failure of Auxiliary feedwater system, also the opening of the valves in the safety depressurization system and the safety injection systems had been failed. Since the operation of the containment spray system had been failed, the partial leak of the containment has been occurred at the late period. [2]

Two versions of MAAP code (MAAP4.0.4 and MAAP5.0.3) were used to consider the effect of the update for the fission product behavior model. In the MAAP4, the cesium was release to the environment as the form of CsI and CsOH. However, in the newly developed MAAP5, some large portions of cesium were known to be released as the form of Cs_2MoO_4 as well as the form of CsI and CsOH. It is known that the Cs_2MoO_4 is easy to be absorbed to the structural material. So, it is expected that the amount of Cs-137 released to the environment calculated from MAAP5 is much smaller than that calculated from MAAP4.

The major results of MAAP4 and MAAP5 code run are summarized in Table 1. As shown in Table 1, there is no difference in view point of the progression of accident

In Table 2, the fission product release fraction to the environment calculated by MAAP4 and MAAP5 are summarized.

Table 1. Major Accident Progression

Case	Core Uncover (S)	RV Fail (S)	CV Fail (S)
MAAP4	6,702	14,205	129,600
MAAP5	7,323	14,791	129,604

Table 2. Fission Product Release Fraction

MAAP4		MAAP5		MAAP5/MAAP4
NOBL,IN	9.89E-01	NOBL,IN	9.93E-01	100.40%
CSI	4.60E-02	CSI	1.86E-01	404.94%
TEO2	2.57E-02	TEO2	2.88E-02	112.25%
SRO	9.89E-05	SRO	1.58E-03	1599.66%
MOO2	2.77E-03	MOO2	6.19E-03	223.96%
CSOH	2.10E-02	CSOH	8.80E-02	418.33%
BAO	1.35E-03	BAO	4.43E-03	329.70%
LA2O3	1.33E-05	LA2O3	8.09E-05	608.48%
CEO2	2.87E-05	CEO2	3.45E-04	1204.82%
SB	5.33E-02	SB	7.69E-02	144.28%
TE2	4.08E-05	TE2	3.91E-03	9598.53%
UO2,ACT	4.28E-08	UO2,ACT	7.03E-07	1642.49%
AG	1.49E-02	AG	2.99E-02	200.22%
		I2	9.89E-01	
		CH3I	9.89E-01	
		CS2MOO4	8.52E-03	
		RUO2	3.02E-05	
		PUO2	5.45E-05	

As shown in Table 2, the fission product release fraction calculated by MAAP5 is more conservative than that calculated by MAAP4.

In Table 3, the mass of Cesium released to the environment was converted using the release fraction and initial core inventory.

Table 3. Mass of Cesium released to the Environment

		Mass (kg)	Cs ratio	Cs Weight (kg)	Released Mass Fraction	Cs release mass (kg)
MAAP4	CsI	36.04	0.5115	18.4362	1.2480E-02	0.2301
	CsOH	254.16	0.8866	225.3259	6.4690E-03	1.4576
	Cs_2MoO_4	-	-	-	-	-
	sum			243.7621		1.6877
MAAP5	CsI	45.38	0.5115	23.2141	1.7610E-02	0.4088
	CsOH	122.66	0.8866	108.7444	1.2150E-02	1.3212
	Cs_2MoO_4	274.23	0.6243	171.2119	3.5100E-03	0.6010
	sum			303.1704		2.3310

As shown in Table 3, the mass of cesium released to the environment due to containment partial leak is calculated in the range from 1.7kg to 2.3kg. In general, it is known that the fraction of Cs-137 in the Cesium is about 30%. So, the mass of Cs-137 released to the environment for STC8 (late containment fail, partial leak) is calculated in the range from 0.5kg to 0.7kg. And this amount of Cs-137 is corresponding to about 1,500~2,200TBq.

2.3 Conclusion based on the Source Term Analysis

The accident sequence for STC8 used in this preliminary assessment is not the most severe case because the containment failure mode is not the rupture but the partial leak. Although the released mass of Cs-137 can be different for each STC accident sequence, it is clearly expected that the amount of Cs-137 for each of all STCs exceed the 1,000TBq.

In the case of newly designed nuclear power plants, such as APR1400 type, since the design features against the severe accident, such as the ECSBS (Emergency Containment Spray Backup Systems), are already installed and the emergency alternative facilities, such as movable diesel generators, are installed according to the Post-Fukushima activities, it is expected that they can meet the New Safety Goal.

However, from the viewpoint of the amount of Cs-137, it is judged that the operating plants cannot meet the New Safety Goal.

2.4 Consideration of the Frequency

According to the above analysis, in the case of the operating nuclear power plants such as OPR1000 and Westinghouse type, the New Safety Goal can hardly be met in terms of the amount of Cs-137 released. So, the approach from the aspect of frequency is considered to be necessary.

The accident frequencies that release the Cs-137 are determined by the STC quantification which came from the results of Containment Event Tree quantification in the Level 2 PSA. The simple structures and logic for Level 2 PSA methodology is shown in Figure 1.

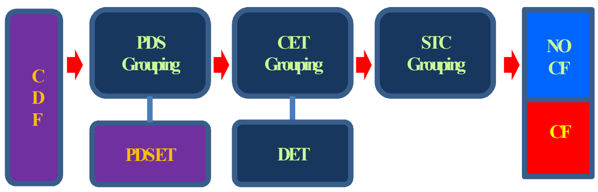

Figure 1. Structure of Level 2 PSA

As shown in Figure 1, the core damage frequency is divided into the frequency for no containment failure (No CF) and for containment failure frequency (CF). In the whole process of Level 2 PSA, the additional frequency is not added to the CDF except the stage for construction of PDS ET (Plant Damage Status Event Tree).

So, the improvement provisions in the Level 2 PSA process can be derived as bellows

1) The optimization of DET branch probabilities
2) Special CET reconstruction in the case of Bypass events
3) Consideration of facilities against the severe accidents

Among the above methods, the 1st option is not expected to have some great effect because the scope of optimization is limited by the uncertainties related with the severe accident phenomena.

The 2nd option is only applicable to Bypass events.

For the 3rd option, there are so many limitations to install the new facilities in the operating nuclear power plants, such as the space and the radioactivity in the containment. And most of these facilities are used only for the mitigation of the specific phenomena, so it is natural that the effects of these facilities confined to some containment failure mode. For example, the effect of the containment filtered venting system that is in the process of installing the domestic NPPs is known to confine the late containment failure due to over-pressurization.

For these reasons, the consideration for frequency should have to be started at the Level 1 PSA in order to reduce the core damage frequency. To reduce the core damage frequency, the initiating event frequency is in progress of re-investigation and the remedies to reflect the emergency alternative facilities and actions developed by the Post-Fukushima Action into the PSA model are being researched.

In addition to these, the studies for re-constructing the PDS event tree are in progress. Even though the construction of the PDS event tree is very important because it links the Level 1 PSA with the Level 2 PSA, the current PDS model did not consider the continuous recovery action for mitigation systems after the core damage. For example, the availability of the containment spray system firstly assessed in the Level 1 event tree. If it was not available at this stage, the recovery of this system is not considered in the PDS event tree. However, actually, the plant staffs should have to recover containment spray system in order to perform the mitigation action according to the Severe Accident Management Guideline. So, if the recovery of the containment spray can be considered in the PDS event tree, it can be expected that the containment failure frequency should be greatly reduced.

3. Conclusions

In order to strengthen the safety for domestic nuclear power plants, all of the domestic operating nuclear power plants are required to prepare the Accident Management Plan within 3 years. Also, this Accident Management Plan should meet the New Safety Goal including the requirement that the sum of the accident frequency that the release of the radioactive nuclide Cs-137 to the environment exceeds the 100TBq should be less than 1.0E-6/RY.

Since the operating nuclear power plants was not designed against the severe accident and they have the limited exclusive mitigation facilities, it is not easy to meet the New Safety Goal. So, it is necessary to develop the alternative methods to meet the New Safety Goal.

In this study, the amount of Cs-137 released to the environment for the representative accident sequence assessed to find the method that the amount of Cs-137 released can be reduced using MAAP code. However, according to the MAAP calculation even though it is performed only for the one STC, it can be expected that the amount of Cs-137 released to the environment for each of all STCs is over 1,000TBq.

Therefore, it is necessary to consider the approach from the aspect of PSA model. In the Level 2 PSA model, the improvement effects is so limited because of the uncertainties related with the severe accident phenomena and the difficulties in the installation of new facilities. In the aspect of Level 1 PSA model, the studies for reflecting the emergency alternative facilities and SAMG actions to the PSA model are progressed.

However, it remains so many issues have to be resolved such bellows.

1) Can the emergency alternative facilities and SAMG actions be credited in the PSA model?
2) How many times does it take to recover the containment spray system after the core is damaged?
3) Are there any resolutions for the requirement in the New Safety Goal if the scope of PSA is enlarged to the all mode all scope?

The New Safety Goal for the operating domestic nuclear power plants is so severe that the provisions for strengthen the safety if they cannot meet the goal.

Therefore, it is necessary to develop the new PSA model reflecting the emergency alternative facilities and SAMG actions to assess the integrated risk including the methodologies to reduce the uncertainties. In addition to this, it is judged that the more fundamental considerations for the safety enhancement plans of the domestic operating nuclear power plants are needed.

REFERENCES

[1] John R. Lamarsh, Anthony J. Baratta, “Introduction to Nuclear Engineering” 3rd Edition, Prentice Hall, 2001
[2] Probabilistic Safety Analysis Report for Shin-Kori 1&2, KHNP, 2011