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Dose assessment for Multi-Unit Simultaneous Accident using MACCS

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1. Introduction

Since the Fukushima accident in 2011, there has been growing attention on the safety of multiple NPPs located in a single site. In particular, there is a growing concern about off-site consequences due to the release of large scale radioactive material caused by multi-unit simultaneous accident. In this situation, many studies are performing in various countries to develop off-site consequence analysis methodologies.

In this study, analysis framework for multi-unit simultaneous accident was developed to find out the exposure dose from released radioactive material. The developed analysis framework is consisted of two steps as shown in Figure.1.[1]

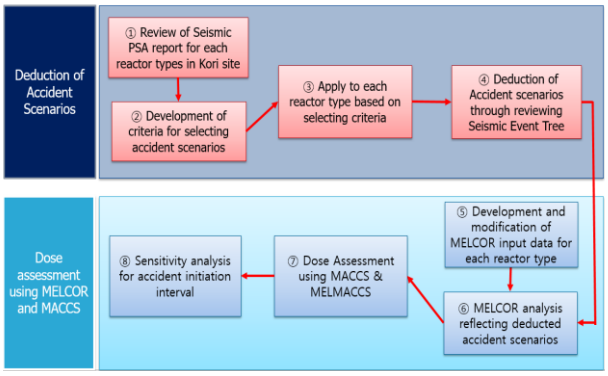

Figure.1 Dose Assessment Framework for Multi-Unit Simultaneous Accident

First step is deduction of accident scenarios. In this step, accident scenarios were derived through reviewing seismic event tree for each reactor type based on core damage frequency ranking. Second step is dose assessment using MELCOR[2] and MACCS[3]. In this step, the released source-term were calculated using MELCOR based on deducted accident scenarios. The calculated source term was converted to MACCS format by entering it in MELMACCS. Finally, dose assessment was performed through MACCS and sensitivity analysis was performed on accident initiation interval.

2. Deduction of Accident Scenarios for each reactor type

Through review of various research results which is related to multi-unit accident, we concluded that seismic event is the most possible cause of multi-unit simultaneous accident. So, in this study, it was assumed that seismic event was occurred in target site. The accident scenarios were derived through seismic PSA report for each reactor type.[4] The screening process of accident scenarios is as follows.

Step 1: Review of seismic-induced initiating events
Step 2: Primary screening based on CDF ranking
Step 3: Determining whether MELCOR analysis is possible or not

The accident scenarios of each reactor were deducted according to above screening process. The final accident scenarios are shown in Table 1.

Table.1. Accident Scenarios for each Reactor Type

Unit(Reactor Type)	Accident Scenario
Unit 1(Type A)	S-CM1)+EDG fail (Assume SBO)
Unit 2(Type B)	S-NCWP2)+EDG fail (Assume SBO)
Unit 3(Type B)	S-CCWP3)-2
Unit 4(Type C)	S-LOEP4)+ EDG fail (Assume SBO)
Unit 5(Type C)	S-LOPCS5)+ EDG fail (Assume SBO)
Unit 6(Type D)	S-LOOP-18+EDG fail (Assume SBO)
Unit 7(Type D)	S-LOOP-5

1) Direct core damage
2) Loss of nuclear service cooling water and LOOP
3) Loss of component cooling water and LOOP
4) Loss of essential power
5) Loss of plant control system

3. Source-term Analysis Results for each Accident Scenarios

3.1 Simulation of Accident Scenarios

KINS has MELCOR input models for each reactor type in target site. Source-term analysis was performed for each accident scenarios which are deducted in chapter 2. Some input data such as system operation, operator action timing were reflected in the MELCOR to analyze severe accident phenomena for deducted accident scenarios. The Stress Test results, EOP, SAMG were utilized for operator action timing and the other inputs, since only simple operation of each system can be verified in the event tree. It is assumed that severe accident coping system were not considered for conservatism. We analyzed up to 72 hours for all reactor types.

3.2 Source-term Analysis Results

The following variables can be printed out by extracting source-term information using MELMACCS.[5]

- Release start time
- Release duration time
- Plume release height
- Sensible heat
- Plume density
- Plume release velocity
- Release Fraction

Among the above variables, the release fraction has the greatest effect on the off-site consequence assessment. The release fraction for unit 1 SBO and unit 7 LOOP are as follows.

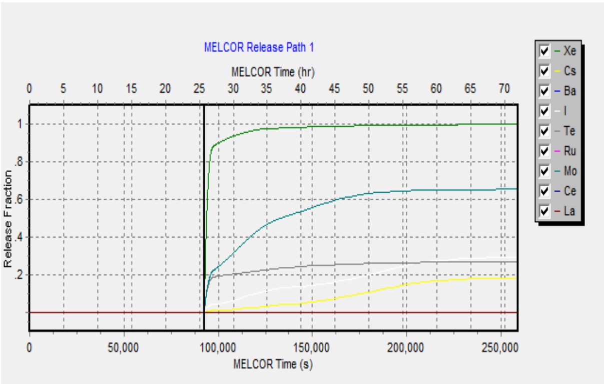

Figure 2. Release Fraction for Unit 1 SBO

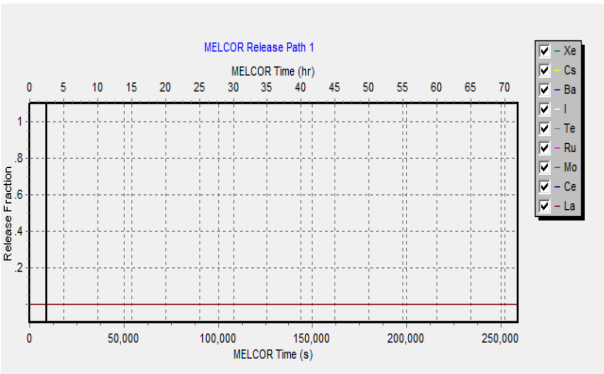

Figure 3. Release Fraction for Unit 7 LOOP

In case of Type A and Type B, the containment was failed as the accident progressed because the calculated pressure reached the containment failure pressure. On the other hand, the containment of Type C and Type D were not damaged since the calculated pressure caused by each accident scenarios did not reach the failure pressure. So, there is no released radioactive materials except for design leakage.

In this study, dose assessment was conducted by entering the source term results for each accident scenario derived using MELMACCS.

4. Improvement of MACCS input model and Analysis results

4.1 Improvement of MACCS input model

In this study, MACCS input model was improved by reflecting SOARCA[6] variables for more realistic analysis. All the SOARCA variables are reviewed, and we selected some variables which can be applied in domestic analysis except for site-specific data. Especially, some variables which are related to atmospheric dispersion and deposition were changed reflecting the latest experimental data. And dose conversion factor and risk conversion factor were updated with ICRP and FGR recommendations.

In the past, most of off-site consequence analysis in Korea assumed that only one plume is released at once. So it did not reflect the weather changes after plume was released. To solve these problems, the number of released plume was increased, and the release duration time of each plume was matched to collection interval of meteorological data, so the weather changes could be considered.[7] The Table 2 shows the evaluated dose results for past model and developed model using same source-term and site-specific data.

Table.2 Dose Results Comparison between Existing and Developed MACCS model

	Distance	Existing MACCS Model	Developed MACCS Model	Change rate(%)
Peak dose Found on spatial grid(Sv)	0.6-0.7 km	2.01E+01	1.83E+00	-91
	1.6-2.0 km	8.11E+00	7.61E-01	-91
	6.0-8.0 km	1.41E+00	1.80E-01	-87
	16.1-20.0 km	2.94E-01	4.75E-02	-84

The results show that the developed MACCS model represents up to 91% reduction compared to existing MACCS model. In this study, developed MACCS model was used for realistic analysis.

4.2 Dose assessment results for multi-unit simultaneous accident.

In this chapter, dose calculation was performed using developed MACCS input model and MELCOR analysis results. WinMACCS version 3.10.0 adds a 'Combinesource' function that can evaluate the multi-source term.[7] The 'Combinesource' can integrate up to 500 source terms and it can analyze multi-source term by adjusting the time interval(time off-set) of each source term. The Figure below shows the overall concept of 'Combinesource'.

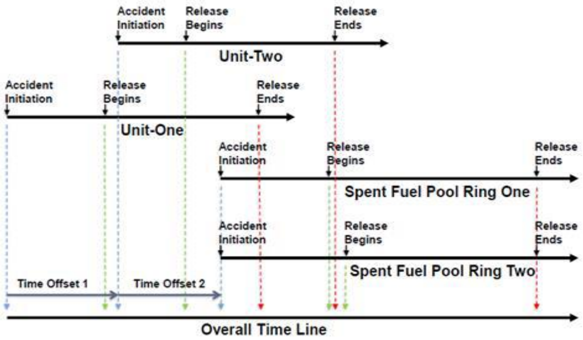

Figure 4. Overall concept of 'Combinesource' function

After combining 7 source-term using 'Combinesource' function, dose assessment was performed for simultaneous multi-unit accident. In order to verify the effectiveness of the relocation, analysis was carried out by separating whether relocation was performed or not. The emergency response including sheltering and evacuation was not considered. The 2011 meteorology data and 2009 population data were used. The results are as follows,

Table.3 Exposure Dose Calculation Results

	Distance	Case 1	Case 2
		Relocation	No Relocation
Peak dose(Sv)	600 m	4.47E-03	1.39E+02
Conditional Early Fatality Risk	1.6 km	0.00E+00	2.92E-01
Conditional Cancer Fatality Risk	16 km	4.26E-05	2.92E-02

In Case 1, the peak exposure was calculated as 4.47E-03 Sv at 600 m, which is the shortest distance that residents near a nuclear power plant can reside. And the conditional early fatality risk at 1.6 km was 0. In Case 2, the peak dose was 1.39E+02 Sv, and conditional early fatality risk was 2.92E-01. It is confirmed that if only relocation was performed, exposure dose decreased by about 1,700 times. Through this results, we confirmed the importance of realistic and well-planed emergency response planning.

4.3 Sensitivity Analysis for Accident initiation interval

In this chapter, sensitivity analysis was performed for accident initiation interval of each reactor types. It was assumed that the accident initiation time of same reactor type was identical each other. The accident sequence took place in the order in which radioactive materials are released to the offsite using MELCOR results. The accident initiation interval for each reactor type are as follows,

Table.4 Accident Initiation Interval for Each Reactor Type

Reactor Type	Accident initiation interval (hr)			
	Case A	Case B	Case C	Case D
	(2hr interval)	(4hr interval)	(6hr interval)	(8hr interval)
Unit 1 (Type A)	0	0	0	0
Unit 2,3 (Type B)	2	4	6	8
Unit 4,5 (Type C)	4	8	12	16
Unit 6,7 (Type D)	6	12	18	24

The sensitivity analysis was performed each accident initiation interval which are shown in Table 4. The sensitivity analysis results are as follows.

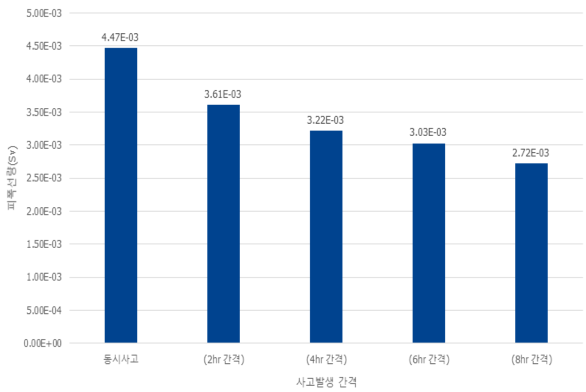

Figure 5. Sensitivity Analysis for Accident Initiation Interval

As shown in Figure 5, the exposure dose was reduced by 20% by delaying the interval only two hours compared to the simultaneous accident (Base case). It was also found that the reduction rate of exposure dose from simultaneous accident to 2hours interval was greater than that of other intervals. Through sensitivity analysis, it was concluded that the accident response at the early phase is important in term of exposure dose.

5. Conclusion

In this study, dose assessment framework of multi-unit simultaneous accident was developed for various reactor types in target site. The MELCOR and MACCS codes were used to calculate off-site consequences using developed analysis framework. The results show that if only relocation was performed, the exposure dose was significantly reduced compared to no-relocation case. Sensitivity analysis was also performed to identify the effect of accident initiation interval. It was found that the reduction rate of exposure dose in the early stage was greater than that of other intervals. Through this study, it was confirmed that the off-site effect can be minimized if well-planned emergency response and intensive resource input in early stage were performed. And this research results can be utilized to establishing a regulatory framework for deterministic analyses in the future.

ACKNOWLEDGMENT

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REFERENCES

[1] Y.J LEE, Multi-unit Off-site Consequence Analysis for Korean NPPs using WinMACCS, ASRAM 2018.
[2] U. S. NRC, MELCOR Computer Code Manuals. NUREG/CR-6119, 2000.
[3] U. S. NRC, Code Manual for MACCS2, NUREG/CR-6613, 1998.
[4] KHNP, Probabilistic Safety Assessment reports for reference plant(confidential)
[5] Katherine McFadden, MelMACCS Models Document, 2016.
[6] U. S. NRC, MACCS Best Practices as Applied in the State-of-the Art Reactor Consequence Analyses(SOARCA) Project, NUREG/CR-7009, 2014.
[7] Nate Bixler, MACCS Overview, 7th International MACCS Users Group Meeting, 2017.