Preliminary MELCOR Calculations for Dose Assessment of Multi-Unit Simultaneous Accidents

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1. Introduction

Since the Fukushima NPP accident, simultaneous accidents for multiple reactor units on the site initiated by external hazards have been considered as the safety issue. Many countries are carrying out the studies to develop a methodology of off-site consequences analysis for the multi-unit. In this regard, the framework for dose assessment of the multi-unit was developed using MELCOR and MACCS [1]. Fig. 1. shows the framework.

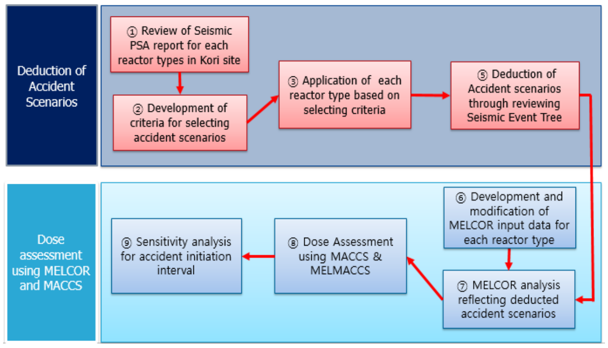

Fig. 1. Developed Framework for Dose Assessment of Multi-Unit Simultaneous Accident [1]

In this study, preliminary MELCOR calculations were conducted for dose assessment of multi-unit simultaneous accidents according to the framework. Accident scenarios were derived and selected from the review of PSA report based on seismic event tree and Core Damage Frequency (CDF) rank. Lastly, source terms released from containment to environment were calculated for the scenarios using MELCOR.

2. Postulated Accident Scenarios for Multi-unit Site

Among various external hazards, it was concluded that seismic event is most likely to cause multi-unit simultaneous accidents after reviewing relevant research results. Having a different kind of 6 PWR units in the postulated site was considered as the multi-unit site. The accident scenarios were derived from the seismic PSA report for the respective reactor type. Table I shows the postulated accident scenarios depending on the CDF in the report. Station-Black-Out (SBO) was the dominant scenario for all units. S-CCWP-2 scenario was particularly selected for the unit 3. It is the scenario in which reactor coolant pump seal failed due to loss of offsite power with loss of component cooling water of the primary system. In this scenario, secondary heat was successfully removed for a short time, but core damage followed. In addition, S-LOOP-5 scenario was determined for the unit 6. It is the scenario where core damage occurs because of loss of offsite power and failure of the safety injection to RCS though the RCS depressurization using safety relief valves at the pressurizer succeeded.

Table I : Postulated Accident Scenarios for the Units

Unit No.	Accident Scenario	Description
1	SBO	-
2	SBO	-
3	S-CCWP-2	Loss of off-site power, Loss of component cooling water of primary system, RCP seal failure, Secondary heat removal
4	SBO	-
5	SBO	-
6	S-LOOP-5	Loss of off-site power, RCS depressurization

3. MELCOR Modeling

The preliminary calculations for the postulated scenarios, a total of 6 cases, were performed with the MELCOR1.8.6 input models of the respective reactor type [2]. The calculations continued for 72 hours since the accidents initiated at 0 sec. A variety of information for operator actions and timing was utilized from the stress test report, Emergency Operating Procedures (EOP) and Severe Accident Management Guideline (SAMG) in order to simulate the scenarios based on simple event tree. On the other hand, any severe accident mitigation actions were not considered in this calculation. Operation of cavity flooding system after accident initiation was only assumed for some unit.

Revised CORSOR-Booth model for high burn-up fuel and hygroscopic model were introduced to the calculations. Reference source terms served in MELCOR were considered as the initial core inventory depending on the thermal power [2].

In order to simulate the failure mechanism of containment, sizes of leak and rupture depending on a pressure in the containment were taken into account with respect to consequence analysis of PSA. In other words, leak and rupture of containment occurred when the containment pressure reached the specific point from PSA results. Design basis leakage rate was considered when the containment pressure unreached the leak and rupture point until the end of calculation time. The locations of leak and rupture were taken into account at the region of containment on the ground level.

4. Calculation Results

Fig. 2. shows the containment pressure behavior for the scenarios of the unit 1, 2 and 3. The orange line shows containment failure of the unit 1 SBO occurring at about 26 hours. At that time, the pressure decreased drastically. The grey line demonstrates containment failure of the unit 2 SBO occurring at about 39 hours. The yellow line shows containment failure of the unit 3 S-CCWP-2 occurring at about 53.4 hours. The failure time of unit 3was significantly delayed in comparison with the unit 2. The RCS was depressurized by secondary heat removal due to operation of auxiliary feed water and the coolant from the Safety Injection Tanks (SITs) was injected into the RCS. Therefore, overall accident sequence was delayed.

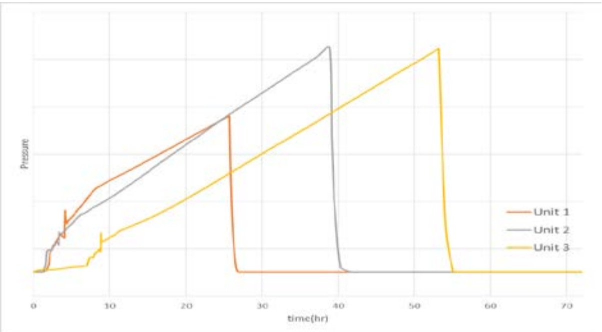

Fig. 2. Containment Pressures for the unit 1, 2 and 3

Fig. 3. shows the containment pressure behavior for the scenarios of the unit 4, 5 and 6. The pressures increased continuously until the end of calculation time without containment failure because the pressures unreached the leak and rupture points. The containment pressurization for the unit 6 was delayed because of the RCS depressurization leading to SITs injection.

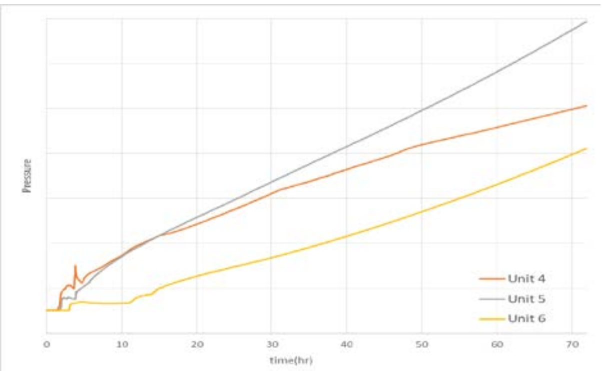

Fig. 3. Containment Pressures for the unit 4, 5 and 6

Table II shows the fractions of radionuclides released to the environment from the results. The unit 1, 2 and 3 where containment rupture occurred have larger fractions than that of the unit 4, 5 and 6 which considered design basis leakage rates. There are significant differences in the order of magnitude between the cases with and without the rupture.

Table II : The Release Fractions of Radionuclides to the Environment

Nuclide	Unit No.					
	1	2	3	4	5	6
Xe	9.98E-01	9.97E-01	9.96E-01	3.96E-03	2.47E-03	2.01E-03
Cs	1.84E-01	4.61E-01	3.84E-01	2.92E-05	6.86E-06	3.62E-06
Ba	1.23E-03	3.17E-03	2.66E-03	1.46E-05	8.22E-07	2.94E-06
I	2.95E-01	5.19E-01	3.97E-01	1.41E-04	8.76E-06	3.66E-06
Te	2.70E-01	4.82E-01	3.61E-01	1.10E-04	1.85E-06	2.19E-06
Ru	2.91E-05	1.37E-05	6.02E-06	4.75E-10	2.03E-08	1.38E-07
Mo	6.54E-01	4.02E-01	2.42E-01	8.23E-06	5.09E-07	1.07E-06
Ce	2.00E-05	9.58E-05	7.33E-05	4.95E-09	5.51E-09	6.19E-09
La	2.44E-05	4.39E-05	2.78E-05	2.61E-08	9.43E-10	2.81E-08

5. Conclusion

For dose assessment of multi-unit simultaneous accidents, we performed preliminary calculations using MELCOR according to the developed framework. Postulated accident scenarios without severe accident mitigation strategy were based on the seismic PSA report and CDF rank for various reactor type on the postulated site. Because containment failure resulting from pressurization occurred in the cases, the cases for unit 1 to 3 showed the larger release fraction compared with the cases for the unit 4 to 6. On the other hand, the design basis leakage rates were considered for the unit 4 to 6 cases since the containment pressure did not reach the failure point. These preliminary calculation results will be utilized for MACCS calculation within the framework for the dose assessment. And the final results can contribute to establishing a regulatory system for the deterministic analysis of multi-unit site in the future.

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[2] L.L. Humphries, MELCOR Computer Code Manual, SAND2015-6691, 2015.