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A Case Study of Multi-Unit Probabilistic Safety Assessment modeling for two units

Dohyun Lim, Wonjong Song, Byeongmun Ahn and Moosung Jae*
Department of Nuclear Engineering, Hanyang University, Seoul, 04763, Korea
**Corresponding author: jae@hanyang.ac.kr*

1. Introduction

After nuclear power plant accident in Japan, Korea have increased negative public perception and opinion of nuclear power plant operation. As a result, safety has been enhanced by Fukushima's follow-up measures to enhance safety, such as coastal barrier enlargement, mobile diesel generator installation and passive autocatalytic recombiner(PAR) installation. In addition, according to the Nuclear Safety Act enacted in 2014, the Incident Management Plan reinforces that the core damage frequency (CDF) and large early release frequency (LERF), which are measures of danger for new nuclear power plants, should be 1/10 of the existing value. But this is a standard for a single unit, and it is not legalized in the concept of site at present. The Fukushima accident has caused serious damage at unit 1~4, which caused a growing interest in the probabilistic safety assessment (PSA) for a site in the world. In the case of the Korea, the reactor is concentrated in one site. However, the methodology of PSA analysis for multi-unit has not been clearly defined, and has not been carried out, and is being studied worldwide.

In this study, the two units of the same type in a site were used as the reference reactor, and the loss of off-site power (LOOP) initial event was quantified to the multi-unit. The initial events for multi-unit were defined and valued, and the inter-unit common cause failure (CCF) values for emergency diesel generator (EDG) were considered. From the viewpoint of Level 3, the combination of source term categories (STCs) is multiplied by the number of STCs per unit, so that the number of STCs of multi-unit increases exponentially as the number of STCs of single unit increases. Therefore, STC grouping was performed to reduce the number of STC combinations. Finally, we used the MACCS program to find the consequences of the simplified STC combinations and combine them with the level 2 results to derive the risk from multi-unit.

2. Process

2.1 Common Cause Initiator

A reference plants of the same type reactor were designated as OPR1000 and each named Reference Plant 1,2. The initiating event frequency for the existing LOOP event is the value obtained for a single unit, and it is necessary to obtain the initial event frequency value for the multi-unit. The unit of core damage frequency (CDF) obtained in the single unit was number of accidents per reactor year and it was based on the number of events per unit of reactor, but in multi-unit, the concept of site year was introduced. This means that several events have occurred on the site, not based on one reactor. KINS assumes four cases for site year. Of the four cases, the site year was decided based on from the point of operation of the first nuclear power plant in the site, and the Shin Kori site and the Shin Wolsung site were considered as separate sites with Kori and Wolsung site, and the period until September 30, 2017 was set as the survey period. As a result, the site year of domestic nuclear power plants is 144.04 years.

Analysis of operating experiences showed that a total of 726 domestic nuclear power plant incidents occurred from 1978 to 2017 in operational performance information system (OPIS) data were analyzed. As a result, three cases were analyzed as a multi-unit loss of off-site Power (MULOOP). As a result of Bayesian update with MUDAP program applying Jefferey's Non-Informative Prior, the common cause initiator (CCI) of MULOOP was calculated as 2.42x10-2 / site Year.

Table Ⅰ MULOOP CCI Calculation Result

Type	**Number of Accidents**	**Site Year**	**IE Frequency (/Site-Year)**
MULOOP	3	144.04	2.42X10-2

2.2 Multi-Unit One-top model construction

Alternative AC diesel generator (AAC D/G) is considered as shared equipment in reference reactor for MULOOP initial event. AAC D/G connection is applied to the precedent unit with priority, and the model assumes that the trailing unit cannot connect to AAC D/G when preceding unit connect to AAC D/G. The inter-unit CCF was modeled using alpha parameters assuming that a total of four diesel generators (D/G) (two EDGs per unit) were in the same group (CCCG).

A multi-unit LOOP one-top model was developed using a single-unit LOOP model using the method proposed by KAERI and shown in Fig.1 [1].

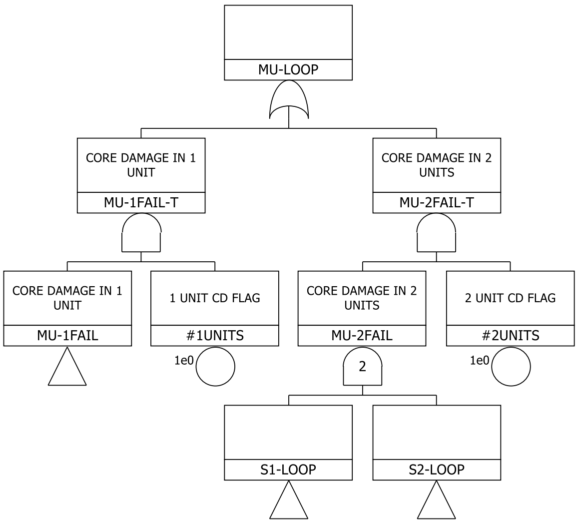

Fig.1. One-top model of the two reference plants

2.3 STC grouping for Multi-Unit Level 2 model construction

As mentioned previously, the number of STCs of multi-unit increases exponentially as the number of STCs of single unit increases in the PSA analysis of multi-unit, it increases exponentially. In Korea, since there are more than 6 nuclear power plants in each site, it is necessary to reduce the number of STCs by performing STC grouping. First, the alpha failure was removed. It was assumed that the branching to the heading of the cavities were filled with water to cool the debris and the containment spray system did not take into account and did not work. MELCOR analysis of the reference Nuclear Power Plant showed that the time and amount of Cs and I out of the containment building were similar in case of base melt through and late containment failure. As a result, a total of 8 STCs were derived.

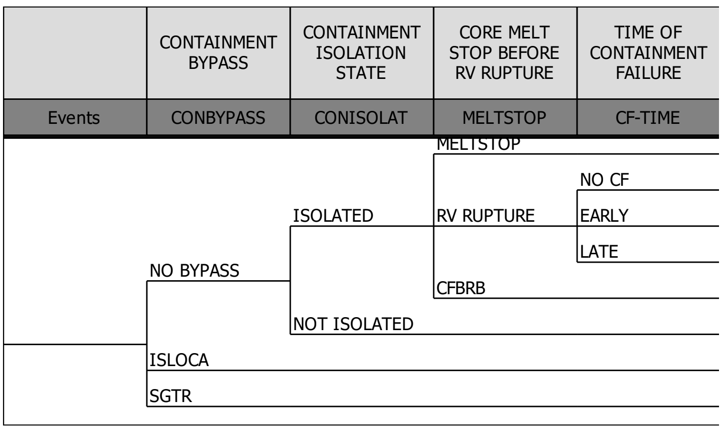

Fig. 2. Simplified STC logic diagram

Table Ⅱ: Simplified STC for multi-unit PSA modeling

STC	Type of Containment Building Failure
1	Core Melt Stop Before Containment Failure
2	Reactor Vessel Failure & Containment Intact
3	Early Containment Failure
4	Late Containment Failure & Basement Melt Through
5	Containment Failure Before Reactor Breach
6	Isolation Failure
7	Interfacing System LOCA
8	Steam Generator Tube Rupture

2.4 MACCS Result

In the case of STC 1, the core is damaged and the RCS system is cooled before reactor vessel failure so the reactor core is damaged, but the reactor vessel is intact. In the case of STC 2, the reactor vessel is damaged but the reactor building is not damaged. In both cases, containment building is intact and nuclide is rarely released to the outside of the containment building. The initial core inventory (kg) of the nuclear fission product of the reference nuclear power plant is Cs = 214.3kg and Cs-137 is 4 ~ 11kg. The PSA in the periodic safety review (PSR), which is a legal requirement, indicates that the total frequency of accidents where Cs-137 emissions exceed 100 TBq is less than 1.0E-6 / year, which is about 32g. The Cs release fraction of STC 1,2 was less than 1 g and not more than 100 TBq, so it was excluded from the L3 evaluation. In the consequence evaluation, STC 1 or 2 and other STC were combined, other STC was applied except for STC 1 or 2.

3. Results

Looking at the multi-unit one top model, the human error and cut set by EDG were large. In fact, if an accident occurs in several units in a time, the human error will be greater than when an accident occurs in a single unit. In this study, we did not consider the human error in multi-unit, but if we consider it, the CDF will increase.

The results obtained using the MACCS program were compared with the results of two units and sum of two single units. The results obtained in two units were evaluated to be larger than the sum of the results obtained in one unit. In this study, evacuation is not considered. It is expected that if the emergency response is considered, the value of consequence will be smaller in the results of two units.

The risk come from CCI was compared between the two units and sum of two single units. In the MACCS results, the results obtained from two units tended to be larger than those obtained from sum of single units, but the value of CDF was smaller in the two exposures. As mentioned earlier, this would be even larger given the untried inter-unit CCF, such as AAC, essential air conditioners, and containment building water pumps, and human error.

4. Conclusion

Only the AAC D/G precedence connection and inter-unit CCF for the EDG were considered. In reality, when an accident occurs in several units it is expected that people fall into panic and human error will be large.

In this model, emergency evacuation is not considered. But if emergency evacuation is considered, it is difficult to group the STC with the radioactive substances release time and quantity. If the time of the core damage and the time of the damage of the containment building is largely different even if the release time and amount are similar, the time for emergency evacuation will be longer and early heath effect will be less.

This method is based on various assumptions and is simply modeled to calculate the risk by CCI in two units so it contains many limitations. Therefore, it is difficult to determine that the methodology used above is a methodology that considers all the aspects of fully developed PSA modeling for multi-unit. the result would give insight to other researchers and be a foundation stone of the Fully developed multi-unit PSA.

Acknowledgements

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