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Seismic Probabilistic Safety Assessment along the Acceleration Level

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1. Introduction

To quantify the seismic risk of nuclear power plants, seismic probabilistic safety assessment (SPSA) was performed. The contribution along the acceleration level was estimated for the sequences inducing core damage. Sensitivity analysis was performed to verify important components for reducing the core damage frequency (CDF).

2. Evaluation of SPSA

2.1 Seismic Probabilistic Safety Analysis

SPSA was performed based on four steps, seismic hazard analysis, component fragility evaluation, accident sequence analysis, and consequence analysis [1]. For the modeling of a component fragility curve, a cumulative lognormal distribution was used as expressed in Equation (1).

$$F(a) = \Phi\left[\frac{\ln(a) - \ln(A_m) + \beta_U \Phi^{-1}(Q)}{\beta_R}\right] \quad (1)$$

where, Φ denotes standard Gaussian cumulative distribution function and A_m is a median ground acceleration capacity. β_R and β_U represent standard deviations of inherent randomness and uncertainty, respectively. The non-exceeding probability level of the median value, Q is introduced to consider the uncertainty in this equation. The system fragility curve is estimated by combining component fragilities following the accident sequence.

The result of a seismic probabilistic safety assessment is expressed as the frequency of adverse consequences, such as a CDF. The frequency of the damage is obtained by convolving plant level fragility with seismic hazard curves. This convolution is expressed by Equation (2).

$$P_F = \int_0^\infty F(a)\left(-\frac{dH(a)}{da}\right)da \quad (2)$$

where, *F(a)* is the system fragility at the given acceleration point and *H(a)* is a seismic hazard curve.

2.2 Analysis Model

For the analysis example of a nuclear power plant, the SPSA model was constructed by the event tree and fault tree for the core melt sequences. By the sequence analysis, the sequences leading to the CDF were determined, which were Loss of Essential Power (LEP), Loss of Component Cooling Water/Essential Chilled Water (LOCCW), Large LOCA (LLOCA), Small LOCA (SLOCA), Loss of Offsite Power (LOOP) and Seismic-Induced General Transient (GTRN) as shown in Fig 1. The LEP sequence was represented by the fault tree in Fig 2. The CDFs by the sequences of LOOP and GTRN were obtained by multiplying the probability of the secondary even tree, while the others were assumed to cause the core damage directly. The parameters of the component fragility in the event tree and fault tree are summarized in Table 1.

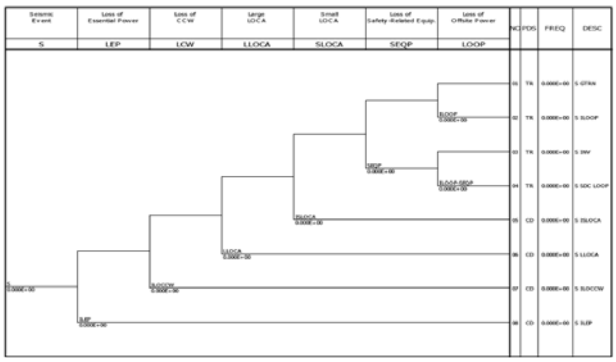

Fig. 1. Event tree for initiating events by a seismic accident.

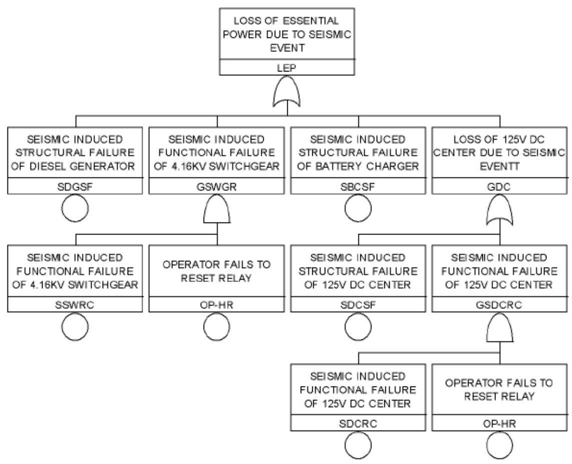

Fig. 2. Fault tree of the LEP sequence.

Table I: Component fragility parameters

Description	Title	A_m	β_R	β_U
Off-site Power	SLOOP	0.30	0.22	0.20
Diesel Generator	SDGSF	0.92	0.30	0.20
4.16kV SWGR	SSWRC	1.33	0.33	0.29
Battery Charger	SBCSF	1.35	0.29	0.31
Inverter	SINSF	1.43	0.29	0.30
480V Motor Control Center	SMCSF	1.48	0.34	0.30
125V DC Control Center (Structure)	SDCSF	1.12	0.29	0.30
125V DC Control Center (Function)	SDCRC	0.75	0.29	0.27
Instrumentation Tube (Primary)	SICPB	1.50	0.30	0.30
Safety Injection Tank	SITSF	1.09	0.36	0.35
CCW Pump	SCCWP	1.30	0.21	0.21

3. Result of Analysis

For the SPSA analysis, a computer code PRASSE was used to calculate the initiating event frequencies for seismic events. In this code, Latin Hypercube Sampling method were used to quantify the uncertainties of the system fragility [2].

A seismic hazard curve normalized at the 0.2g level was used to evaluate the contribution regarding acceleration level. The CDF was calculated for each sequences with the acceleration interval of 0.1g as shown in Fig. 3. The major sequence causing the core damage was the LEP with 79.3% portion. Although the capacity of offsite power was much smaller than others, the contribution of the LOOP was only 11.8% because its probability of secondary event tree was estimated to be 0.0123. But the LOOP caused high CDF at the low acceleration range around 0.3g. At the lowest acceleration level, the GTRN yielded the highest CDF because it was composed by the complementary event of other sequences.

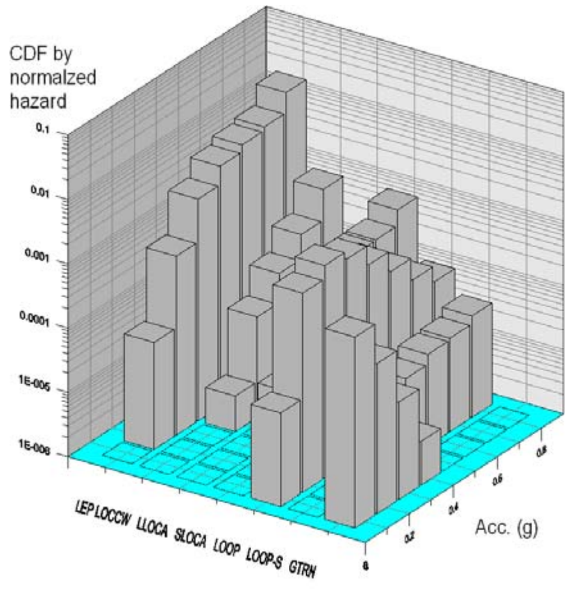

Fig. 3. Contribution to the CDF with regard to acceleration levels

Fig. 4 shows the decrease in CDF resulted by a 10% increase of the median capacity for each component. The capacity of the structural failure of diesel generator (SDGSF) was estimated to be the most influential failure mode by an earthquake event. The functional failure of 125V DC control center (SDCRC) which has the lowest capacity in the LEP sequence could not affect much because it could be recovered by an operator's action. The fragility component related with a random failure component could not affect the change of CDF. Except for the component in the LEP, the offsite power (SLOOP) was estimated to be the important component but not severe.

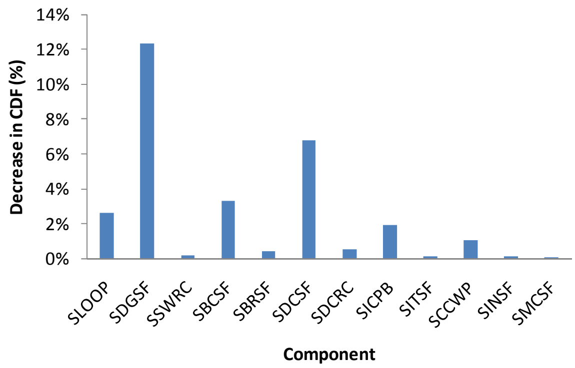

Fig. 4. Decrease in CDF by increasing median capacity of each components.

4. Conclusions

In this study, the probabilistic seismic evaluation was performed, and the importance of each sequences and components to the core damage was studied. The LEP was the main sequence to the CDF and its contribution increased as the acceleration level increased. Therefore maximum acceleration criteria for the convolution procedure should be carefully considered. In spite of its minor contribution to the CDF, the capacity of offsite power need to be improved to prevent the core damage by a low intensity earthquake.

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REFERENCES

[1] ANS-IEE-NRC, 1983. PRA Procedures Guide: A guide to the performance of probabilistic risk assessments for nuclear power plants, NUREG/CR-2300.

[2] Choi, I.K., Kim, J.H. and Park, J.H., 2010. Probabilistic Risk Assessment of NPP Systems for Earthquake Events. ESREL 2010, Rhodes, Greece.