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Original Article

Sensitivity analysis of failure correlation between structures, systems, and components on system risk

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A B S T R A C T

A seismic event caused an accident at the Fukushima Nuclear Power Plant, which further resulted in simultaneous accidents at several units. Consequently, this incident has aroused great interest in the safety of nuclear power plants worldwide. A reasonable safety evaluation of such an external event should appropriately consider the correlation between SSCs (structures, systems, and components) and the probability of failure. However, a probabilistic safety assessment in current nuclear industries is performed conservatively, assuming that the failure correlation between SSCs is independent or completely dependent. This is an extreme assumption; a reasonable risk can be calculated, or risk-based decision-making can be conducted only when the appropriate failure correlation between SSCs is considered. Thus, this study analyzed the effect of the failure correlation of SSCs on the safety of the system to realize rational safety assessment and decision-making. Consequently, the impact on the system differs according to the size of the failure probability of the SSCs and the AND and OR conditions.

1. Introduction

The Fukushima nuclear power plant accident in 2011 resulted in an increased interest in site safety and the safety of multiple units. The Fukushima nuclear power plant accident was a single seismic event that resulted in accidents in several units. In other words, one external event affected several units simultaneously. This implies the existence of a correlation between the failure probability of SSCs constituting a nuclear power plant, and the importance of probabilistic safety assessment of multiple units is emphasized [1,2]. In multi-unit probabilistic safety assessment, particularly in the case of external events, the correlation between the failure probability of SSCs is essential [3].

In general, the safety of nuclear power plants is evaluated by performing a probabilistic safety assessment. The correlation between the failure probabilities of the SSCs is conservatively assumed to be independent or completely dependent and the order of risk change as per the assumption considered [4,5]. Therefore, the assumption is unrealistic [6]. In general, the effects of natural disasters on external events affect all SSCs constituting a nuclear power plant rather than just one component of the nuclear power plant. In addition, the impact is not the same for all SSCs and varies according to the type and intensity of natural disasters. Therefore, a probabilistic safety evaluation should be conducted by considering the appropriate correlation for failure between SSCs [7].

Research on probabilistic safety assessment considering the failure correlation between SSCs for rational risk calculation and risk-based decision making is ongoing. The importance of the failure correlation of SSCs in the seismic probabilistic safety assessment of nuclear power plants was highlighted in the *Wash*-1400 [8] study. Thereafter, the “Seismic Safety Margins Research Program (SSMRP)” at Lawrence Livermore National Laboratory in the US conducted a survey on the risk due to failure correlation between SSCs [4]. Reed et al. [9] proposed a procedure for estimating the probability of simultaneous failure owing to correlation by identifying common factors that resulted in correlations in response and strength calculations to derive the seismic fragility of the system considering the correlation. Further, the author developed a method and program for quantifying system risk using MCS(Monte-Carlo Simulations) and direct integration methods [6,10].

Further, the failure correlation between SSCs must also be

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Table 1
Required failure correlation coefficient according to the number of SSCs constituting the system.

Number of SSCs	Number of failure correlation coefficients between SSCs required for PSA
2	1
5	10
10	45
50	1225
100	4990
1000	499,500

considered; that is, an appropriate correlation coefficient must be taken into account when performing a probabilistic safety assessment. Through probabilistic seismic response analysis [11], Ebisawa et al. proposed a method for considering the failure correlation between SSCs. They proposed a methodology to directly extract failure correlation coefficients between SSCs located in auxiliary buildings of standard Korean nuclear power plants from the response database [12]. In addition, using Ebisawa's methodology and SSI coherence theory, a method for deriving the seismic response correlation coefficient of failure correlation between SSCs in multiple units was proposed [2], and the risk of multiple units was evaluated successfully using the proposed method [13]. Segarra et al. performed a probabilistic safety assessment of the correlation between SSCs using a Bayesian network. Jung et al. proposed a method of using a CCF (common cause failure) that utilizes the failure correlation between SSCs in a probabilistic safety assessment. In addition, the author briefly examined the change in system fragility considering the failure correlation between the SSCs [6]. However, studies on the sensitivity of system risk to the failure correlation between SSCs are scarce.

This study analyzed the variations in the failure probability and system risk because of the failure correlation between SSCs under various conditions. The failure probability was confirmed for the AND and OR conditions by changing the size of the failure probability and the correlation coefficient. Further, for the above conditions, an essential power event loss was constituted, and the risk changes owing to independence and complete dependence were examined by grouping the SSCs constituting the loss of essential power events into pairs.

2. Variation in the failure probability due to failure correlation between SSCs

2.1. Failure correlation coefficient between SSCs

Various methods [4,9] take into account the failure correlation between SSCs of failure probability; however, the correlation coefficient is typically used. In the SSMRP, the failure correlation coefficient between the SSCs is calculated using equation (1).

$$\rho_{12}=\frac{\beta_{R1}\beta_{R2}}{\sqrt{\beta_{R1}^2+\beta_{C1}^2}\sqrt{\beta_{R2}^2+\beta_{C1}^2}}\rho_{R1}\rho_{R2}+\frac{\beta_{C1}\beta_{C2}}{\sqrt{\beta_{R1}^2+\beta_{C1}^2}\sqrt{\beta_{R2}^2+\beta_{C1}^2}}\rho_{C1}\rho_{C2} \quad (1)$$

where $\rho_{1,2}$ is the seismic failure correlation coefficient between the 1 and 2 SSCs to be obtained; ρ_{R1} and ρ_{R2} are the seismic response correlation coefficients between the SSCs; and ρ_{C1} and ρ_{C2} are the seismic capacity correlation coefficients between the SSCs. In addition, β_{R1} and β_{R2} are the standard deviations of the logarithmic standard normal distribution of the seismic response of SSCs 1 and 2, respectively, and β_{C1} and β_{C2} are the standard deviations of the logarithmic standard normal distribution of the seismic capacity of SSCs 1 and 2, respectively.

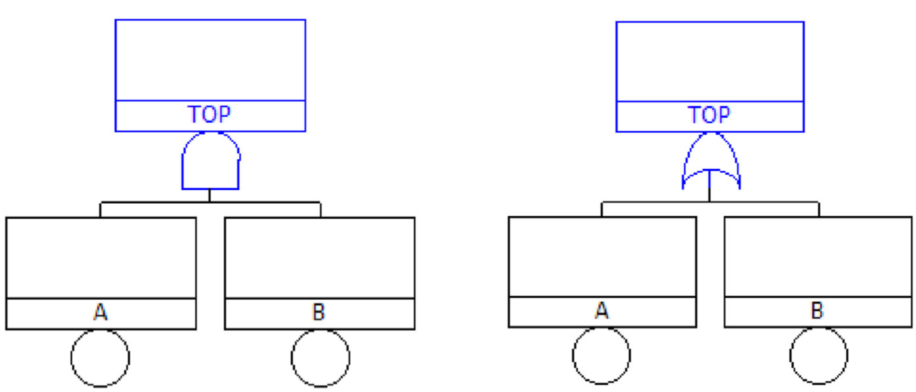
The number of necessary failure correlation coefficients is determined according to the number of SSCs configured for the probabilistic safety assessment of the system. With an increase in the number of SSCs, the required failure correlation coefficient increases significantly. For N number of SSCs used in the seismic probabilistic safety evaluation, the number of failure correlation coefficients required is $\frac{N^2-N}{2}$. Table 1 summarizes the necessary failure correlation coefficients based on the number of SSCs. However, determining the failure correlation coefficient between SSCs for the entire system constituting a nuclear power plant can be challenging.

2.2. Variation in failure probability according to failure correlation coefficient between SSCs

This section presents an analysis of the variation in the failure probability according to the AND and OR gates (conditions) based on the failure correlation coefficients with two events, A and B. For both events A and B, as shown in Fig. 1, the damage probability and the failure correlation coefficient were changed for the AND and OR conditions to confirm the failure probability of the simple system (AND gate and OR gate).

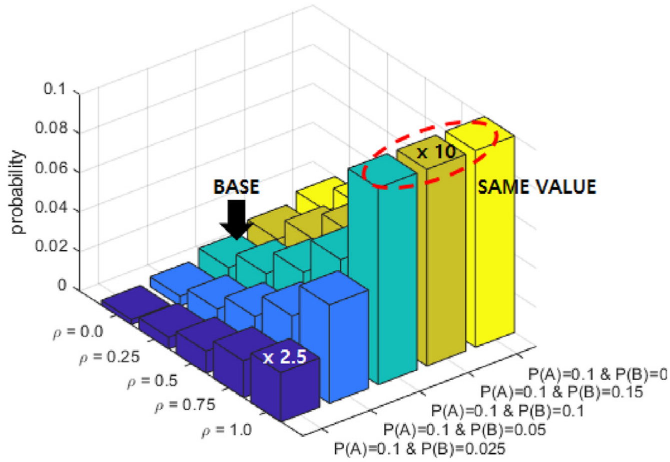
The failure probability of event A was fixed at 0.1, and that of event B was changed to 0.025, 0.05, 0.10, 0.15, and 0.20. Further, the failure correlation coefficients were changed to 0 (independent), 0.25, 0.5, 0.75, and 1.0 (completely dependent). Fig. 2 shows the change in the failure probability for different combinations of AND ($A \cap B$) and OR ($A \cup B$) conditions.

As shown in Fig. 2, in the AND condition, the failure probability increased with an increase in the failure correlation coefficient;

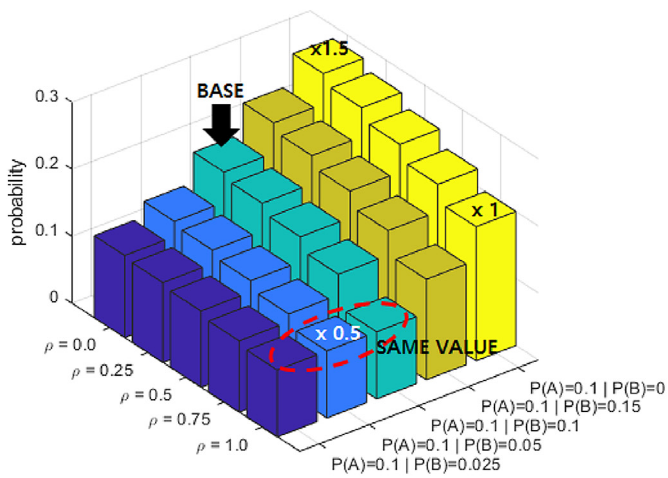


(a) AND gate (b) OR gate.

Fig. 1. Fault tree of system.



(a) AND gate

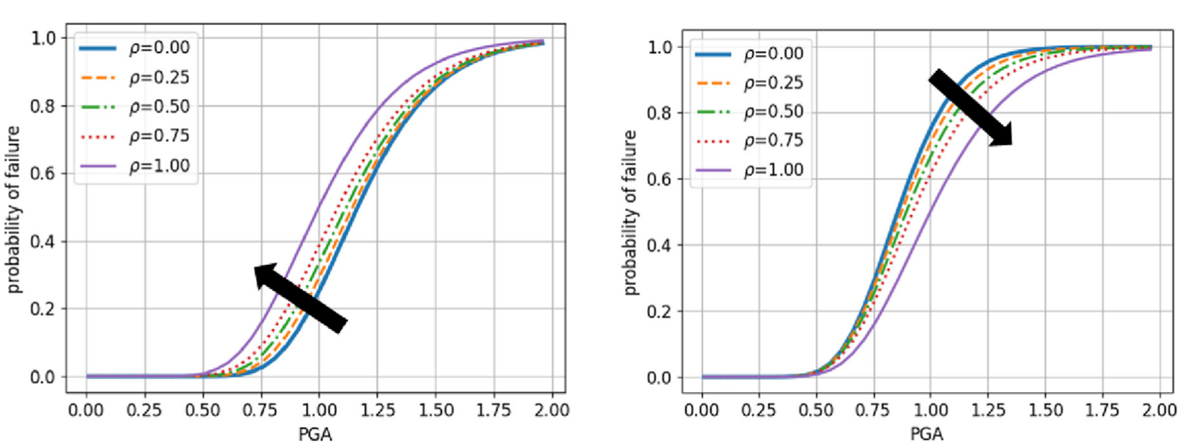


(b) OR gate

Fig. 2. Failure probability of AND system and OR system.

whereas, in the OR condition, it decreased. Further, in both conditions, the larger the failure correlation coefficient and failure probability, the more significant is the change in the system's (AND and OR conditions) failure probability. In AND condition, the change in the failure probability of the system according to the variation of the B value was large in the completely dependent condition, and in the OR condition, the change in the failure probability of the system according to the variation of the B value was large in the independent condition. This change was found to be larger in the AND gate than in the OR gate. Moreover, in the AND condition, when the failure probability of the system was 0.1 (maximum), the failure correlation coefficient was 1.0, and the failure probability of B was the same at 0.1, 0.15, and 0.2. Conversely, in the OR condition, when the correlation coefficient was 1.0 and the probabilities of B were 0.025, 0.05, and 0.1, the failure probability of the system was 0.1 (maximum). On comparison of the independent and fully dependent conditions based on the case wherein the failure probabilities of A and B were the same ($P(A) = 0.1$ and $P(B) = 0.1$), the AND condition increased 10 times ($P(B) = 0.2$), whereas the OR condition decreased by 50% ($P(B) = 0.025$) or increased by 50% ($P(B) = 0.2$). Under AND and fully dependent conditions, when the failure probability of B was 0.025, the failure probability of the system increased by 2.5 times (0.025). Further, for the failure probability of $P(B) = 0.2$, the failure probability was 0.1 and turned on 10 times. In contrast, under OR and fully dependent conditions, when the failure probability of B was 0.025, the failure probability decreased by 50%, and for the failure probability of $P(B) = 0.2$, the failure probability increased by 50%. Therefore, it is evident that the AND gate is more sensitive than the OR gate when considering the sensitivity of the failure correlation coefficient.

In the AND condition, the probability of failure was expected to decrease because of the failure correlation coefficient. Conversely, in the OR condition, the probability of failure was expected to decrease because of the failure correlation coefficient. This can also be confirmed from the fragility curve under the same conditions, and the results of the EEM's paper are shown in Fig. 3 [6]. The study on EEMs [6] used $A_m = 1.0$, $B_R = 0.2$, and $B_U = 0.2$ as the parameters for fragility for both A and B, and the fragility curves of the system were derived for the AND and OR conditions. In a similar manner, in the AND condition, with an increase in the failure correlation coefficient, the damage probability increased; whereas, in the OR condition, it decreased.



(a) AND gate (b) OR gate.

Fig. 3. Fragility curves for AND & OR gate [6].

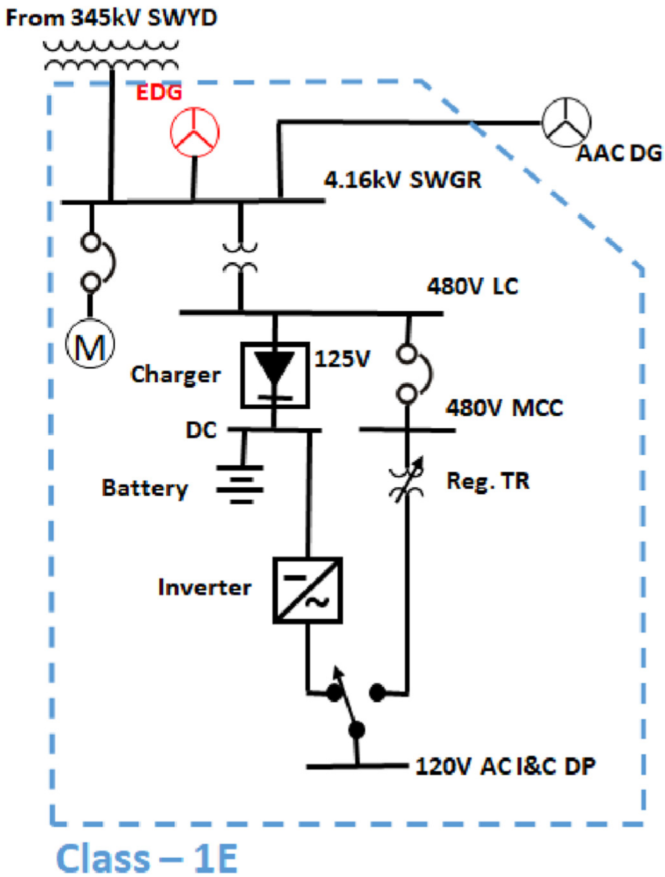


Fig. 4. Simplified power system in nuclear power plant (NPP) [7].

3. Changes in fragility and risk of loss of essential power events due to failure correlation between SSCs

This section presents the sensitivity analysis performed based on the failure correlation between SSCs for the loss of essential power events. When power is not supplied to the safety-related system of the nuclear power plant, and power is not even supplied from outside the nuclear power plant, which can cause damage to the core, this is referred to a loss of essential power incident. Conservatively, in this study, it was assumed that a loss of essential power immediately damaged the core [7]. The power supply system of a nuclear power plant is shown in Fig. 4, and the fault tree of the loss of an essential power event is shown in Fig. 5 [7]. The seismic fragilities of the SSCs related to the loss of essential power events were assumed, as presented in Table 2 [7].

3.1. *Sensitivity according to the failure correlation coefficient between SSCs by the seismic intensity*

A sensitivity analysis was performed considering the seismic intensity for the loss of essential power events with the failure correlation between SSCs. All components were assumed to be independent and completely dependent. The sensitivity analysis was performed using RAW (Risk achievement worth importance), RRW (Risk reduction worth importance), and FV (Fussell-Vesely's measure) analyses, which are widely used in probabilistic safety assessments. Fig. 6 shows the RAW, RRW, and FV results with respect to seismic intensity. Table 3 presents the RAW, RRW, and FV results for each SSC for 0.2 and 0.5 g.

As shown in Fig. 6, the values of RAW, RRW, and FV change according to the seismic intensity. The importance of SSCs increases or decreases according to the seismic intensity under the same independent or fully dependent conditions. This indicates that the order of importance of SSCs by seismic intensity may change because of the failure correlation between the SSCs. In addition, if

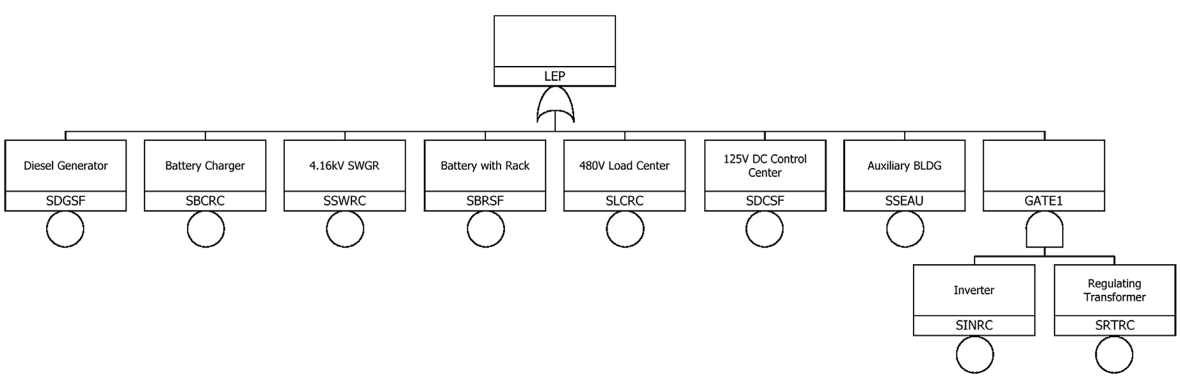


Fig. 5. Fault tree of seismic-induced loss of essential power [7].

Table 2
Configuration of seismic-induced loss of essential power event [7].

Structures or components	Median	β_R	β_U	Failure modes	Event codes
Diesel generator	1.243	0.396	0.330	structural failure	SDGSF
Battery charger	1.133	0.308	0.308	Functional failure	SBCRC
4.16 kV SWGR	1.463	0.363	0.319	Functional failure	SSWRC
Battery rack	1.606	0.363	0.341	Structural failure	SBRSF
480 V Load Center	1.650	0.352	0.319	Functional failure	SLCRC
125 V DC Control Center	1.738	0.363	0.319	Structural failure	SDCSF
Auxiliary building	2.200	0.352	0.407	Structural failure	SSEAU
inverter	1.507	0.363	0.330	Functional failure	SINRC
Regulating transformer	1.430	0.363	0.330	Functional failure	SRTRC

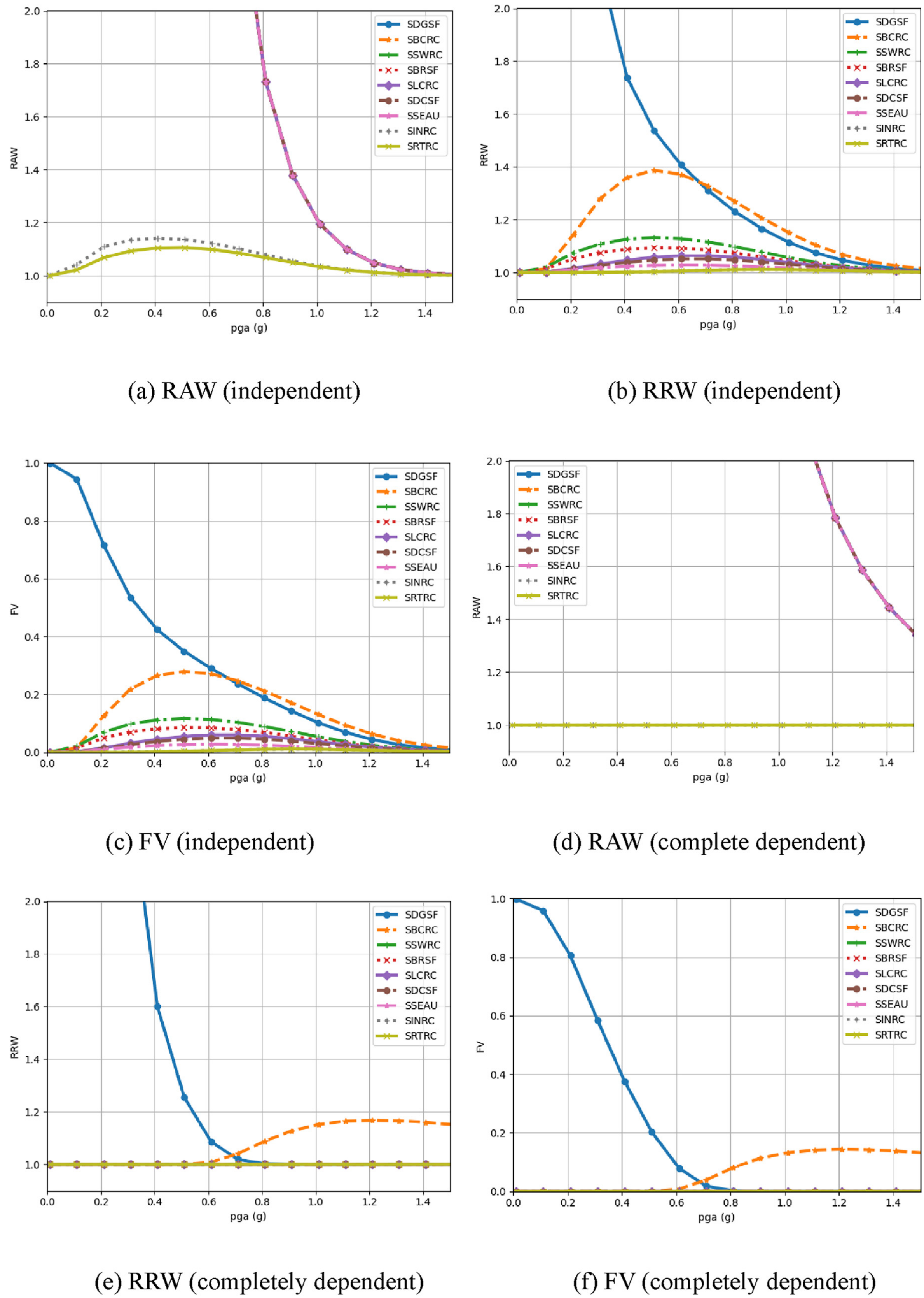


Fig. 6. Importance of SSCs based on Seismic Intensity.

Table 3
Importance of SSCs for Seismic Intensity of 0.2 and 0.5 g

	SSCs	Independent 0.2 g	Complete dependent 0.2 g	Change ratio 0.2 g	Independent 0.5 g	Complete dependent 0.5 g	Change ratio 0.5 g
RAW	SDGSF	100674.08	81603.45	0.19	10.81	27.80	1.57
	SBCRC	100674.08	81603.45	0.19	10.81	27.80	1.57
	SSWRC	100674.08	81603.45	0.19	10.81	27.80	1.57
	SBRSF	100674.08	81603.45	0.19	10.81	27.80	1.57
	SLCRC	100674.08	81603.45	0.19	10.81	27.80	1.57
	SDCSF	100674.08	81603.45	0.19	10.81	27.80	1.57
	SSEAU	100674.08	81603.45	0.19	10.81	27.80	1.57
	SINRC	1.10	1.00	0.09	1.14	1.00	0.12
	SRTRC	1.06	1.00	0.06	1.11	1.00	0.10
RRW	SDGSF	5.01	7.09	0.42	1.56	1.29	0.17
	SBCRC	1.13	1.00	0.12	1.38	1.00	0.28
	SSWRC	1.07	1.00	0.06	1.13	1.00	0.12
	SBRSF	1.05	1.00	0.05	1.09	1.00	0.09
	SLCRC	1.01	1.00	0.01	1.06	1.00	0.05
	SDCSF	1.01	1.00	0.01	1.05	1.00	0.04
	SSEAU	1.01	1.00	0.01	1.03	1.00	0.03
	SINRC	1.00	1.00	0.00	1.00	1.00	0.00
	SRTRC	1.00	1.00	0.00	1.00	1.00	0.00
FV	SDGSF	0.74	0.82	0.11	0.36	0.22	0.38
	SBCRC	0.11	0.00	1.00	0.28	0.00	1.00
	SSWRC	0.64	0.00	1.00	0.12	0.00	1.00
	SBRSF	0.05	0.00	1.00	0.09	0.00	1.00
	SLCRC	0.01	0.00	1.00	0.05	0.00	1.00
	SDCSF	0.01	0.00	1.00	0.04	0.00	1.00
	SSEAU	0.01	0.00	1.00	0.03	0.00	1.00
	SINRC	0.00	0.00	1.00	0.00	0.00	–
	SRTRC	0.00	0.00	1.00	0.00	0.00	–

the failure correlation between the SSCs by seismic intensity is different, an appropriate failure correlation to the seismic intensity should be applied. However, it may be inefficient to consider the failure correlation between SSCs based on seismic intensity. Therefore, similar to the method of selecting the intensity of the reference earthquake in the seismic probabilistic safety assessment, it is expected that the effective application of the failure correlation between SSCs to the intensity significantly affects the seismic probabilistic safety assessment result. However, this requires clarification through further research.

3.2. Changes in system risk according to the failure correlation coefficient between SSCs

The seismic fragility and risk for a pair of SSCs were calculated under the assumption of independent and complete dependence. The seismic hazard curves shown in Fig. 7 were used to calculate the risk of the loss of essential power [14]. Further, the average seismic hazard curve was employed considering each probability for the six seismic hazard curves.

Fig. 8 shows the seismic fragility curves related to the loss of essential power when all SSCs are independent and when each pair

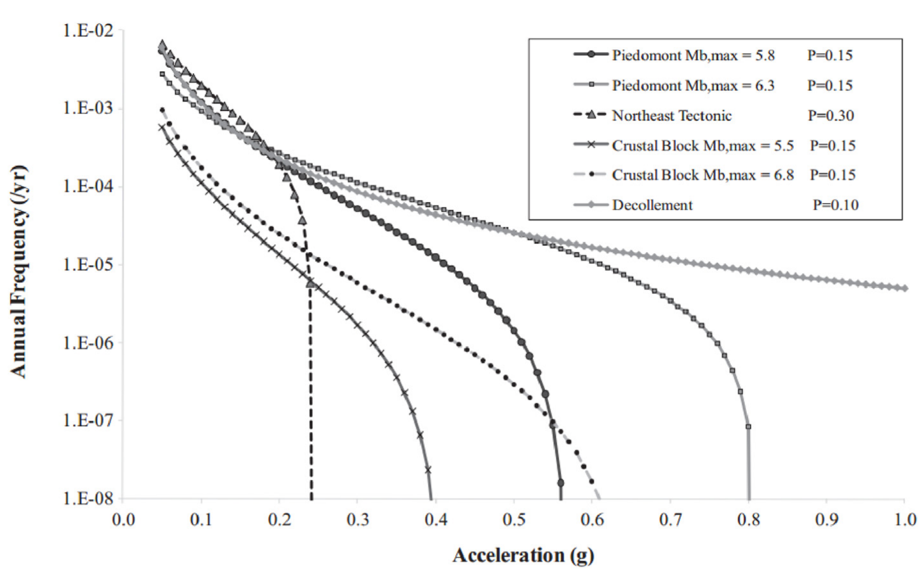


Fig. 7. Seismic hazard curves [14].

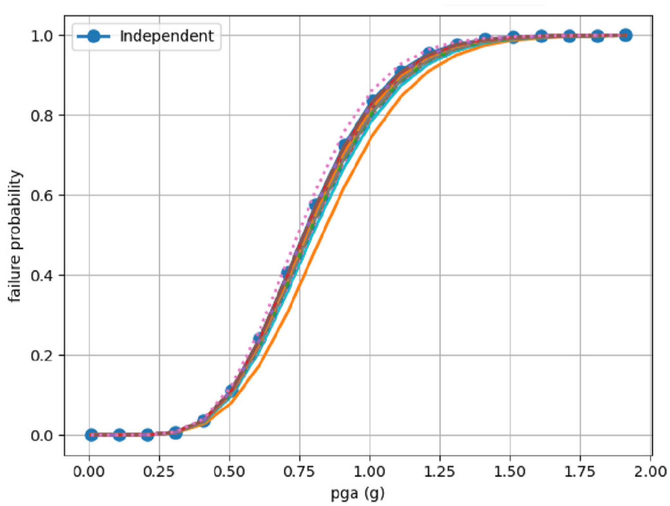


Fig. 8. Seismic fragility curves of loss of essential power.

is completely dependent. The blue line (thick and with marker) represents the result when all SSCs are independent, whereas the others represent the completely dependent cases of pairs of SSCs. The seismic fragility changed when each pair was completely dependent. Table 4 presents the rate of change of the risk value based on the independent case, when each pair was completely dependent. For the essential power loss event, there were 8 cases wherein the risk was increased by failure correlation and 28 cases

Table 4
Ratio of change of risk due to the failure correlation.

	Risk (/yr)	Ratio
Independent cases	2.59E-06	–
SDGSF & SBCRC	1.98e-6	−0.2363
SDGSF & SSWRC	2.32E-06	−0.1036
SBCRC & SSWRC	2.39E-06	−0.1036
SINRC & SRTRC	2.47E-06	0.0951
SDGSF & SBRSF	2.49E-06	−0.0749
SBCRC & SBRSF	2.53E-06	−0.0749
SSWRC & SBRSF	2.58E-06	−0.0749
SDGSF & SLCRC	2.58E-06	−0.0471
SBCRC & SLCRC	2.32E-06	−0.0471
SSWRC & SLCRC	2.39E-06	−0.0471
SBRSF & SLCRC	2.47E-06	−0.0471
SDGSF & SDCSF	2.49E-06	−0.0388
SBCRC & SDCSF	2.53E-06	−0.0388
SSWRC & SDCSF	2.58E-06	−0.0388
SBRSF & SDCSF	2.58E-06	−0.0388
SLCRC & SDCSF	2.39E-06	−0.0388
SDGSF & SSEAU	2.47E-06	−0.0218
SBCRC & SSEAU	2.49E-06	−0.0218
SSWRC & SSEAU	2.53E-06	−0.0218
SBRSF & SSEAU	2.58E-06	−0.0218
SLCRC & SSEAU	2.59E-06	−0.0218
SDCSF & SSEAU	2.47E-06	−0.0218
SSEAU & SRTRC	2.49E-06	0.002
SSEAU & SINRC	2.53E-06	0.0018
SDCSF & SRTRC	2.59E-06	0.0013
SDGSF & SINRC	2.59E-06	−0.0011
SDGSF & SRTRC	2.49E-06	−0.0011
SBCRC & SINRC	2.53E-06	−0.0011
SBCRC & SRTRC	2.59E-06	−0.0011
SSWRC & SINRC	2.59E-06	−0.0011
SLCRC & SRTRC	2.53E-06	0.001
SDCSF & SINRC	2.59E-06	0.0009
SLCRC & SINRC	2.59E-06	0.0006
SSWRC & SRTRC	2.59E-06	−0.0005
SBRSF & SINRC	2.59E-06	−0.0003
SBRSF & SRTRC	2.83E-06	0.0003

wherein the risk was decreased by failure correlation. It was evident that the loss of essential power was because the OR gate dominated. Furthermore, the cases where the risk rises can be confirmed in the case of the combination of AND gates were SINRC and SRTRC.

In the case of a combination of SSCs with a large FV value based on FV, the degree of change in risk was large. In the case of SSEAU and SRTRC, both of which exhibit small failure probability values, it was a combination with AND; however, the risk increased. In the case of the SINRC and SRTRC combination, which is the minimum cut set, the AND combination increased the damage probability and increased the change rate to the greatest extent. Furthermore, in the case of SDGSF and SINRC, the influence of SDGSF was dominant. Therefore, in contrast to the SSEAU and SRTRC combinations, the failure probability decreased; however, the change rate was small. Thus, the SINRC and SRTRC combination via AND exhibited a small damage probability and importance compared to other SSCs, and therefore had a minimal effect on the entire system.

4. Conclusions

A probabilistic safety assessment is conducted for the evaluation of the safety of nuclear power plants. In particular, in the case of an external event (natural hazard), it affects all the SSCs constituting the nuclear power plant. Therefore, there is a correlation between the failure probability of SSCs and research that considers this effect is in progress. In multi-unit issues, there are many failure correlations between SSCs that must be considered. In this study, the failure probability and risk change of the system were analyzed using the failure correlation between SSCs. In the AND condition, with an increase in the failure correlation, the failure probability increased, whereas, in the OR condition, it decreased. Further, the change in failure probability in each condition was larger in the AND condition than in the OR condition. In addition, it was confirmed that the greater the failure probability and failure correlation coefficient, the greater the change in the failure probability of the system. Moreover, the change in risk owing to the failure correlation of each pair of SSCs was confirmed based on the loss of essential power events. It is confirmed that the risk of a loss during the essential power events owing to the failure correlation of each pair of SSCs is generally reduced. This is because the OR condition was primarily used for the loss of essential power events. Further, it was found that the amount of risk change was large in the case of a combination of devices with high-importance SSCs. The discussion presented in this study was elaborated through this example; therefore, this must be generalized through investigations of further cases in the future. In addition, considering that the failure correlation of all SSCs constituting events and all hazard intensities is expected to be inefficient, an efficient methodology is required.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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