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DEVELOPMENT OF EVALUATION METHOD FOR SEISMIC ISOLATION SYSTEMS OF NUCLEAR POWER FACILITIES -SEISMIC PRA FOR THE SEISMIC ISOLATED NUCLEAR PLANT-

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ABSTRACT

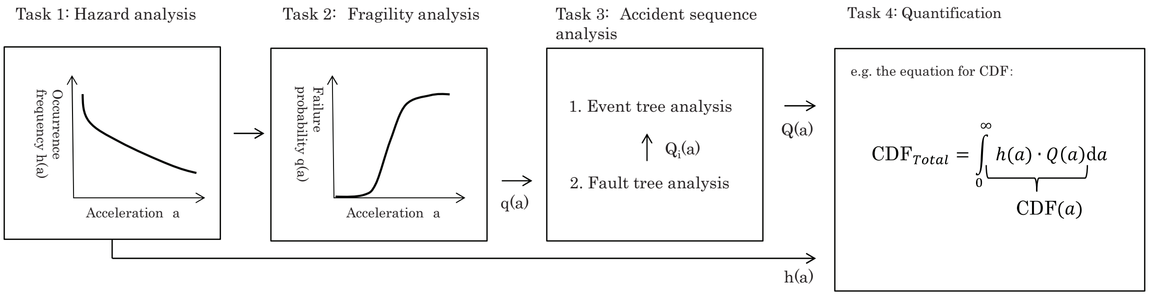
This study is part of a series titled “Development of an evaluation method for seismic isolation systems of nuclear power facilities”. In this study, a seismic level 1 probabilistic risk assessment (PRA) was performed for a nuclear power plant equipped with a seismic isolation system in order to evaluate its effects under the design basis and also design extension conditions. The study also clarifies the risk feature of the seismically isolated plant and suggests ways to improve the plant reliability by comparing the results of PRA with those of a conventional nuclear power plant.

INTRODUCTION

The study series titled “Development of an evaluation method for seismic isolation systems of nuclear power facilities” has developed methods to estimate the performance characteristics of the isolation system and crossover pipes between an isolated building and a non-isolated building, and the effects on the plant. In addition, risk studies have been conducted for facilities, such as fragility analysis for the important components for safe shutdown of the plant. However, a study on the total risk for the plant has not yet been performed. Therefore, this study carries out a seismic level 1 PRA to quantify the effects of installing the seismic isolation system in the nuclear power plant and identifies the weaknesses by calculating the core damage frequency (CDF) and Fussell Vesely (FV) importance.

METHOD

The evaluation process is shown in Figure 1. The process is divided into four tasks: hazard analysis, fragility analysis, accident sequence analysis and quantification. This paper focuses on tasks 3 and 4; the details of the others are discussed in other papers in this study series [1].



h(a): seismic frequency of a seismic acceleration a
Qi(a): failure probability of system i at a seismic acceleration a
Q(a): conditional core damage probability at a seismic acceleration a
CDFtotal: core damage frequency over all acceleration

Figure 1. Evaluation process of level 1 seismic PRA

Task 1: Hazard analysis

In this task, the seismic hazard curve, which is the annual probability of exceedance of the seismic acceleration a, is evaluated, and then used to calculate the seismic frequency, h(a).

Task 2: Fragility analysis

In this task, the responses of the building and equipment to earthquake motion are evaluated. From this, the conditional failure probability of the building or equipment at each seismic acceleration (fragility curve) is calculated as q(a).

Task 3: Accident sequence analysis

This task includes the following two subtasks.

Subtask 1: Event tree analysis

Event trees are developed to identify accident sequences leading to core damage. Also, Q(a), which is the conditional core damage probability of the plant at a seismic acceleration a, is calculated by using the inputs obtained through the next subtask 2.

Subtask 2: Fault tree analysis

Fault trees are developed for a system i whose failure affects the plant operation, or for a system i for plant safe shutdown to calculate its failure probability Qi(a) at acceleration a with the fragility data q(a) evaluated in task 2.

Task 4: Quantification

To obtain CDF at a seismic acceleration a, CDF(a), Q(a) is multiplied by seismic frequency, h(a), and CDF(a) are integrated over all seismic accelerations:

$$CDF(a) = h(a) \cdot Q(a) \tag{1}$$

$$CDF_{total} = \int_0^\infty CDF(a)\, \mathrm{da} \tag{2}$$

where,
h(a): seismic frequency at acceleration a
Q(a): conditional core damage probability at acceleration a
CDF(a): core damage frequency at acceleration a
CDF_{total}: core damage frequency over all accelerations

Also, FV importance is evaluated by:

$$FV = {CDF_{total,i}}/{CDF_{total}} \tag{3}$$

where,
$CDF_{total,\,i}$: core damage frequency due to failure of component i
CDF_{total}: total core damage frequency

PRA MODEL

The event trees and fault trees for PWR and BWR plants are developed based on the following assumptions:

- Only the reactor building is seismically isolated.
- Even though some of the seismic isolation devices composing the isolation system are destroyed partly or completely by an earthquake, they are all considered to be unreliable (perfect correlation).
- The influence of failure of the reactor building itself is ignored since its failure probability is much lower than that of the seismic isolation system.
- Random failures of mitigation systems are ignored.
- Failure of any critical components for which fragility analysis is performed leads to core damage.

For example, the developed event tree and fault tree for failure of the reactor building or structures for the seismically isolated BWR plant are shown in Figures 2 and 3 respectively. In this case, eight sequences are recognized as accident sequences.

sequence 3: In this sequence, failure of control of core reactivity occurs during loss of offsite power (LOSP*) since the systems for scram function or components inside RPV whose destruction results in interrupting scram function are seismically destroyed even though other structures such as seismic isolation system, reactor building, primary containment vessel (PCV) and reactor pressurized vessel (RPV) survive, and onsite power is available.

*: If LOSP is not caused by an earthquake, other critical components for safe shutdown must survive such earthquake since offsite power system is one of the seismically weakest systems. Under such condition, core damage should not occur.

sequence 4: In this sequence, station black out (SBO) occurs since emergency onsite AC power systems lose their functions because of such earthquake during LOSP. And then core damage occurs as a result of failure to remove residual heat in core because of loss of residual heat removal (RHR) system run by AC power despite success in controlling core reactivity.

sequence 5: In this sequence, failure to control core reactivity occurs during SBO since the AC power systems lose their functions because of an earthquake, and additionally, scram simultaneously fails because of loss of the scram system or interruption caused by destruction of the structures inside RPV.

sequence 6: In this sequence, loss of onsite DC power occurs because of an earthquake, which results in loss of almost all mitigation systems and then core damage.

sequence 7: In this sequence, loss of circuit or instruments for mitigation systems is caused by an earthquake, which results in loss of almost all mitigation systems and then core damage.

sequence 8: In this sequence, such loss of coolant (LOCA) occurs that it is impossible to mitigate because of destruction of pipes connecting to RPV in PCV by an earthquake, which results in core damage.

sequence 9: In this sequence, destruction of the reactor building or important structures for the plant such as PCV, RPV and suppression pool occurs because of an earthquake, which results in LOCA and simultaneously loss of mitigation systems, and then core damage occurs.

sequence 10: In this sequence, destruction of the seismic isolation system occurs because of an earthquake, which results in destruction of other structures on it such as the reactor building and PCV, and finally leads to core damage. This sequence is specific to the seismically isolated plant, and this is the only difference between the conventional plant and the seismically isolated plant.

On the other hand, in sequences 1 and 2, core damage should not occur since many mitigation systems are available.

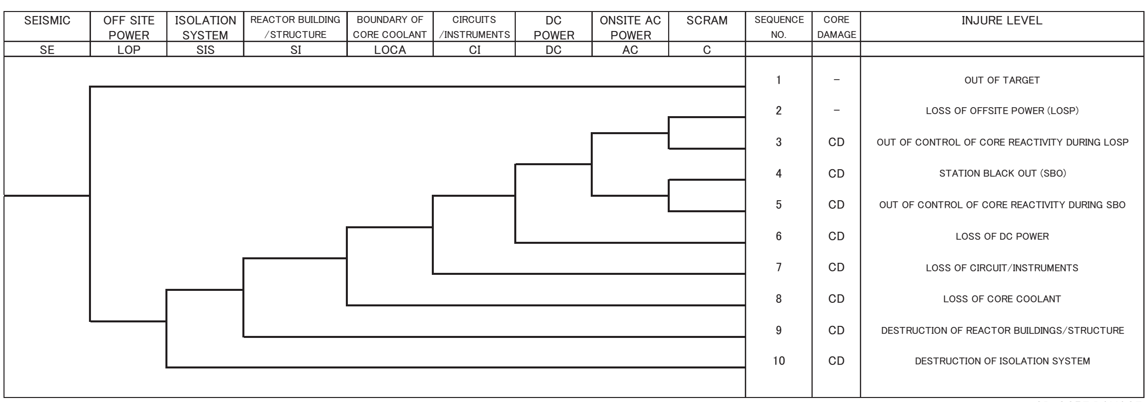


Figure 2. Example of the event tree (BWR plant)

The fault tree in Figure 3 is for failure of the reactor building or critical structures composing the plant such as PCV and RPV. It is conservatively assumed that destruction of each structure due to an earthquake results in core damage since actual effects on the plant have not been clarified sufficiently.

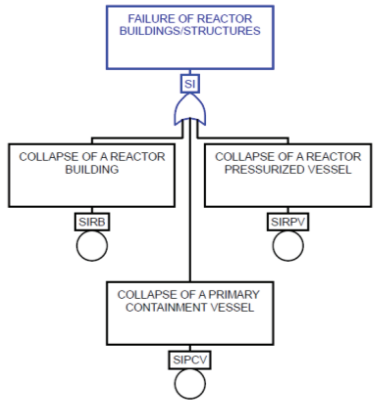


Figure 3. Example of the fault tree (BWR plant)

Inputs for quantification are summarized below.

Seismic hazard

Typical hazard data at a certain nuclear power plant site in Japan are used (shown in Figure 4).

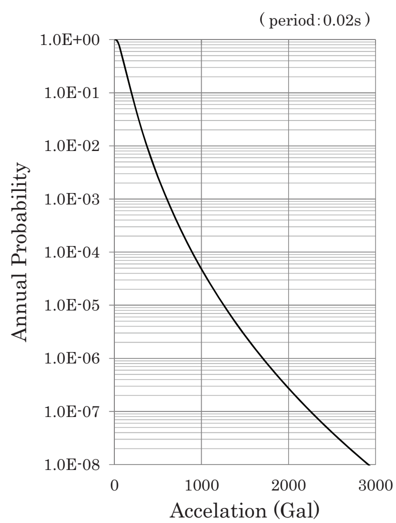


Figure 4. Seismic hazard curve

Fragility

The fragility data of the most critical components for core damage estimated in another paper [1] are used. The failure probabilities for other components are ignored. Additionally, the possibility of destruction of the reactor building is not considered. The fragility data are shown in Tables 1 and 2 for the PWR plant and the BWR plant, respectively.

Table 1: Fragilities for components of PWR plant

Component	Seismically isolated plant (Am^{*1}, βr^{*2}, βu^{*3}, $HCLPF^{*4}$)
Component A	1.80, 0.15, 0.15, 1.11
Component B	2.60, 0.13, 0.13, 1.67
Component C	1.93, 0.10, 0.10, 1.41
Auxiliary component A	2.12, 0.07, 0.07, 1.67
Auxiliary component B	1.94, 0.07, 0.07, 1.55
Electric board A	2.47, 0.14, 0.23, 1.34
Electric board B	1.67, 0.07, 0.15, 1.16
Electric board C	2.26, 0.14, 0.23, 1.22
Electric board D	2.26, 0.14, 0.23, 1.22
Main steam pipe (crossover pipe)	5.26, 0.134^{*5}, 3.84
Cooling seawater pipe (crossover pipe)	8.83, 0.203^{*5}, 5.50
Seismic isolation system	1.81, 0.075^{*5}, 1.52

*1: acceleration in G corresponding to median failure probability with 50% confidence level
*2: logarithmic standard deviation for aleatory uncertainty
*3: logarithmic standard deviation for epistemic uncertainty
*4: acceleration in G corresponding to low failure probability (5%) with high confidence (95%) level
*5: combined logarithmic standard deviation, $\beta_c = \sqrt{\beta_r + \beta_u}$

Table 2: Fragilities for components of BWR plant

Component	Seismically isolated plant (Am^{*1}, βr^{*2}, βu^{*3}, $HCLPF^{*4}$)
Cable tray	4.73, 0.31, 0.31, 1.70
Pipe A	4.37, 0.27, 0.27, 1.79
Pipe B	5.89, 0.32, 0.32, 2.05
Component A	3.23, 0.15, 0.15, 1.97
Component B	4.12, 0.19, 0.19, 2.20
Component C	6.11, 0.17, 0.17, 3.49
Component D	2.37, 0.11, 0.11, 1.65
Component E	6.89, 0.25, 0.25, 3.02
Component F	3.54, 0.15, 0.15, 2.16
Component G	2.61, 0.12, 0.12, 1.76
Crossover pipe	3.38, 0.33^{*5}, 1.57
Seismic isolation system	1.81, 0.075^{*5}, 1.52

*1~5: see Table 1 notes

RESULTS and DISCUSSION

The results of the PRA for PWR plant and BWR plant are shown in Tables 3, 4 and 5.

Results for PWR plant

CDF is estimated as 2.9E-6 per reactor year which is comparable to a conventional PWR plant in Japan whose CDF is around 2.8E-6 per reactor year despite using severer hazard data than that of the aseismic PWR plant site. This is because the fragilities of the components of the seismically isolated plant are improved, thereby decreasing their failure probabilities. According to FV importance, the event that has the highest contribution to CDF is the failure of Component A having the lowest seismic capacity. The failure of a component causing a large loss of coolant accident (LOCA) exceeding the mitigating capability of the emergency core cooling systems (ECCSs) results in the release of radioactive substances. The event with the second-highest contribution to CDF is the failure of the seismic isolation system causing an accident sequence specific to the seismically isolated plant.

Additionally, sensitivity analysis is performed, taking into consideration the design improvement of Component A against vertical vibration that is amplified by the isolation system and so is a crucial factor for the fragility of main Component A. The results are shown in Table 2. CDF changes to 1.4E-6 per plant year. Besides, the failure of the seismic isolation system has the highest contribution to CDF.

Table 3: Analysis results for PWR plant

CDF(per plant year)	2.9E-6		
dominant sequence	A large LOCA caused by failure of Component A results in the release of radioactive substances.		
FV importance	Rank	Object	Value
	1	The component A	0.64
	2	The seismic isolation system	0.12
	3 & under	The others	less than 0.03

Table 4: Sensitivity analysis results for PWR plant

CDF(per plant year)	1.4E-6		
dominant sequence	The failure of the seismic isolation system leading to destruction of the reactor building and loss of the critical components of the seismically isolated plant.		
FV importance	Rank	Object	Value
	1	The seismic isolation system	0.32
	2	The component A	0.22
	3 & under	The others	less than 0.08

Results for BWR plant

CDF is estimated as 1.0E-6 per reactor year, which is less than half that of a conventional BWR plant in Japan, 2.4E-6 per reactor year, evaluated using the same hazard data because fragilities of almost all components have improved, and the worst one has changed to around 1.5 G in high confidence low probability of failure (HCLPF) of the seismic isolation system for the seismically isolated plant from around 1.3 G of the weakest component for the conventional plant. The event with the highest contribution to CDF is the one caused by failure of the seismic isolation system, which also has the highest FV importance since its fragility is the worst in the seismically isolated plant as mentioned before. The component having the second-highest FV importance is the seawater pipe, which is a typical crossover pipe between an isolated building and a non-isolated building, whose FV importance is around 0.07, about ten times less than that of the seismic isolation system even though their fragilities do not vary much. This difference is derived from the extent of the impact on the plant when they are crushed. If the seismic isolation system is destroyed, constructions on it and also all mitigation systems could be

damaged simultaneously, while breaking of the seawater pipe might cause only SBO with loss of function of some mitigation systems activated by AC power.
The above results indicate characteristics similar to those indicated by the sensitive analysis result of the PWR plant.

Table 5: Analysis results for BWR plant

CDF(per plant year)	1.0E-6		
dominant sequence	The failure of the seismic isolation system leading to destruction of the reactor building and loss of the critical components of the seismically isolated plant.		
FV importance	Rank	Object	Value
	1	The seismic isolation system	0.54
	2	The crossover pipe	0.069
	3	The cable tray	0.065
	4 & under	The others	less than 0.03

CONCLUSION

The results of the PRA for PWR and BWR plants show that installation of the seismic isolation system for nuclear plants improves their reliability. However, the unique accident sequence caused by failure of the seismic isolation system has the highest contribution to CDF for both types of plants. Therefore, in order to improve plant reliability, it is necessary to develop a fail-safe seismic isolation system or to perform fragility analysis for the system itself in greater detail.

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REFERENCES

[1]Haraguchi, R. *et.al*. (2015). “Development of an evaluation method for seismic isolation systems of nuclear power facilities - Seismic fragility evaluation of equipment -”, SMiRT23, Manchester
[2]Atomic Energy Society of Japan. (2007). “Implementation Standard Concerning the Seismic Probabilistic Risk Assessment of Nuclear Power Plants: 2007 (AESJ-SC-P006: 2007)”, AESJ, Japan