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IMPROVEMENT OF SEISMIC SAFETY OF NUCLEAR POWER PLANTS BY INCREASE OF EQUIPMENT SEISMIC CAPACITY

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SUMMARY

The effects of the seismic capacity of nuclear facilities on the safety of nuclear power plants are investigated by the evaluation of the relation between the seismic capacity of the equipment and the core damage frequency (CDF). A case study is carried out for the Yonggwang Nuclear Units 5&6, which are operating pressurized water reactors in Korea. For the seismic evaluation, the equipment important for a CDF are selected and a probabilistic safety assessment (PSA) for a seismic event is performed using various seismic capacities. It is demonstrated from the results that the increase of the seismic capacity of the equipment can reduce the CDF significantly. This means that the seismic safety of the operating nuclear power plants can be significantly improved by increasing the equipment seismic capacity. For improving the seismic safety of the Yonggwang Nuclear Units 5&6, the increases of the seismic capacities of an Offsite Power and a Diesel Generator are the most effective in the PGA range of 0.3g to 0.5g. In the case of an Offsite Power, at 0.4g, an increase of its seismic capacity of 25% and 50% leads to a reduction of 33% and 45% in the CDF, respectively.

INTRODUCTION

A nuclear power plant is designed to ensure the survival of all the buildings and emergency safety systems during a design basis earthquake. Therefore, a seismic analysis for the nuclear power plant must consider all the interrelated factors that determine the release of radioactive material to the public. During an earthquake, since all the facilities of the plant are excited simultaneously, there may be a significant correlation between the component failures. Accordingly, the redundancy of safety systems in the nuclear power plant is very important. Therefore, all of the safety-related structures, systems and components in the nuclear power plant should be designed to have a sufficient seismic capacity. However, even though the facilities in the plant are designed to be safe during a design basis earthquake, they may be damaged or failed by strong ground motions greater than the design basis earthquake as well as a particular earthquake of which the frequency contents are different from those in the seismic design.

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Due to these uncertainties in the earthquake ground motions, it is necessary to improve the seismic capacities of the safety facilities enough to ensure the seismic safety of the plant during strong earthquakes.

Kelly [1], Hall [2], and Ebisawa et al [3] proposed the use of base isolation systems for improving the seismic capacity of various components. The results of their studies indicate that the use of base isolation in light secondary equipment or a large component can be beneficial in reducing the accelerations experienced by the component. Ebisawa et al [3] conclude in their study that the seismic base isolation can improve the seismic resistance of nuclear components and decrease their functional failure probability.

This study evaluates the seismic safety of a nuclear power plant in the case that the seismic capacities of the equipment are increased through the improving systems such as the base isolation device. The relations between the seismic capacity of the equipment and the core damage frequency (CDF) in the nuclear power plant are investigated through a case study. The assessment procedures for the seismic core damage frequency in the nuclear power plant are addressed and then a case study for the Yonggwang Nuclear Units 5&6, operating pressurized water reactors (PWRs) in Korea, is described.

SEISMIC CORE DAMAGE FREQUENCY ASSESSMENT

Procedures

The seismic risk of core damage for a nuclear power plant can be calculated by the following procedures [4];

- Determine the local earthquake hazard, for example, hazard curve, site spectra or time histories for each plant site
- Identify accident scenarios induced by an earthquake, for example, initiating events and event trees for the plant which lead to a radioactive release
- Determine the failure modes for the safety and support systems
- Determine the fragilities for the important structures and components
- Determine the responses of all the structures and components for each earthquake level
- Compute the mean values and probability distributions of the accident sequence and core damage frequencies
- Perform sensitivity analysis to identify the dominant contributors to seismic risk and the relative contributions of the hazard curve, fragility and response uncertainties for the overall uncertainty in the core damage frequency

More detailed descriptions for each step are presented below.

Seismic Hazard Characterization

The seismic hazard at a given plant site can be characterized by a hazard curve which gives the probability of the exceedance of different peak ground accelerations. The hazard curve is derived from a combination of recorded earthquake data, estimated earthquake magnitudes of known events for which no data is available, local geological investigations, and expert judgments from seismologists and geologists familiar with the region. The region around the site is divided into zones, and each zone has an assumed uniform mean rate of earthquake occurrence. Then, for the region under consideration, an attenuation law is determined which relates the ground acceleration at the site to the ground acceleration at the earthquake source, as a function of the earthquake magnitude. The uncertainty in the attenuation law is specified by the standard deviation of the data regarding the mean attenuation curve. Finally, the hazard curve is computed by a statistical combining of the zonation, mean occurrence rate, magnitude distribution for each zone, and the attenuation law.

Identification of Accident Scenarios

In the event of an earthquake, the safety system in a nuclear power plant brings the plant to a safe shutdown condition. At this step the possible paths that a plant would follow are identified. These paths involve a seismic induced initiating event that causes shutdown, and a success or failure designation for plant systems affecting the course of the events.

The seismic analysis should be based on a subset of the initiating events and accident sequences developed for the internal event analyses of the nuclear power plant. Typically, the minimum set of the initiating events includes both loss of coolant accidents (LOCA) and transient events. In addition, the site-specific failure events, which act as initiating events, may be added to the minimum set.

In computing the frequency of the initiating events, a hierarchy tree must be established. The order of this hierarchy tree is defined such that, if one initiating event occurs, the occurrence of other initiating events further down the hierarchy is not significant in the level of the plant response. The seismic event trees should be taken directly from those developed for the internal events analysis, with modifications to include any seismically-induced failures.

Determination of Failure Modes and Fragilities

To determine failure modes for the safety systems, fault tree methodology, which can identify all groups of components in a system that would result in failure of the plant safety system, is used. The fault trees developed for internal events analysis may be used directly as the seismic fault trees, with certain modifications. Since the seismic fault trees include failures of basic events due to seismic ground motions, random failures, human error, and test and maintenance outages, the seismic failure modes such as local structural failures and the failure of critical passive components must be added to the internal fault trees. Failure of the plant safety systems due to building structural failures is also considered as a seismic failure mode.

Component seismic fragilities are obtained from a data base of generic fragility functions for seismically-induced failures or developed on a plant-specific basis for components not fitting the generic component descriptions. Fragility functions for the generic categories are developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. A generic fragility for any particular component can be estimated by selecting a suite of site-specific fragilities for that component.

Seismic Response Analysis

Building and component seismic responses are computed at several peak ground acceleration values on the hazard curve. Three basic aspects of the seismic response - best estimates, variability, and correlation - must be estimated. Building loads, accelerations and in-structure response spectra are obtained from the multiple time history analyses using the plant design models for the structures combined with a best-estimate model of the soil layer underlying the plant.

To compute the failure probability of critical components and safety systems, it is necessary to measure the maximum load or acceleration that the component experiences during an earthquake, as well as to measure the load or acceleration level at which it fails. Uncertainties in physical and dynamic characteristics of the soil, structures, and subsystems as well as the inherent variability of the free field earthquake motion influence on the response of the safety system to an earthquake.

Fragility Analysis

Component failure is taken as either loss of the pressure boundary integrity or loss of operability. Failure or fragility is characterized by a cumulative distribution function which describes the failure probability under the given loading. Loading may be described by the local spectral acceleration or moment, depending on the component and failure mode. The fragilities should be related to the appropriate local response to permit an accurate assessment of the effects of common-cause seismic failures in the evaluation of the accident sequences.

Since developing fragilities is usually the critical path item in a seismic risk assessment, it is necessary to reduce the work load through the use of proper methods. For example, screening of the accident sequences using conservative point estimate values for the seismic failure probabilities can reduce the work substantially.

Computation of Core Damage Frequency

Total core damage frequency is defined as the sum of the frequencies of all the accident sequences leading to core damage. In the quantification process, conditional accident sequence probabilities are determined at a number of peak ground acceleration (PGA) values, and then these are de-conditioned by an integration over the seismic hazard curve.

The frequency density of the core damage is calculated by multiplying the conditional probability of the core damage and seismic hazard. Then the total core damage frequency can be calculated by integrating the frequency density of the core damage as in the following equation:

$$F_{CD} = \int P_{CD}(PGA) F_{EQ}(PGA) d(PGA)$$

where $P_{CD}(PGA)$ is the cumulative failure probability of core damage as a function of the peak ground acceleration, and $F_{EQ}(PGA)$ is the hazard curve.

A CASE STUDY

To evaluate the improvement of the seismic safety of a nuclear power plant through an increase of the seismic capacity of the equipment, a case study for the Yonggwang Nuclear Units 5&6, Korean standard light water nuclear power plants with a capacity of one million kW each, is carried out.

Seismic Hazard Curve

The aggregate seismic hazard curves are derived from 130 seismic hazard curves at the plant site. Fig. 1 shows the final eight seismic hazard curves used in the analysis. These seismic hazard curves are obtained by considering the earthquake activity parameters, seismogenic zone, and weights for each attenuation law suggested by experts.

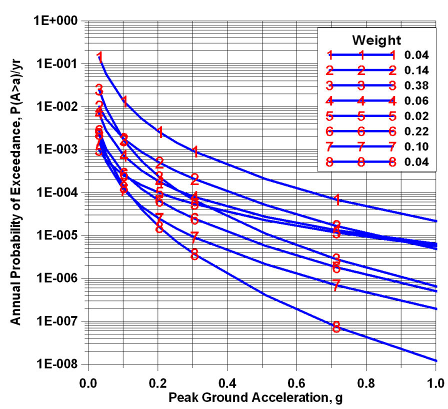

Fig. 1 Seismic hazard curves

Initiating Events

The initiating events induced by a earthquake are determined by the fragility analysis of the structures and components as well as by the analysis of the malfunction and failure effect of the electric components.

The initiating events considered in the external event analysis of the Yonggwang Nuclear Units 5&6 are as follows [5]:

- Loss of essential power (LEP)
- Loss of secondary heat power (LHR)
- Loss of component cooling water/essential chilled water (LOCCW)
- Small loss of coolant accidents (SLOCA)
- Loss of offsite power (LOOP)
- Seismic induced general transient (GTRN)

The occurrence frequencies for the initiating events are calculated as

$$P[IE(LEP)] = LEP$$
$$P[IE(LHR)] = \overline{LEP} * LHR$$
$$P[IE(LOCCW)] = \overline{LEP} * \overline{LHR} * LOCCW$$
$$P[IE(SLOCA)] = \overline{LEP} * \overline{LHR} * \overline{LOCCW} * SLOCA$$
$$P[IE(LOOP)] = \overline{LEP} * \overline{LHR} * \overline{LOCCW} * \overline{SLOCA} * LOOP$$
$$P[IE(GTRN)] = 1 - P[IE(LEP)] - P[IE(LHR)] - P[IE(LOCCW)] - P[IE(SLOCA)] - P[IE(LOOP)]$$

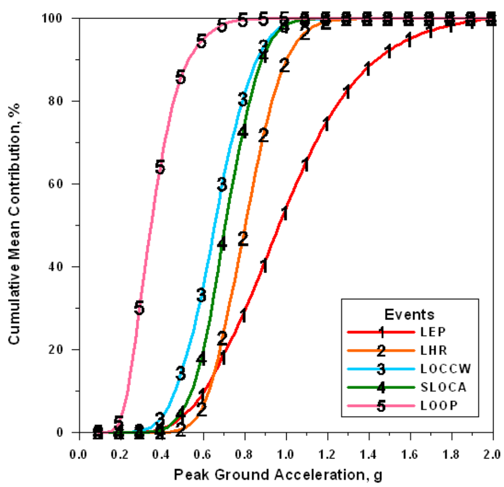

Fig. 2 Contribution of initiating events for core damage

Table 1 Occurrence frequency and core damage frequency for initiating events

Initiating event	Occurrence frequency	CDF
Loss of essential power	3.68E-06	3.68E-06
Loss of secondary heat power	1.16E-06	1.16E-06
Loss of component cooling water/ essential chilled water	2.48E-06	5.25E-08
Small LOCA	3.82E-08	3.82E-08
Loss of offsite power	1.12E-04	1.20E-06
General transient	2.79E-03	8.73E-07
Total		6.96E-06

Fig. 2 shows the cumulative mean contribution of each initiating event for the core damage. It is found that the loss of offsite power occurs at the lowest PGA, whereas the loss of essential power occurs at the highest PGA. It is also found from Table 1 that the loss of essential power is an important initiating event in calculating the total core damage frequency. The core damage frequency for the loss of essential power occupies more than a half of the total value 9.96E-06.

The loss of essential power, loss of secondary heat power, and small LOCA directly induce core damage, whereas the loss of component cooling water/ essential chilled water, loss of offsite power, and general transient are coupled to the secondary event trees.

Failure Modes

Fig. 3 shows the contribution of a component failure for the plant core damage. It is found that the failure of the Diesel Generator can contribute about 30% to the core damage. Based on the contribution shown in Fig.3, four high contribution components are selected from the plant components, and their failure modes and seismic capacity are listed in Table 2. In the table, HCLPF (High Confidence and Low Probability of Failure) that has a 95% confidence of not exceeding a 5% probability of producing failure indicates the seismic resistance of the component or equipment in terms of the gravitational acceleration and can be calculated by

$$HCLPF(g) = A_m(g) \times \exp[-1.65(\beta_R + \beta_U)]$$

where A_m is a median value of the ground acceleration, β_R and β_U are the lognormal standard deviation for the randomness and uncertainty, respectively.

Table 2 shows that the functional failure of Offsite Power occurs at around 0.15g, while the structural failure of a Battery Rack occurs at around 0.51g. The failure modes of the Diesel Generator and the Condensate Storage Tank are known as the concrete coning due to the anchorage failure and the sliding failure, respectively.

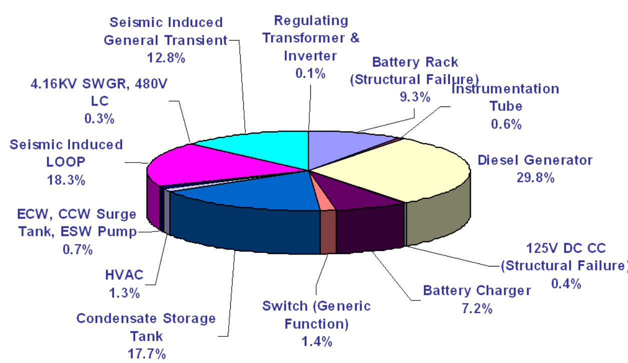

Fig. 3 Contribution of components for core damage

Table 2 Failure modes of selected equipment

Equipment	Failure mode	Mean frequency of failure	HCLPF(g)
Diesel Generator	Concrete coning	1.95E-06	0.38
Offsite Power	Functional failure	1.12E-04	0.15
Condensate Storage Tank	Structural failure (Sliding)	1.16E-06	0.41
Battery Rack	Structural failure	6.11E-07	0.51

When the seismic capacities of the components or equipment increase, the plant fragility curve will vary significantly according to their contribution to the initiating events. For instance, in the event of the seismic-induced loss of essential power as shown in Fig. 4, the increase of the seismic capacity of the Diesel Generator can improve the seismic resistance of the plant greatly, while the increase of the seismic capacity of the Offsite Power does not influence the seismic resistance of the plant. In the case of increasing the seismic capacity of all the selected equipment, the seismic resistance of the plant will be improved significantly.

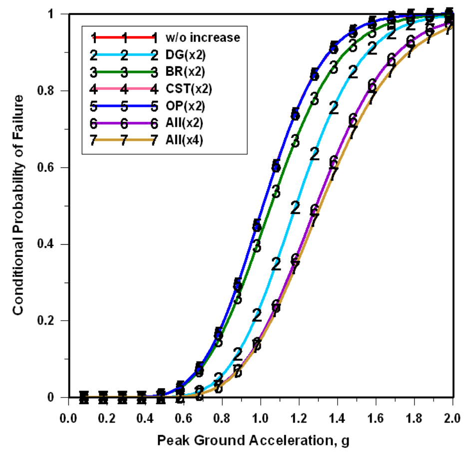

Fig. 4 Fragility curves with an increasing seismic capacity of equipment for the LEP

Core Damage Frequency

The core damage frequencies of the plant are obtained by a probabilistic safety assessment for a seismic event. To investigate the effect of the seismic capacity of the equipment on the core damage frequency, seismic capacity increase ratios of 25％ and 50％ are applied for the selected equipment in the analysis as shown in Table 2. Fig. 5 shows the relation between the cumulative mean frequency of the failure and the peak ground acceleration for the selected equipment with different increase ratios. In the legend of Fig. 5, 25, 50, and 75 indicate the seismic capacity increase ratios 25%, 50%, and 75%, respectively. The core damage frequencies and their ratios to the original value shown in Table 1 are summarized in Table 3. It is found from Fig. 5 and Table 3 that the failure of the Diesel Generator influences the core damage frequency significantly. In other words, increasing the seismic capacity of the Diesel Generator can improve the seismic safety of the plant remarkably. As shown in Table 3, when the Diesel Generator has a 25％ and 50％ increased seismic capacity, the core damage frequency will decrease by 16.2％ and 22.3％ respectively. This indicates an increase of more than 25％ in the seismic capacity of the Diesel Generator can improve the seismic safety of the plant by more than 16％.

Fig. 6 plots the ratios of the core damage frequency for the equipment with an increased seismic capacity to that with the original capacity according to the peak ground acceleration. It is found that the ratios of the core damage frequency are significantly influenced by the value of the peak ground acceleration, up to 1.0g. The effect of the seismic capacity of the equipment on the plant safety is remarkable in the PGA range of 0.3g to 0.5g. If the seismic capacities of all the selected equipment are improved, the core damage frequencies may decrease by about 5% and 30% at 0.2g and 0.3g, respectively. At 0.4g, increasing the seismic capacity of the Offsite Power will be more effective, and, under 0.6g, increasing both of the seismic capacities of the Offsite Power and the Diesel Generator will be more effective. In the case of the Offsite Power, at 0.4g, an increase of its seismic capacity of 25 and 50％ leads to a reduction of 33％ and 45％ in the CDF, respectively.

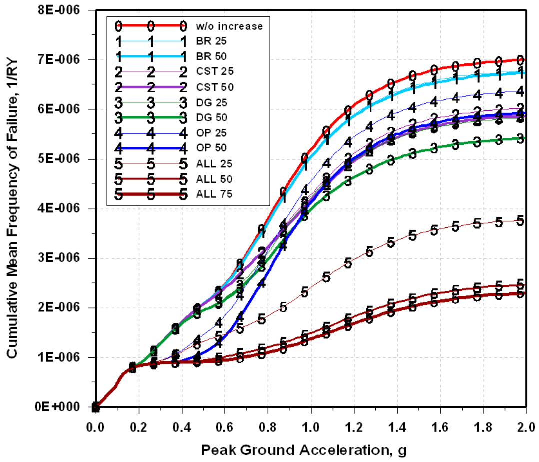

Fig. 5 Effect of seismic capacity on CDF

Table 3 CDF ratios for an increase of equipment seismic capacity

Equipment	Increase ratio (%)	CDF	CDF ratio (%)
Battery Rack	25	6.78E-06	2.6
	50	6.74E-06	3.2
Condensate Storage Tank	25	6.02E-06	13.5
	50	5.86E-06	15.8
Diesel Generator	25	5.83E-06	16.2
	50	5.41E-06	22.3
Offsite Power	25	6.36E-06	8.6
	50	5.93E-06	14.8
All	25	3.77E-06	45.8
	50	2.47E-06	64.5
	75	2.30E-06	67.0

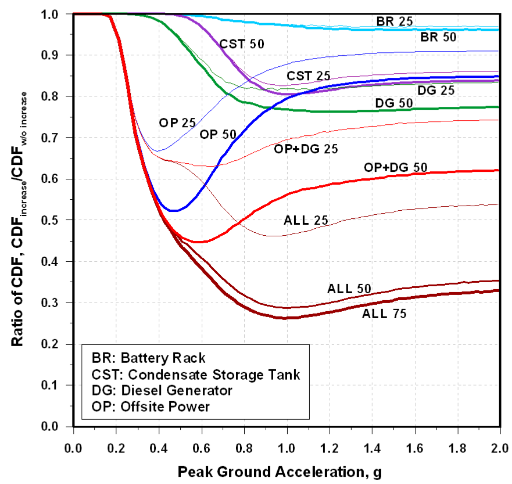

Fig. 6 Ratios of CDF by increasing seismic capacity

CONCLUSIONS

This study investigates the effects of the seismic capacity of nuclear equipment on the safety of a nuclear power plant through a case study. The relations between the seismic capacity of the equipment that can influence the plant safety and the core damage frequency are evaluated. The following are drawn from the results of a case study for the Yonggwang Nuclear Units 5&6:

- Seismic capacities of the Diesel Generator, Condensate Storage Tank, and Offsite Power contribute to the seismic safety of a nuclear power plant remarkably.
- Increasing the seismic capacities of the Diesel Generator by more than 25% can improve the seismic safety of the plant by more than 16%. In the case of increasing the seismic capacities of the equipment which exert a high contribution to core damage, the core damage frequency may be decreased by more than 50%.

- Core damage frequency is more sensitive to a value of the peak ground acceleration less than 1.0g. In this range, increasing the seismic capacity of the Offsite Power is more effective for improving the seismic safety of the plant.
- The effect of the seismic capacity of the equipment on the plant safety is remarkable in the PGA range of 0.3g to 0.5g. In the case of the Offsite Power, at 0.4g, an increase of its seismic capacity of 25 and 50% leads to a reduction of 33% and 45% in the CDF, respectively.
- Increase of the seismic capacity of the equipment or components will improve the seismic safety of a nuclear power plant significantly.

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