Application of inter-period correlation to the seismic PRA model of a nuclear power plant considering the detailed location of the structures and components providing the safety function.

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Abstract: In contemporary seismic risk assessments, important issue lies in addressing the concurrent failure of multiple safety systems due to seismic events. To mitigate this issue, it is imperative to accurately account for the damage correlation among components. Although inter-period correlation has been suggested as a viable approach, its application to actual plants presents issue. This study endeavors to develop a model incorporating these methods and to validate their applicability to operational plants.

Keywords: PRA, Inter-period Correlation, Seismic Motion Analysis, Multi-unit Site.

1. Introduction

Japan is a country prone to natural disasters such as earthquakes and tsunamis, underscoring the need to improve the safety of nuclear power plants. The 2011 incident at Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Plant prompted a review of nuclear safety regulations, culminating in the establishment of new regulatory standards[1]. These new standards require improved risk management and a probabilistic approach to risk assessment rather than a deterministic methodology. In particular, the seismic risk of nuclear power plants is significantly high compared to other hazards, requiring a precise and quantitative assessment to strengthen nuclear plant safety.

An Important lesson from the Fukushima Daiichi Nuclear Power Plant disaster is the need to address the simultaneous failure of multiple safety systems. Given the high seismic activity in Japan, it is crucial to consider the simultaneous failure of systems due to seismic events. In seismic risk assessment, it is essential to assess the damage to components induced by seismic motion and to consider the interdependencies between them. If these damage correlations remain ambiguous, it becomes impossible to predict the effectiveness of risk mitigation strategies such as diversification, which involves changing the location of components to minimise simultaneous damage.

Although numerous methods have been proposed to deal with correlations, the periodic correlation method, which comprehensively evaluates response correlations by considering the periodic characteristics of component responses and installation locations, has not yet been implemented in an actual plant and remains at the proposal stage. An additional advantage of this method is its scalability; since the correlation is based on the periodic characteristics of the component responses, there is no need to consider the correlation coefficients individually during modelling, allowing both multi-unit and single-unit sites to be evaluated.

In this context, the objective of this study is to verify the applicability of a risk assessment method that incorporates periodic correlations of component responses to a real plant and to evaluate the impact of damage correlations. This will be demonstrated using Unit 7 of the Kashiwazaki Kariwa Nuclear Power Plant as a case study, focusing on the overall safety system.

2. Related Researches

In March 1999, the Japan Atomic Energy Research Institute (JAERI) developed a methodology for probabilistic safety assessment (PSA) of nuclear power plant risks due to earthquakes and validated its applicability to a light water reactor model plant. The JAERI seismic PSA methodology consists of four steps: (1) seismic hazard assessment, (2) building and component response assessment, (3) component load-bearing capacity assessment, and (4) seismic system reliability analysis. This approach is similar to the seismic PSA methodology developed in the United States.

(1)Seismic Hazard Assessment utilises Japan's extensive seismological expertise. Potential earthquake sources at the target site are modelled, and the magnitude and frequency of earthquakes are estimated using the Gutenberg-Richter equation. (2) In the response evaluation of buildings and components, synthetic seismic waves are generated based on the magnitude and epicentral distance of the most severe earthquake for design purposes identified in step (1). Time history response waveforms and floor response spectra for each floor are then obtained from the seismic response analysis. (3) To evaluate the resistance of the components, the results of the seismic tests are integrated with design information and general data from the United States. (4) The seismic system reliability analysis uses a Monte Carlo method with correlation coefficients to assess the impact of damage correlations, which are considered critical in the seismic PSA method, in conjunction with accident sequence analysis and severity assessment.

In this study, the analytical model is developed following the methodology established by JAERI. While the overall analytical flow mirrors the JAERI method, this study differs in its treatment of the correlation between the seismic source model and the resulting damage. The JAERI method uses the most severe earthquake estimated from seismological knowledge for design, whereas this study uses a large dataset of seismic records specific to the target site. In terms of damage correlation, the JAERI method determines correlation coefficients for component responses based on the rules of NUREG-1150, while this study adopts an inter-period correlation method that takes into account the periodic characteristics of the components and their locations. These methodological details are explained in the following research methodology section.

3. Research Method

In this study, a quantitative risk assessment method for nuclear power plants with different redundant components having different natural periods has been proposed by reference [3], emphasizing the periodic characteristics of the component responses. This method uses the Ground Motion Prediction Equation (GMPE) for seismic motion analysis and incorporates a Monte Carlo simulation method, integrating the inter-period correlation approach proposed by Baker and Jayaram to account for the variation and correlation in component responses.

The process of this study is illustrated in the flowchart in Figure 1 and Figure 2.

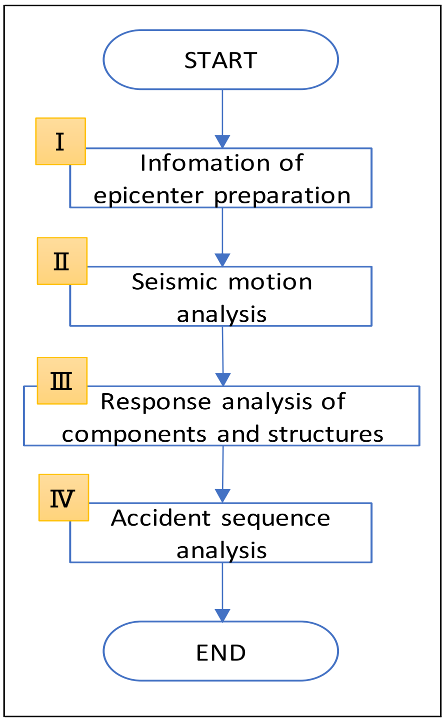

Figure 1: Flowchart of core damage assessment

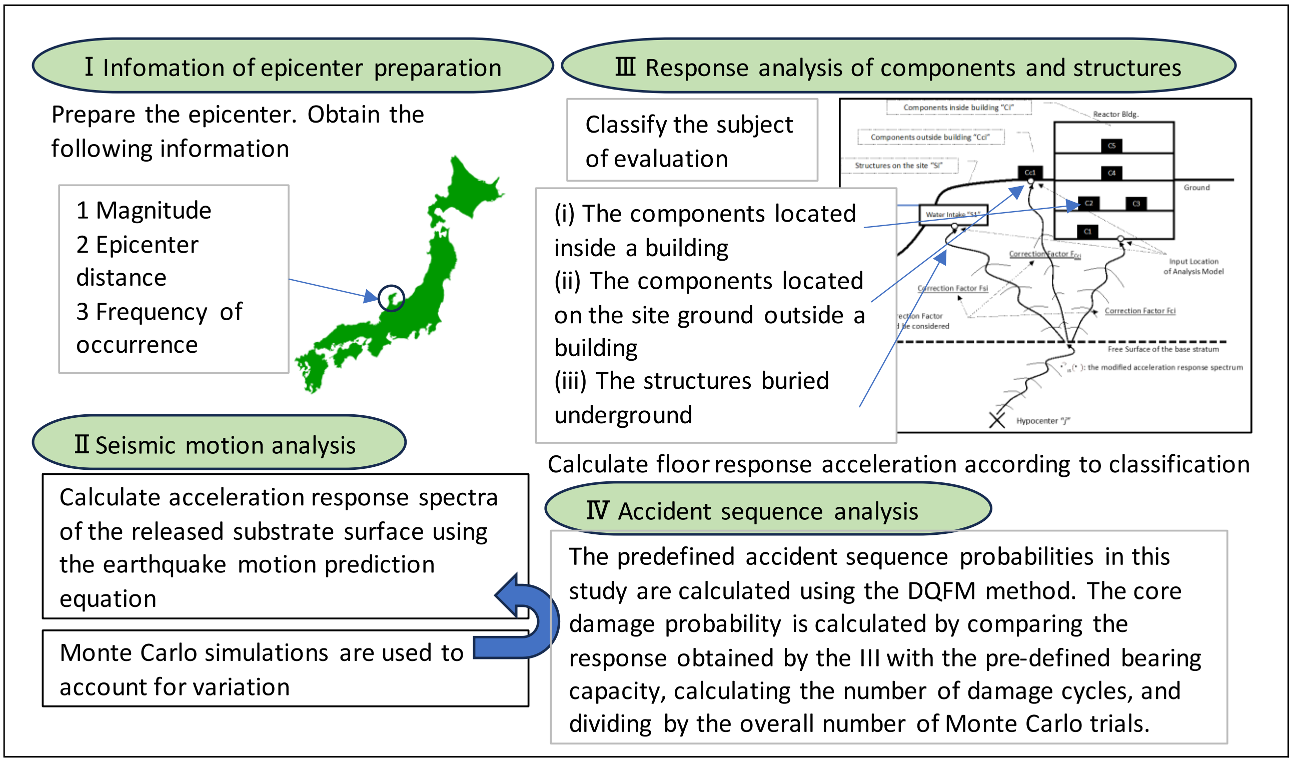

Figure 2: Details of core damage assessment

3.1. Information of epicenter preparation

Epicentre information is generated using data specific to each region in which the nuclear power plant under consideration is located. This source information consists of numerous seismic records detailing epicentre distance, magnitude and frequency of occurrence. The rationale for using an extensive set of seismic records is that a seismic hazard assessment that considers all earthquakes affecting a particular site may not adequately capture inter-period correlations between structures affected by seismic waves. To account for inter-period correlations, a diverse set of seismograms is required. The source information Ej (where Ej is the total number of earthquake records for the target epicentre) is characterised by epicentral distance, magnitude and frequency of occurrence and these data are used in steps 3.2-.3.4.

3.2. Seismic motion analysis

Seismic motion analysis is performed using the seismic source information. The Morikawa GMPE equation is used for this analysis. The acceleration response spectrum at the free surface of the base stratum is derived from the GMPE equation. To obtain the acceleration response spectrum at the ground surface, a correction is applied between the free surface of the base stratum and the location of each component, taking into account their specific positions within the nuclear power plant. This study also integrates the inter-period correlation using correlation coefficients based on the concept proposed by Baker and Jayaram [4], as shown in Figure 3.

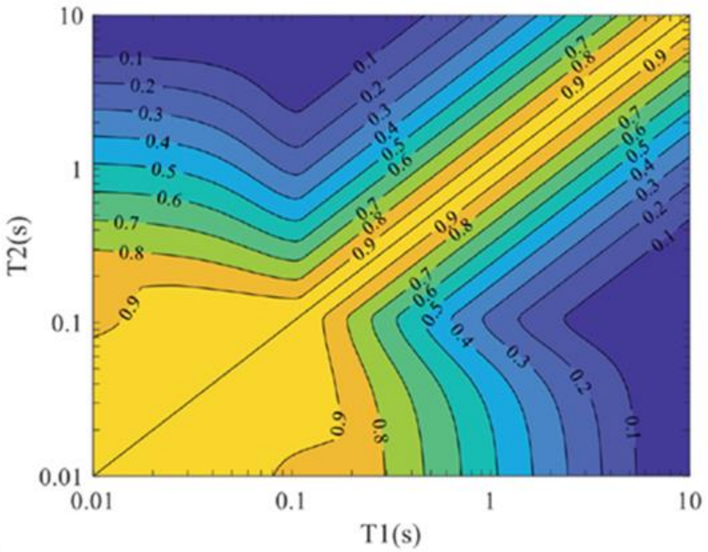

Figure 3: Correlation coefficients of inter-period correlations

3.3. Response analysis of component and structure

Core damage scenarios are defined by an integrated fault tree that considers the combination of events leading to core damage. In fault tree analysis, it is essential to consider the uncertainty of each event when assessing the probability of the top event occurring. Although there are various methods for estimating uncertainty, this study uses Monte Carlo simulation because of its suitability for complex models, as it does not require the creation of a rigorous mathematical model due to the randomness in data such as component parameters. Whilst it is ideal to model all components and elements potentially damaged by an earthquake within the fault tree, a limited scope is used for simplicity. In this study, a scenario is considered where core damage occurs when two redundant systems, consisting of the residual heat removal (RHR) system, the reactor coolant water (RCW) system, the reactor coolant seawater (RSW) system and the emergency diesel generator (EDG), fail simultaneously during a power loss. The components selected for risk assessment are based on this core damage scenario. The location of each component is classified as follows:

i) Components located inside the building (main components of RHR, RCW, RSW, EDG)
ii) Components located directly on the ground outside the building (light oil tanks)
iii) Underground structures (water intake)

The soil response acceleration is calculated separately for each site category. Uncertainties are incorporated from the Direct Quantification of Fault Tree Using the Monte Carlo Simulation (DQFM) method [2] using the Monte Carlo simulation described above. Specific calculation methods are discussed below.

3.4. Accident Sequence Analysis

The accident sequence is evaluated using the DQFM method. The damage is determined by comparing the response acceleration at the natural period of the component, as calculated in the component response evaluation (3.3), with the predefined load-bearing capacity of the component. Uncertainties in the load-bearing capacity are considered using a Monte Carlo simulation. The core damage probability is calculated by dividing the number of instances where the core damage logic is satisfied by the total number of Monte Carlo simulation runs. The core damage frequency is then determined by multiplying this probability by the frequency of the target earthquake.

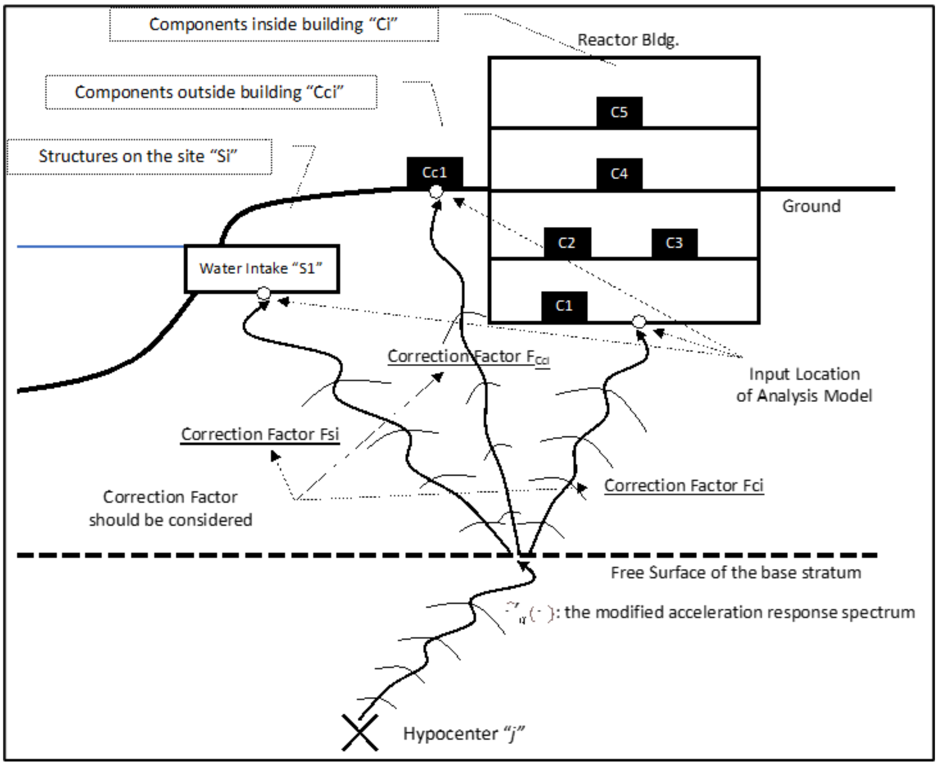

Figure 4: Classification of components at risk

4. Result

This study is solely an investigation aimed at developing evaluation methods, rather than presenting research findings.

5. Conclusion

In this study, an analytical model was developed and its applicability was confirmed, which is capable of evaluating the periodic characteristics of responses and calculating accident sequence frequencies based on the location of nuclear power plant components by analysing seismic motions using various source information for nuclear power plant sites. The applicability to multi-unit sites was also validated.

In the development of a realistic nuclear power plant model, an issue arises regarding the correction of acceleration response spectra for each classification of components subject to risk assessment in "3.3. Component Response Assessment". For components inside the building (Classification i), data on the installation position, natural period and damping coefficient from Reference [5] are used to correct for the release base area and the installation position of the component. For components installed directly on the ground outside the building (Classification ii), corrections are made using the ground gain factor. For underground structures (Classification iii), although various assessment methods have been proposed, this study targets a complete plant and uses the DQFM method, which requires consideration of increased computational costs. Current assessment methods for underground structures include the relatively simple two-dimensional finite element method and the more sophisticated three-dimensional finite element method. The challenge is to select and incorporate into the programme an evaluation method that provides sufficient analysis accuracy at low computational cost.

Consideration must also be given to the computational cost of performing the evaluation for the entire plant. The computational cost of the method used to evaluate the above underground structures should also be considered.

References

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