Current Status and Issues of Fragility analysis for Nuclear Power Plant SSCs

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1. Introduction

Since the implementation of accident management measures to ensure the safety of nuclear power plants, interest in seismic safety of nuclear power plants has increased significantly. This is because the effects of earthquake induced risk have been shown to be very significant in the final results of probabilistic safety assessments to achieve the safety goals of nuclear power plants. Therefore, utility has conducted updating the seismic fragility of NPP SSCs and overpressure fragility of reactor containment buildings to evaluate seismic risk of representative nuclear power plants for all kinds of reactor type. This paper summarized the current status and major issues of seismic fragility and containment overpressure analysis.

2. Seismic fragility of NPP SSCs

The seismic risk assessment of nuclear power plants requires the evaluation of seismic fragility of structures and equipment related to accident scenarios for seismic events. In this study, the seismic fragility analysis of representative NPPs was performed using the latest technology and the latest information and data. The basic direction of the seismic fragility analysis was to follow the procedures and methods of the 2009 edition of the ASME/ANS standard [1] and to perform the seismic fragility analysis to ensure the quality corresponding to Capability Category II of the standard.

The reference earthquake required to perform the seismic fragility analysis was the NUREG/CR-0098 spectrum, which is consistent with the ASME/ANS standard requirement and is consistent with each site due to the lack of site-specific uniform hazard spectrum for Korean nuclear power plant sites. The key issues that emerged from the seismic fragility analysis are summarized below.

- Validation of the use of NUREG/CR-0098 spectrum as a reference earthquake in the absence of site-specific uniform hazard spectra
- Assessing and reducing uncertainty in fragility parameters for SOV (Separation Of Variables) method
- Ultimate capacity of concrete anchors due to changes in ACI (American Concrete Institute) design code
- Vulnerability of buildings to activity and conduction
- Evaluation of devices with inadequate data and information for seismic vulnerability assessment and vulnerability assessment of non-seismic devices

3. Containment overpressure fragility

The probabilistic overpressure fragility analysis of the reactor containment building for the Level 2 PSA (Probabilistic Safety Assessment) has been performed with relatively simple calculations. However, the ultimate pressure capacity analysis of the reactor containment building at the design stage has been performed based on nonlinear analysis using a three-dimensional FEM model. Therefore, nonlinear response analysis using a three-dimensional FEM model is required to analyze the probabilistic overpressure fragility analysis of the reactor containment building for the Level 2 PSA, and this study developed probabilistic overpressure fragility analysis methods based on a three-dimensional nonlinear analysis. Figure 1 shows the nonlinear response analysis results of the containment building according to the increase of internal pressure.

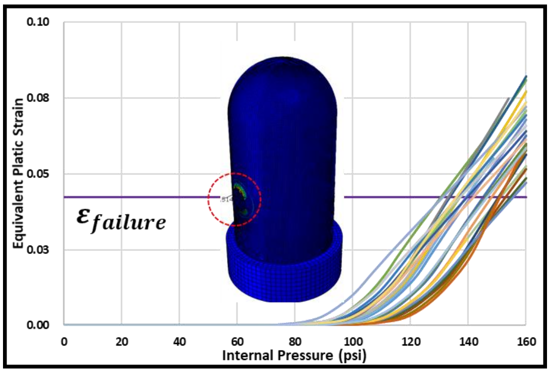

Fig. 1. Non-linear analysis of containment building.

The methods developed in this study are a sampling-based approach utilizing LHS (Latin Hypercube Simulation) sampling and a SOV method similar to the seismic fragility analysis method. Figure 2 shows an example containment overpressure fragility curve based on the sampling approach. The main issues derived from the probabilistic overpressure fragility analysis of the reactor containment building are as follows.

- Nonlinear models of components (concrete, reinforcing steel and tendons) for nonlinear analysis of the reactor containment building
- Prestressing forces in tendons in the current state of the plant during operation
- NPP-specific material property values
- Failure criteria for various failure modes (leak and rupture)
- Uncertainty of fragility parameters (modeling uncertainty, etc.)
- Calculation of combined fragility for multiple damage modes

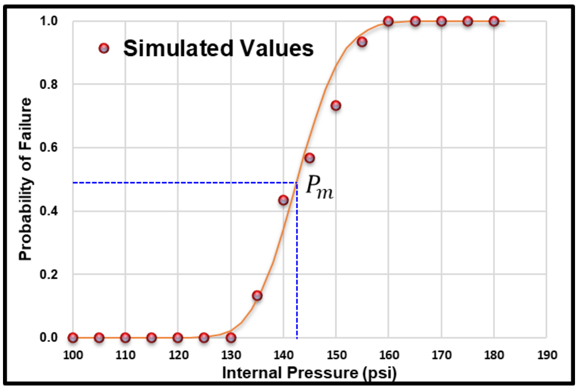

Fig. 2. Example containment overpressure fragility curve developed form simulated values using random sampling.

4. Probabilistic Seismic Response Analysis

The seismic fragility analysis of NPP SSCs have been performed based on seismic design data. The seismic design of Korean nuclear power plants is mostly designed using the standard response spectrum presented in the U.S. NRC R.G. 1.60 [2], rather than the site-specific design earthquake. In addition, in the past seismic design, the design response was calculated from the fixed foundation structure model using the lumped mass stick model, and the FRS (Floor Response Spectrum) was calculated from the seismic analyses by using this model. However, due to the change of SRP (Standard Review Plan) [3], which is a criterion for considering SSI, it is necessary to reflect the effect of soil-structure interaction, and especially in recent years, seismic response analysis using 3D finite element models has been widely used in re-analysis for seismic fragility analysis. In particular, the EPRI report [4] requires the re-analysis of seismic response when the shape of the reference earthquake is significantly different from the design response spectrum and the effects of soil-structure interaction must be reflected.

In this study, a feasibility study of seismic fragility analysis based on probabilistic SSI analysis was conducted by calculating ISRS (In-Structure Response Spectra) from the results of probabilistic soil-structure interaction analysis using a three-dimensional structure and using the results to re-evaluate the fragility of several structures and devices.

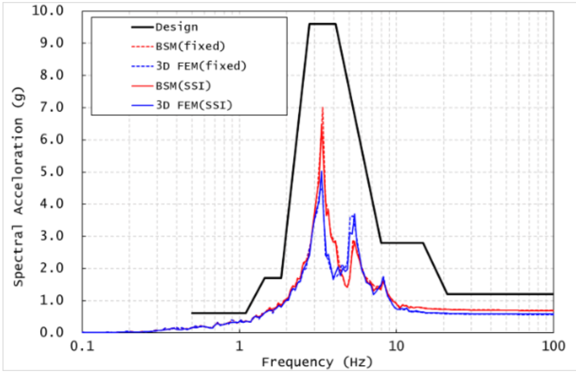

Fig. 3. Example comparison of ISRS according to the seismic analysis models.

5. Summary

In this study, the seismic fragility and containment overpressure fragility analysis methodology for the representative nuclear power plants operating in Korea were established and used to re-evaluate the seismic fragility of NPP SSCs and overpressure fragility of containment buildings. In addition, the ISRS was re-evaluated through probabilistic SSI analysis using a three-dimensional structure model to reflect SRP regulations and recent trends, and the seismic fragility was re-evaluated and evaluate its applicability in the future.

REFERENCES

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