INSIGHTS FROM PROBABILISTIC SAFETY ASSESSMENT-BASED SEISMIC MARGIN ANALYSIS OF WESTINGHOUSE NPP IN KOREA

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Nuclear power plants of Kori site built before 1985 were applied old technical standard for the seismic qualification of SSCs. These plants should be addressed as Unresolved Safety Issue A-46 (Seismic Qualification of Equipment in Operating Nuclear Plants) to verify the seismic adequacy of essential equipment that have not been qualified in accordance with current standard. Thus, the nuclear regulatory and licensees in Korea have been re-evaluated the seismic adequacy of SSCs and the seismic margin assessment to demonstrate a sufficient margin over the SSE to assure plant safety and to find any "weak links" which might limit the safe shutdown capability. In this paper, the seismic margin is re-assessed for the old Kori plant in order to show that the plant high confidence low probability of failure (HCLPF) is higher than the review level earthquake (RLE). The seismic margin of the old Kori plant is re-assessed by a probabilistic safety assessment-based seismic margin assessment (PSA-based SMA), following a methodology developed by the US NRC. The data before re-evaluating the seismic adequacy of SSCs and the seismic margin assessment are used and recent data in the new technical report are used to assess the seismic margin. Also, the Level 1 internal event PSA model has reflected the design changes and the revisions after 2011 were used.
This paper summarizes the methodology of the PSA-based SMA, including the results and insights gained from a Level 1 PSA-based seismic margin assessment.

I. INTRODUCTION

Nuclear power plants of Kori site built before 1985 were applied old technical standard (Ref. 1) for the seismic qualification of SSCs. Kori plants are the Westinghouse nuclear power plants. These plants should be addressed Unresolved Safety Issue A-46 (Seismic Qualification of Equipment in Operating Nuclear Plants) to verify the seismic adequacy of essential equipment that have not been qualified in accordance with current standard (Ref. 2). Thus, the nuclear regulatory and licensees in Korea have been re-evaluated the seismic adequacy of SSCs and seismic margin assessment to demonstrate sufficient margin over the SSE to assure plant safety and to find any "weak links" which might limit the safe shutdown capability. In this paper, the seismic margin is re-assessed for the old Kori plant in order to show that the plant high confidence low probability of failure (HCLPF) is higher than the review level earthquake (RLE).

The seismic margin of the old Kori plant is re-assessed by a probabilistic safety assessment-based seismic margin assessment (PSA-based SMA), following a methodology developed by the US NRC. The data before re-evaluating the seismic adequacy of SSCs and seismic margin assessment are used and recent data in the new technical report are used to assess the seismic margin. Also, the Level 1 internal event PSA model has reflected the design changes and the revisions after 2011 were used.

II. METHODOLOGY

The Level 1 internal event PSA model has been updated after Fukushima accident, 2011. This model would be the starting point to re-evaluate the seismic margin for the old Kori Plant. The PSA-based SMA is based on the safe shutdown equipment list (SSEL). High confidence of low probability of failure (HCLPF) capacities, which are obtained for each SSCs in the SSEL.

Seismic-induced initiating events are determined to develop the SSEL. Seismic event trees are developed by considering all potential sequences leading to core damage. For the systems modelled in these event trees, seismic fault trees are constructed based on the success criteria. The seismic event trees and fault trees are quantified to make cutsets that lead to core damage. These cutsets has information about seismic-induced failures, random unavailabilities of equipment, and operator error probabilities. The plant-level HCLPF capacity should be determined based on the sequence-level HCLPF values for all sequences. The MIN-MAX method specified by the NRC seismic margin methodology is acceptable for computing sequence-level HCLPF.

II.A. Safe Shutdown Equipment List(SSEL)

The first step for developing the SSEL is to determine the potential initiating events that could occur as a result of a seismic event. The potential initiating events identified in the internal events PSA must be reviewed and taken into account to ensure that all potential initiating events are covered in the SMA (Ref. 3, 4).

The SSEL provides a documented list of the plant structures, systems, and components that could be used for responding to an earthquake or mitigating potential core damage initiated by a seismic-induced event. These are necessary to perform the safety functions and include front line and support systems. The support systems provide services to the front line systems. The SSEL is identified from the Level 1 internal event PSA model. The internal events PSA fault trees do not provide a complete list of equipment for the SMA; structural items must be added to the list, for example, electrical panels and cabinets, walls, and buildings. For each safety function, the safety system(s) and its components must be identified. The manual valves, check valves, small relief valves, and other passive equipment are not included in the SSEL. Based on earthquake experience data and past seismic evaluations, these are judged to be seismically rugged. However, during a seismic walk-down, these items must be checked. The sources of information for the SSEL include the seismic qualification equipment list, the basic event list for the internal events PSA, design or operational flow sheets, and elementary wire drawings. Component information that is required for items on the SSEL includes the component identification, description, redundancy, component location (room and elevation), type/class of component, normal operating position, fail-safe position, manufacturer, power supply (control and power), and any other conditions that may apply.

For this study, the initial list of equipment was identified using the result of initial PSA-based SMA(2003), EPRI SMA(2007), and recent data of seismic margin assessment.

II.B. Seismic Fragility Analysis

The seismic fragilities are calculated for SSCs developed from the SSEL. A fragility analysis is performed to obtain the seismic margin of SSCs that could have an effect on safe shutdown of the plant following a seismic event. In this analysis, the seismic margin values of SSCs modeled in the accident sequences are obtained. The seismic margin is expressed in terms of HCLPFs (Ref. 5).

$$\mathrm{HCLPF} = A_m \times \exp(-1.65 \times (\beta_R + \beta_U)). \qquad (1)$$

or

$$\mathrm{HCLPF} = A_m \times \exp(-2.33 \times \beta_C). \qquad (2)$$

- A_m : median capacity
- β_R : logarithmic standard deviation representing the randomness
- β_U : logarithmic standard deviation representing the uncertainty
- β_C : composite logarithmic standard deviation

The median capacities and HCLPFs are expressed in terms of the peak ground acceleration (PGA). An earthquake of 0.3g PGA is defined as the review level earthquake for the old Kori plants.

For this study, the HCLPF values for the SSEL was identified using the result of initial PSA-based SMA(2003), EPRI SMA(2007), and recent data of seismic margin assessment.

II.C. Seismic Accident Sequence Analysis

Seismic event trees are developed with considering all potential sequences leading to core damage. For the systems modelled in these event trees and seismic fault trees are constructed based on success criteria.

Seismically-induced initiating events are identified from the internal events analysis results. However there are some major differences between the seismic and internal events for the purpose of identifying an initiating event category, which are as follows: seismic events may simultaneously damage multiple redundant systems and components in the plant. The Identified seismic initiating event categories are modeled as hierarchy structures.

The seismic fault trees are developed from the internal events PSA model to include the important accident sequences. This model also contains random failures and human errors from the internal events PSA. System models are modified to accommodate a seismic event.

The seismic event trees and fault trees are quantified to make cutsets that lead to core damage. These cutsets has information about seismic-induced failures, random unavailabilities of equipment, and operator error probabilities. The plant-level HCLPF capacity should be determined based on the sequence-level HCLPF values for all sequences. The MIN-MAX method (Ref. 6, 7) specified in the NRC seismic margin methodology (Ref. 8) is acceptable for computing sequence-level HCLPF.

The Min-MAX method used to determine the HCLPF capacity of an accident sequence from the HCLPF capacities of the contributing SSC failures, or the HCLPF capacity of the plant as a whole from the HCLPF capacities of a group of seismic-induced accident sequences. The overall HCLPF capacity of two or more SSCs that contribute to a sequence using OR Boolean logic is equal to the lowest individual HCLPF capacity of the constituents of the group. If AND Boolean logic is used, the HCLPF capacity of the group is equal to the highest individual HCLPF capacity of the constituents. When evaluating several accident sequences to determine the “plant level HCLPF capacity”, the plant-level HCLPF capacity is equal to the lowest of the sequence-level HCLPF capacities.

II.D. Assumptions

The major assumptions for the SMA model are as follows:

- It is assumed that the seismic event would result in a LOOP, since offsite power equipment is not met 0.3g HCLPF.
- No credit is taken for non-safety-related systems. They are assumed in the model to have failed or to be non-functional due to the seismic event.
- Failure of the reactor trip signal is not modeled since the breakers for motor generator sets would be de-energized following a LOOP due to a seismic event, thereby causing the release of control rods into the core even if the reactor trip function fails.

- Failure of buildings that are not seismic category I does not impact SSCs designed to be seismic category I. Seismic spatial interactions between SSCs designed to be seismic category I and any other buildings will be avoided by proper equipment layout and design.

III. RESULTS

Two seismic-induced initiating events are listed below.

- S-LOEP: Seismic-induced Loss of Essential Power
- S-LOOP: Seismic-induced Loss of Off-site Power

Seismic-induced LOEP represents failure of safety related essential power. Theses failures may lead to core damage, because the SSCs for safe shutdown cannot be operated. It is not necessary to perform the accident sequence analysis. On the contrary, the seismic event tree is developed to evaluate accident sequence for S-LOOP (Fig. 1).

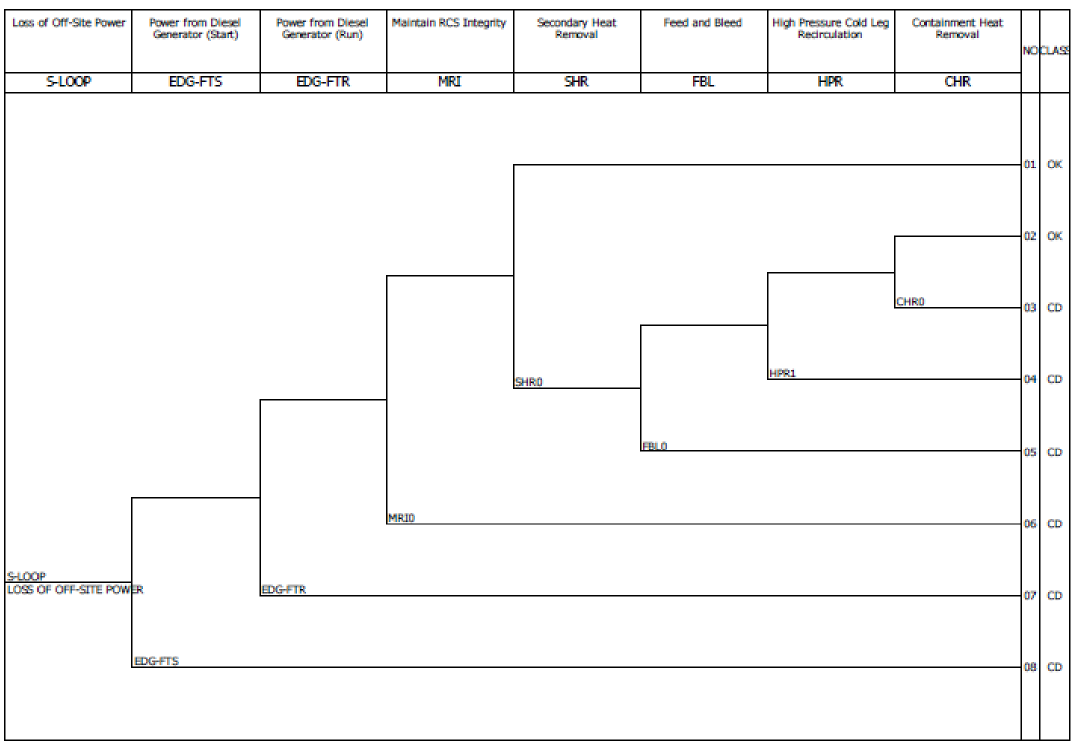

Fig.1. Event Tree of Seismic-induced LOOP

The dominant sequence HCLPFs are shown below.

- %S-LOOP(< 0.3g) & EDG A Fails to Run(0.3g) & EDG B Fail Fails to Run (0.3g) = Seq No.8 (0.3g)
- %S-LOOP(< 0.3g) & CCF of EDG A Fails to Run(0.3g) = Seq No.8 (0.3g)

The Refueling Water Storage Tank (RWST) in the SSEL is lower than RLE, 0.3g. This component may be unavailable to perform the feed & bleed operation, but there is another way to mitigate the accident. It is possible to safe shutdown operating the secondary heat removal.

IV. INSIGHTS

The SMA results identified some risk insights as follows:

- There are some important safety-related SSCs for which seismically induced failure would lead directly to core damage. In this SMA, these SSCs have HCLPF values in excess of 0.30 g except for the RWST(< 0.3g). RWST is affecting the feed & bleed operation after the seismic-induced initiating event. However, there is another success path to mitigate the accident using the steam generator, auxiliary feedwater and related components (secondary heat removal). These components have enough seismic margin.

V. CONCLUSIONS

The Level 1 PSA-based SMA was re-evaluated for the old nuclear power plants. The results of the initial SMA conducted in 2003, EPRI SMA conducted in 2007 and the latest Level 1 internal event PSA model were used to re-evaluate the SMA. The methodology for the SMA is specified by the NRC. The result obtained in this analysis show that the most SSCs in the SSEL have more than the HCLPF capacity of 0.3g PGA except for RWST. RWST is an important component to mitigate the seismic-induced accident but it is confirmed that there is another path operating the secondary heat removal from the seismic event tree.

ACKNOWLEDGMENTS

This work was supported by Central Research Institute of Korea Hydro & Nuclear Power Co., Ltd (KHNP CRI) in Korea.

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