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Review of Risk Reduction Methods using Probabilistic Safety Assessment Insights and Improved Technology

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1. Introduction

Although several efforts have been made since the Fukushima accident and the station blackouts of domestic nuclear power plants, the public still have concerns regarding nuclear power plant accidents and the regulations for accident management are being reinforced in advance. Korea Hydro & Nuclear Power Co. (KHNP) has a mission to explain that nuclear power generation is more efficient and safer than the other power production methods currently available in the industry. In particular, it must be demonstrated with quantitative but easily understandable methods that the safety designs and equipment of nuclear power plants can mitigate accidents sufficiently and that the employees in nuclear power plants are ready to manage any accident conditions.

2. Strategies for Risk Reduction

According to the recent legalization of the periodic safe review (PSR) and accident management, KHNP is preparing to revise the probabilistic safety assessment (PSA) in order to integrate the mitigation strategies including system improvements, movable components, and mitigation equipment from the Fukushima countermeasures so that nuclear power plants are able to manage risk based on the strengthened safety performance goals. In addition, KHNP is considering implementing the PSA combined with the procedures including steps to mitigate accidents or to proactively prevent accidents using flexible equipment.

In terms of analytic improvements, the uncertainties in the PSA model will be reduced during the analysis of the external events because several engineering judgments, conservative assumptions, and parameters are included during the hazard analysis, code simulation, and modeling process used to develop the PSA inputs to quantify risk values.

2.1. Accident Response Equipment

Recently, US Utilities reviewed the application of the diverse and flexible mitigation strategies (FLEX) including the movable equipment in the PSA model; they are preparing its implementation through discussions with the regulator [1].

For domestic plants, the accident response equipment or components such as the movable diesel generators, the containment filtered vent system (CFVS), the external cooling water injection line to the reactor and steam generators, and the movable diesel-driven pumps are being installed as Fukushima countermeasures. In addition, various efforts are in progress to improve the safety of nuclear power plants through including several new equipment and structures such as a passive autocatalytic recombiner (PAR), a coastal protection wall against extremely high waves, waterproof doors, and alternative cooling water sources, as depicted in Fig. 1. These accident response systems, structures, and components will be reflected in the PSA model in order to confirm the safety improvement effects leading to changes in the major accident sequences and the resultant quantitative risk reduction.

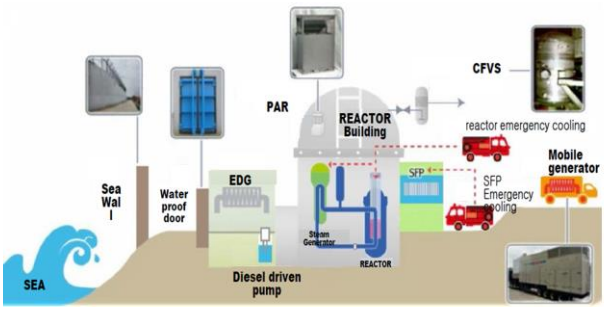

Fig. 1. Accident response equipment and structures for nuclear power plants [2].

2.2. External Event Analyses

Realistic methodologies with technical foundations must be developed through research in order that the analyzed results might be similar to the actual event conditions in order to reduce the current significant uncertainties in the external PSA that have resulted from several expert judgments and engineering parameters provided under insufficient technical assumptions. In particular, the fire PSA might have a large uncertainty in the model and its analysis method compared with other external analyses.

According to recent nuclear regulation information [3], approximately 50% of nuclear licensees have attempted to use the new fire analysis methodology, NUREG/CR-6850 [4], and are preparing to transition to performance-based regulations in the fire protection area. This new methodology has been continuously improved through repeated simulations and experiments in order to improve the conservative methods and assumptions, and to approach actual physical conditions. An available data in this method is the fire ignition frequency

[5], and it is expected to be easily applicable to revise the PSA model in the short term, whereas US experiences [3] indicate that the application of whole methodology may cause difficulty in extracting meaningful risk insights.

For the seismic PSA, the risk result is relatively high although there are minor effects of seismic events that affect domestic nuclear power plants based on historical evidence. Therefore, as one improvement, the current seismic PSA models must be revised using new hazard analyses of the plant sites. For example, the application of a new seismic hazard curve to the PSA model could reduce the seismic risk of the plant by approximately 30% (see Fig. 2) according to the recently revised PSA report in the periodic safety review for one nuclear power plant in KHNP.

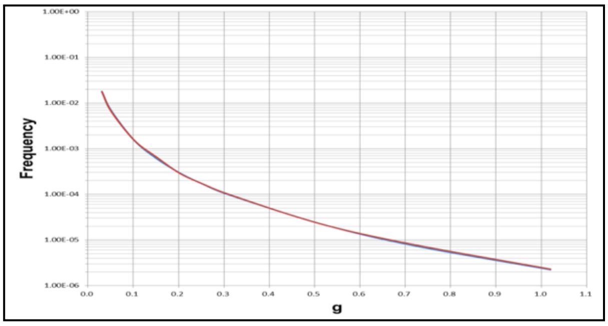

Fig. 2. Seismic hazard curve of an example plant site.

In the case of the flooding PSA, the water flow rate from the failed pipes will be separated into various flow rates, more than the previous ones, in order to locate and reduce the conservative risk evaluation factors using the recent evaluation methodology [6] pertaining to the rupture frequencies of the pipes. In this process, a detailed evaluation regarding the conservative operator recovery should be conducted regarding whether operators can mitigate the flooding condition within the limited time. Here, the drain system and the protection devices from water spray, which are installed as safety improvements, should be considered in the PSA model.

2.3. Quantification Code

Recently, the US Nuclear Regulatory Commission (NRC) and multi-national cooperation have indicated that there have been technical developments based on experiments and simulation codes in the external risk evaluation area. In this process, EPRI developed ACUBE, which is an improved module of the PSA quantification code CAFTA, and began its implementation in the external PSA in order to calculate the precise risk value [7].

Table 1. PSA code and its improvement status [7]

Organization	Development	PSA Application	Quantification	Refined Solution
Domestic Industry	KEPCO E&C	SAREX	FTREX	None
Domestic Regulation	KAERI	AIMS	FTREX	None
U.S. Industry	EPRI	CAFTA	FTREX	ACUBE
U.S. Regulation	NRC/INL	SAPHIRE	Specific S/W	CUT_BDD
European Industry	ScandPower	RiskSpectrum	RSAT	MCS_BDD

KHNP also needs to conduct the quantification of its cutsets using a refined method using more upgraded algorithm. The current risk results might have conservative values because the cutsets have been calculated based on the minimal cut upper bound (MCUB) method combined with the initiating frequencies that have relatively large values. If software with the precise calculation algorithms is used, it is expected that more effective risk reduction alternatives will be provided through the removal of the uncertainty implied in the PSA model. KHNP expects that the improvement effect of this new quantification code could be present during the seismic PSA revision prior to other external events. Its application will be begun after additional applicability reviews using the PSA models of the domestic plants.

3. Conclusions

As seen in the process of the periodic safety review (PSR) of domestic nuclear power plants, the risk management objectives such as core damage frequency and large early release frequency are not easy to be met without continuous safety improvements and the integration of the improved technologies into the PSA evaluation methodologies.

To be prepared for this reinforced risk management, KHNP should apply the accident response equipment, which is being installed as Fukushima countermeasures according to their short-term and long-term schedules, and the procedures related to the PSA. In addition, the uncertainties due to the conservative assumptions or expert judgments must be reduced using the development and application of PSA analysis methods that could refine the results in order that they approach the actual event conditions.

In conclusion, because external event analyses have a large portion of uncertainty factors in the current analysis methodologies, the technical efforts in various perspectives, as well as safety improvements including the installation of the accident response hardware, need to be implemented in order to explain nuclear safety to the public and to manage the licensing activities for continuous operation of nuclear power plants or new plant construction.

References

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