Development of a Regulatory Model for Seismic Probabilistic Safety Assessment

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Abstract: Through the Mid- and Long-term Program for Nuclear Research and Development conducted from 2007 to 2012, efforts were made to develop regulatory risk models referred to as MPAS (Multi-Purpose probabilistic Assessment of Safety) for all types of nuclear reactors operated in Korea. However, the MPAS model's scope was limited to Level 1 internal events. In this study, we have developed a Seismic PSA (Probabilistic Safety Assessment) model for OPR-1000 and APR-1400 reactors based on the pre-developed Level 1 internal event MPAS models. Following the SPRA process described in EPRI (Electric Power Research Institute) Seismic Probabilistic Risk Assessment Implementation Guide, we have incorporated the results of site-specific seismic hazard and vulnerability assessments, considering site characteristics. The seismic acceleration range was divided into four segments, and the excess occurrence frequency for each acceleration range was determined. These ranges influence variations in HRA (Human Reliability Analysis) and seismic induced equipment failure probability values. For model quantification, we utilized AIMS-PSA for fault tree and event tree analysis, and ARES (Advanced Risk assessment program considering Earthquake Scenario) for Seismic PSA quantification.

Keywords: Seismic PSA, Seismic PRA, Regulatory risk model

1. Introduction

In 2015, the amendment of the Nuclear Safety Act in Korea mandated the inclusion of Probabilistic Safety Assessment (PSA) results in the Accident Management Program (AMP). This change highlighted the need of independent PSA models for regulators to verify the operator's assessment. Efforts to address this need had been underway even before the amendment. Through the Mid- and Long-term Program for Nuclear Research and Development conducted from 2007 to 2012, the Korea Atomic Energy Research Institute (KAERI) and the Korea Institute of Nuclear Safety (KINS) developed regulatory risk models referred to as MPAS (Multi-Purpose probabilistic Assessment of Safety) for all types of nuclear reactors operated in Korea [1]. After 2018 models for APR-1400 were also developed [2].

Enhancements were made to the MPAS models based on these representative models, incorporating the specific characteristics of each site unit. However, the currently available MPAS models are restricted to internal events, with no models addressing external events. Consequently, this study developed an MPAS model for seismic events, one of the external events, for a reference site, and performed a probabilistic safety assessment applying this model to the reference site.

2. Methodology

To develop the seismic event MPAS model, we utilized the procedures outlined in the Seismic Probabilistic Risk Assessment (SPRA) Implementation Guide [3]. As shown in Figure 1, the SPRA Implementation Guide includes an SPRA flowchart.

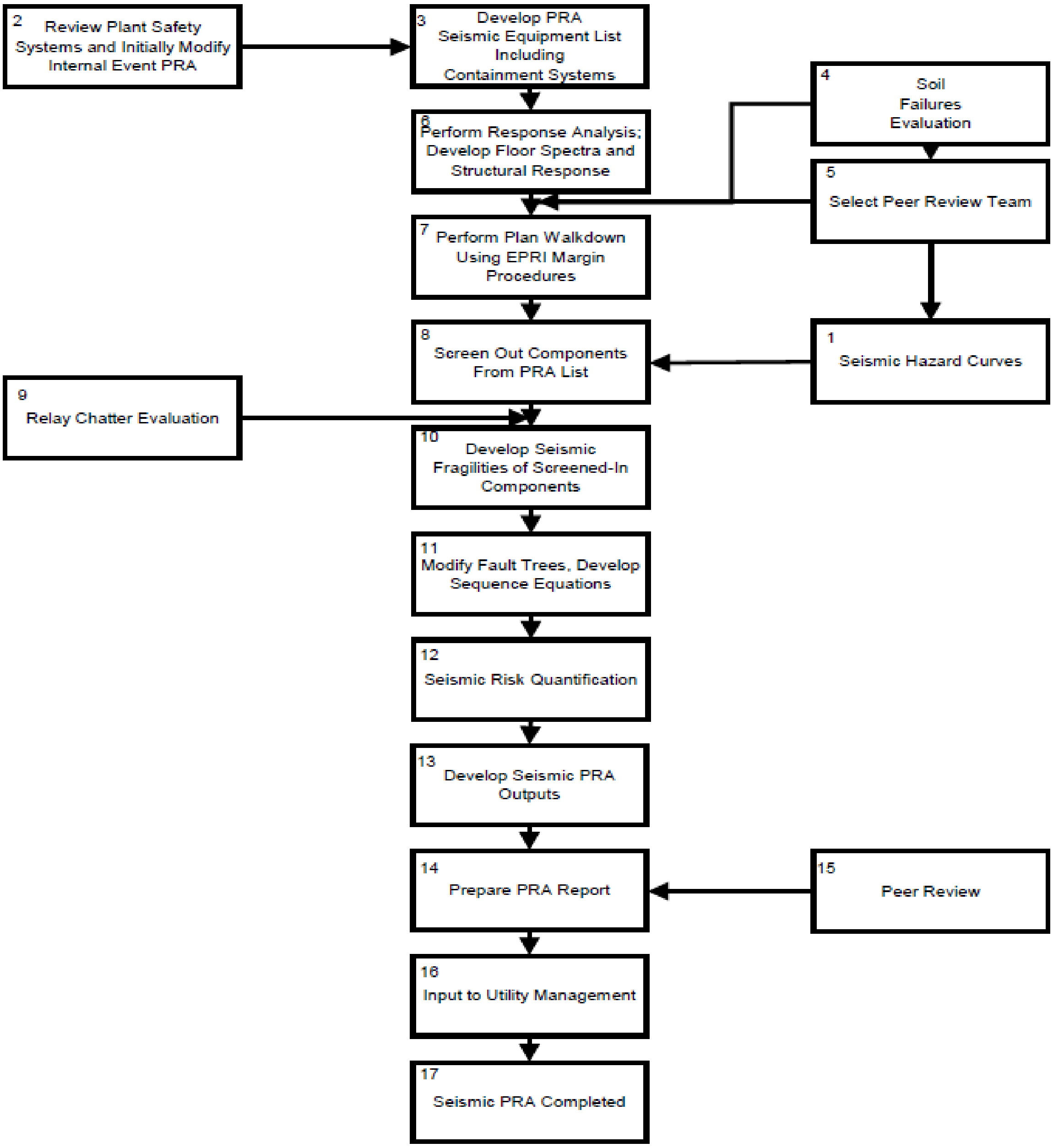

Figure 1. SPRA flowchart [3]

Ideally, every step in Figure 1 should be followed to perform a comprehensive assessment. To derive accurate values, additional analyses, such as seismic walkdowns, are necessary for variables like the Seismic Equipment List and Seismic Fragilities. However, due to limitations in analysis capabilities and data availability, the values provided by the operator have been utilized. Out of the 17 steps, we performed analysis on steps 11, 12, and 13 to develop the PSA model.

As mentioned earlier, the development of the seismic event MPAS model requires the internal event MPAS Model. For the seismic model development, equipment modeled for internal events but unsuitable for seismic conditions was selectively removed, and Event Trees (ET) for seismic-induced initiating events were developed. The selection of each initiating event was based on the failure mode and effect analysis (FMEA) of screen-in equipment from the pre-analyzed Seismic Equipment List (SEL). The event trees were developed in two stages: the first-stage event tree which is the primary event tree was constructed in the order of expected contribution to the Core Damage Frequency (CDF) from each seismic-induced initiating event, and the second-stage event tree which is the secondary event tree was constructed for initiating events requiring mitigation measures.

Branches for seismic-induced initiating events that could not be mitigated in the primary event tree included the initiating equipment and, if necessary, related Human Reliability Analysis (HRA) events. Initiating events that could be mitigated and therefore considered in the secondary event tree were matched with the relevant branches in the modified Fault Trees (FT), FT's from the internal event model, for seismic analysis. Screen-in equipment which do not directly initiate an event but impacted accident mitigation due to failure was directly incorporated into the FT.

Finally, the developed model was quantified using the ARES (Advanced Risk Assessment Program considering Earthquake Scenarios) code developed by the Korea Atomic Energy Research Institute (KAERI).

3. Model Development and Quantification

3.1. Reference Site

"Hanul" site, where Framatome, OPR-1000, and APR-1400 types of reactors are in operation in Uljin, Korea is selected as the reference site. Because of limited scope of this study, the analysis on Framatome is not conducted but models for OPR-1000 and APR-1400 were developed and compared.

3.2. Seismic Hazard Curves

The seismic hazard assessment utilized the latest seismic hazard data, reflecting the most recent domestic earthquake history (including the Gyeongju and Pohang earthquakes) and the results of the Ministry of the Interior and Safety's 4th phase investigation project (2017-2021) [4]. Compared to the seismic hazard data in the existing Final Safety Analysis Report, the reevaluated seismic hazard data showed higher estimates for ranges below 0.2g but lower estimates for ranges exceeding 0.2g. The 0.2g threshold is also the design basis ground acceleration for operating nuclear power plants, indicating a reduced frequency of exceedance in the segment exceeding the design basis earthquake. Table 1 below details the mean, 15th, 50th, and 85th percentile values for Peak Ground Acceleration (PGA) at the reference site.

Table 1. Seismic hazard for the reference site

PGA(g)	15th percentile	50th percentile	85th percentile	Hazard (mean)
0.01	5.38E-03	1.33E-02	3.14E-02	1.68E-02
0.05	2.90E-04	1.05E-03	3.50E-03	1.80E-03
0.1	5.21E-05	2.38E-04	1.05E-03	5.23E-04
0.2	3.11E-06	2.85E-05	1.69E-04	8.81E-05
0.3	3.38E-07	5.47E-06	3.82E-05	2.29E-05
0.4	3.20E-08	1.43E-06	1.23E-05	7.89E-06
0.5	3.07E-09	4.21E-07	4.71E-06	3.25E-06
0.6	3.59E-10	1.35E-07	2.05E-06	1.49E-06
0.75	1.91E-11	2.58E-08	6.94E-07	5.28E-07
1	1.64E-13	1.81E-09	1.24E-07	1.12E-07

3.3. Seismic Fragilities

The seismic fragility used in this analysis was selected based on a review of seismic fragility data for each reactor type at the Hanul site. Previous seismic PSAs for severe accident policy and accident management program used the seismic PSA advisory report for Kori Units 3 and 4 as a basis for seismic fragility assessment. The screening criteria for SEL equipment were applied as follows in each case.

- High Seismic Capacity Equipment: Am > 1.5g
    - Equipment was screened out based on the High Confidence of Low Probability of Failure (HCLPF) values from the site-specific seismic hazard analysis.
- Accident Management Program Seismic PSA

- Based on the PSHA results for the Hanul site, equipment with a median seismic capacity greater than 1.5g Peak Ground Acceleration (PGA) was screened out.

Based on the review results for each case, the seismic fragility results from the accident management plans were applied to the reference reactor types OPR-1000 and APR-1400. The screened-in components for each reference reactor type are presented in Table 2.

Table 2. Screened-in components for reference reactors

Unit	Component	FMEA
OPR-1000	Safety Injection Tank	Loss of SITs
	Instrument Tube (Primary System)	Small LOCA
	125V DC Cabinet	Loss of Essential Power
	Battery Rack	Loss of Essential Power
	Battery Charger	Loss of Essential Power
	Inverter - Structural	Loss of Essential Power
	Inverter - Functional	Loss of Essential Power
	Regulating Transformer	Loss of Essential Power
	480V Load Center	Loss of Essential Power
	4.16kV SWGR	Loss of Essential Power
	ILS Cabinets	Loss of Control
	ESW Travelling Screen	Loss of CCW/Chilled Water System
	Essential Chiller	Loss of CCW/Chilled Water System
	ECW Compression Tank	Loss of CCW/Chilled Water System
	Loss of Off-Site Power	Loss of Off-Site Power
APR-1400	4.16kV SWGR (Functional)	Loss of Essential Power
	Emergency Diesel Generator	Loss of Essential Power
	Safety Injection Tank	Large LOCA
	Instrument Tube (Primary System)	Small LOCA
	Loss of Off-Site Power	Loss of Off-Site Power

3.4. Seismic Event Tree Development

The primary seismic event trees for each reactor type were developed using the FMEA for the screened-in equipment. For each initiating event derived from the FMEA results, the headings were organized in descending order of their contribution to core damage. If the probability value of a preceding branch was sufficiently large, the success branch of the following heading was removed, leaving only the failure case to simplify the event tree. Each branch logic in the event tree consists of combinations of basic events related to the equipment causing the initiating events.

For initiating events where mitigation measures cannot be performed, direct core damage was considered. For events where mitigation measures are feasible, the event tree was structured to transfer to secondary event trees. The secondary event trees were based on existing internal event trees, with modifications to remove mitigation measures unavailable during seismic events (e.g., non-seismic equipment like AAC DG). Figure 2 illustrates the event trees for the Loss of Offsite Power (LOOP) initiating event for both internal and seismic events as an example.

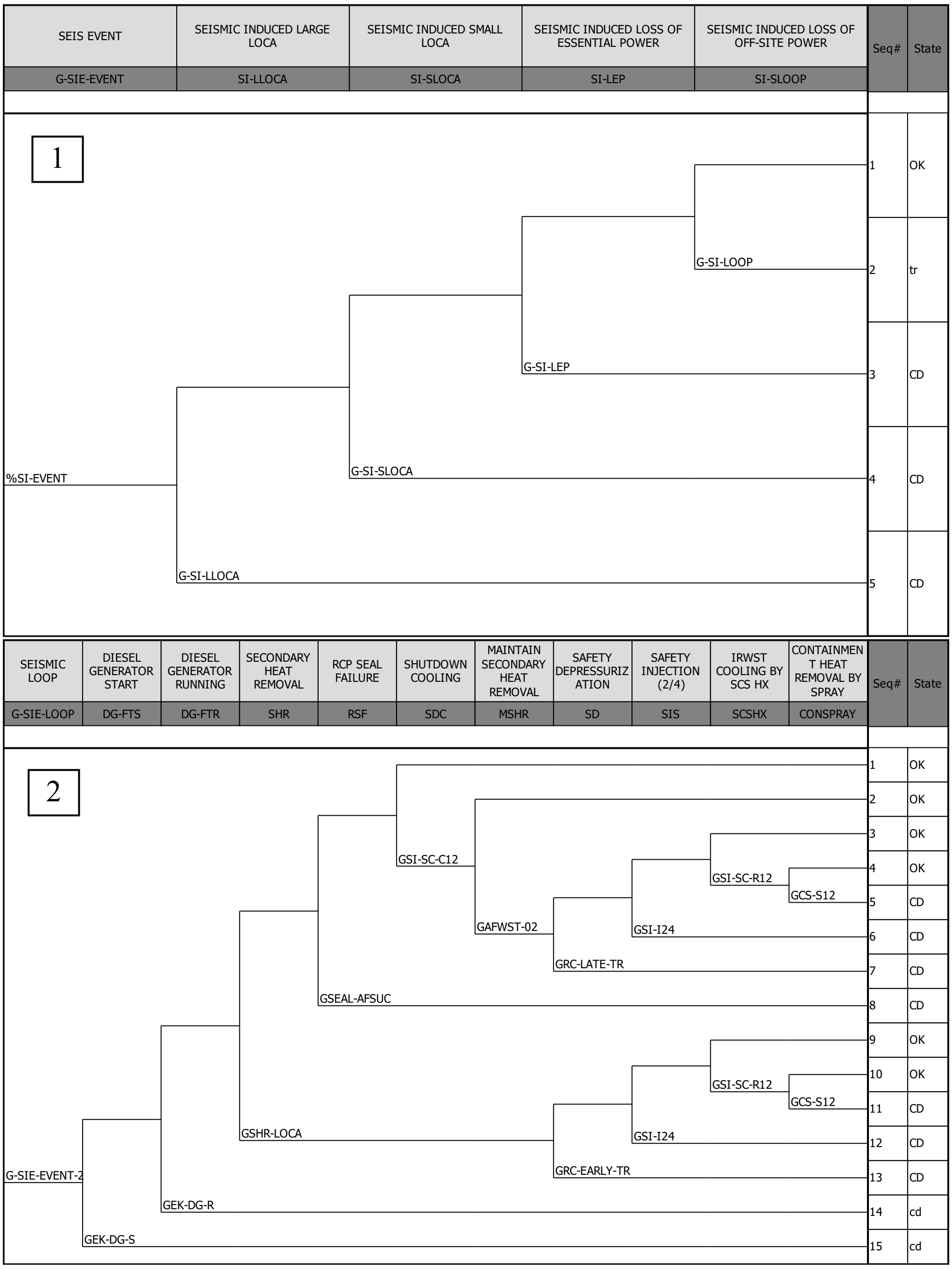

Figure 2. 1) Primary and 2) Secondary event tree for APR-1400

3.5. Quantification

To quantify the seismic PSA model, we utilized the ARES code. The inputs required for quantification include the seismic hazard curve, seismic binning, fragility values for the target equipment, seismic Human Reliability Analysis (HRA) results, and seismic fault trees. The seismic hazard curve used for inputs was divided into four damage states over the range of 0.1g to 1.0g. For the OPR-1000, the first bin was set at 0.2g, corresponding to the Safety Shutdown Earthquake (SSE) criteria, while for the APR-1400, the first bin was set at 0.3g. The remaining bins were defined in Table 3.

Table 3. Seismic Bins for reference plants

Seismic BIN	Range		Criteria
	APR-1400	OPR-1000	
Bin1	0.1g-0.3g	0.1g-0.2g	Up to the plant's SSE
Bin2	0.3g-0.4g	0.2g-0.3g	Damage to non-safety equipment and the lowest PGA value for safety-related equipment and structure
Bin3	0.4g-0.5g	0.3g-0.5g	Widespread damage to non-safety-related equipment and structures
Bin4	0.5g-1.0g	0.5g-1.0g	Widespread damage to both safety and non-safety-related equipment and structures

The seismic hazard curve and seismic fragility values used were from previous analyses, while the seismic HRA was conducted using the EPRI's approach [5] for HRA events considered in the seismic model.

4. Results

The quantification results showed that the Core Damage Frequency (CDF) of the OPR-1000 at the reference site was approximately 13% higher than APR-1400. For both reactor types, the highest CDF was found in Bin 4. The seismic bins, ranked in descending order of CDF contribution, are Bin 4, Bin 3, Bin 1, and Bin 2 for APR-1400, while for OPR-1000, they were Bin 4, Bin 1, Bin 3, and Bin 2, indicating differences in Bins 3 and 1. In both reactor types, the proportion of CDF in Bin 1 was dominated by LOOP, with 97% for APR-1400 and 99% for OPR-1000. This is attributed to the high probability of LOOP occurrence even in low acceleration range, due to the low fragility of the transmission towers causing LOOP. Excluding LOOP, the other initiating events demonstrated higher CDF percentiles in higher seismic bins.

In overall CDF trends, the proportion of CDF due to loss of power events (LOOP and LEP) was high, with 81% for APR-1400 and 72% for OPR-1000. This aligns with the logic of the primary seismic event tree, so an importance analysis was conducted on minimal cutsets excluding the power loss events to identify other major factors. The importance analysis, based on Fussell-Vesely importance, identified the top five events excluding the power loss events. For the APR-1400, the highest was seismic-induced structural failure of the Safety Injection Tank (F-V: 0.1733), followed by RCP Seal failure (F-V: 0.0391) and loss of control of the ESW pump due to the failure of the digital output module (F-V: 0.0015). For the OPR-1000, the highest was seismic induced structural failure of the essential chiller (F-V: 0.0748), followed by seismic induced structural failure of the ECW compression tank (F-V: 0.0412), seismic induced structural failure of the ESW traveling screen (F-V: 0.0345), and seismic induced structural failure of the ILS cabinet (F-V: 0.0281). The overall CDF results of the APR-1400 and OPR-1000 reference plants are illustrated in Table 4 and 5 respectively.

Table 4. CDF fractions by initiating event for APR-1400 reference plant

Seismic BIN	CDF fraction				
	SI-LOOP*	SI-LEP*	SI-SLOCA*	SI-LLOCA*	Total
BIN1	83.00%	1.39%	0.07%	9.98%	26.98%
BIN2	9.84%	9.44%	3.09%	17.55%	8.34%
BIN3	3.87%	13.41%	7.52%	16.94%	9.24%
BIN4	3.29%	75.77%	89.33%	55.53%	55.44%
TOTAL	100.00%	100.00%	100.00%	100.00%	100.00%

Table 5. CDF fractions by initiating event for OPR-1000 reference plant

Seismic BIN	CDF fraction						
	SI-LOOP	SI-LOCCW*	SI-LOC*	SI-LEP	SI-SLOCA	SI-LLOCA	Total
BIN1	47.16%	0.39%	0.07%	0.26%	0.00%	1.66%	19.74%
BIN2	30.73%	4.91%	2.21%	5.67%	0.47%	11.76%	15.69%
BIN3	17.84%	27.15%	20.39%	33.57%	10.36%	38.51%	24.75%
BIN4	4.27%	67.56%	77.33%	60.50%	89.16%	48.07%	39.82%
TOTAL	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%	100.00%

*SI: Seismic Induced, LOOP: Loss Of Offsite Power, LEP: Loss of Essential Power, SLOCA: Small Loss Of Coolant Accident, LLOCA: Large Loss Of Coolant Accident, LOCCW: Loss Of Component Cooling Water, LOC: Loss Of Control

5. Conclusion

In this study, we developed seismic event MPAS models for the APR1400 and OPR1000 reactor types using the pre-existing Level 1 internal MPAS models. The procedures from the EPRI SPRA Implementation Guide were utilized for the development of the MPAS models, focusing on event tree and seismic fault tree development, seismic risk quantification, and seismic PSA output. The seismic event trees were developed in two stages. The primary event tree was organized with seismic-induced initiating events in the order of their expected contribution to the Core Damage Frequency (CDF). The secondary event tree was developed if the mitigation measures are feasible. Each branch of the primary event tree included events for screened-in equipment specific to each reactor type, and HRA events were simulated as needed.

For the quantification of the seismic model, we used the ARES code developed by KAERI. The CDF was calculated by dividing the 0.1g to 1.0g range into four bins. The quantification results indicated that the CDF of the OPR1000 was approximately 13% higher than that of the APR1400. This is likely due to the higher number of screened-in equipment items that are vulnerable to seismic events in the OPR1000. Additionally, in Bin 1 (0.1g to 0.2g/0.3g), the frequency of LOOP was high for both APR1400 and OPR1000, at 81% and 72% respectively. This high frequency is attributed to the low fragility of the transmission towers that cause LOOP, resulting in a high failure probability in lower seismic bins. Excluding power loss events, the most critical event for APR1400 was the seismic induced structural failure of the safety injection tank due to seismic activity, and for OPR1000, it was the seismic induced structural failure of the essential chiller.

The MPAS models developed in this study can be used as regulatory verification models for individual OPR-1000, APR-1400 units respectively and can also serve as the base seismic model for ongoing multi-unit PSA research.

Acknowledgements

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