An Initiating-Event Analysis for PSA of Hanul Units 3&4: Results and Insights

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1. Introduction

Recently, the at-power internal events Level 1 probabilistic safety assessment (PSA) for the Hanul units 3 and 4 was updated [1]. As a part of the PSA, an initiating-event (IE) analysis was newly performed by considering the current state of knowledge and the requirements of the ASME/ANS probabilistic risk assessment (PRA) standard related to IE analysis [2].

This paper describes the methods of, results and some insights from the IE analysis for the PSA of the Hanul units 3 and 4.

2. Methods

Table I summarizes the differences of the methods and data used between the previous PSA for the Hanul units 3 & 4 [3] and this study. In comparison with the previous IE analysis, the current IE analysis performed a more systematic and detailed analysis to identify potential initiating events. In addition, the current analysis used the latest data for calculating IE frequencies.

Table I: Summary of the Differences between the Previous and Current IE Analysis

Task	Previous IE Analysis [3]	Current IE Analysis
1. IE Identification	- Develop a Master Logic Diagram - Review the existing IE classification	- Review the existing IE classification - Analyze the domestic industry experience - Perform an FMEA - Analyze multi-unit IEs
2. Data		
2.1 Generic	- EPRI ALWR URD [4] - NUREG/CR-3862 [5]	- NUREG/CR-6928 [6] - NUREG-1829 [7]
2.2 Plant-specific (Korean industry)	- Plant experiences of 4 units (~2002)	- Plant experiences of 20 units (1993~2012)
3. Calculation of IE Frequencies		
3.1 LOCAs	- Generic data	- Generic data
3.2 Transients	- Bayesian update of generic data	- Domestic industry data only
3.3 ISLOCA	- Integral equations - Screening out of high-pressure, isolable or < 1 inch lines	- Fault tree modeling - No screening out of high-pressure, isolable or >= 3/8 inch lines

2.1 Identification and Grouping of Initiating Events

Fig. 1 shows the process of the identification and grouping of potential initiating events, which was used in this study.

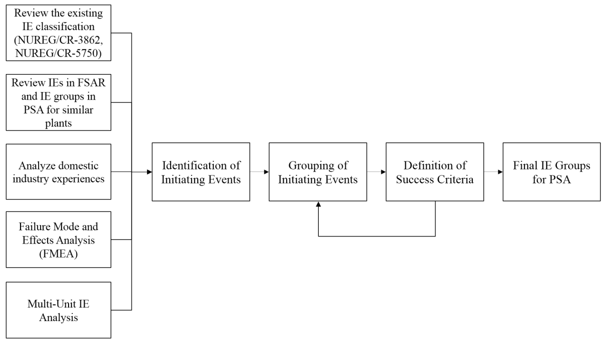

Fig. 1. The process of the identification and grouping of initiating events.

Firstly, potential initiating events were identified by thoroughly reviewing several documents (i.e., NUREG/CR-3862 [5], NUREG/CR-5750 [8], final safety analysis report [9], and PSA reports for similar nuclear power plants [10-12]), investigating Korean nuclear power plant operating experiences, and performing a failure mode and effects analysis (FMEA) for important components of each system and a multi-unit initiating event analysis.

Then, the potential initiating events were grouped based on their impacts on plant responses and the availability of safety-related mitigating systems. Finally, a total of 20 initiating event groups were selected for the at-power internal events Level 1 PSA as shown in Table II.

2.2 Calculation of Initiating Event Frequencies

The IE frequencies were calculated by the following five different approaches according to the event characteristics and data availability.

1) For IEs that have never occurred in Korean nuclear power plants (NPPs), the generic data [6-7] were used. Large, medium, small LOCA, RVR, MSLB-IC, and MSLB-OC belong to this category. To obtain frequencies consistent with each LOCA break size definition, LLOCA and MLOCA IE frequencies were adjusted by using

an interpolation approach, which was suggested by Eide, et al. [13].

Table II: Initiating Event Groups for At-Power Internal Events PSA of the Hanul Units 3&4

Category	IE Group	Remark*
Loss-of-Coolant Accident (LOCA)	1. Large LOCA (LLOCA)	
	2. Medium LOCA (MLOCA)	
	3. Small LOCA (SLOCA)	
	4. Steam Generator Tube Rupture (SGTR)	
	5. Interfacing System LOCA (ISLOCA)	
	6. Reactor Vessel Rupture (RVR)	
Transient	7. Main Steam Line Break: Inside CTMT (MSLB-IC)	Divided into 2 groups
	8. Main Steam Line Break: Outside CTMT (MSLB-OC)	
	9. Total Loss of Main Feedwater (LOFW)	
	10. Loss of Condenser Vacuum (LOCV)	
	11. Partial Loss of CCW System (PLOCCW)	
	12. Total Loss of CCW System (TLOCCW)	Added
	13. Loss of Class 1E 4.16kV Bus A (LOKV)	
	14. Loss of Class 1E 125V DC A (LODCA)	
	15. Loss of Class 1E 125V DC B (LODCB)	Added
	16. Loss of Offsite Power (LOOP)	
	17. Station Blackout: EDG fail to start (SBO-S)	
	18. Station Blackout: EDG fail to run (SBO-R)	Added
	19. General Transients (GTRN)	
	20. Anticipated Transient Without Scram (ATWS)	

* Differences from the previous PSA for the Hanul units 3&4

2) For IEs that have occurred at least once in Korean NPPs, the frequencies were estimated from the 20-year (1993~2012) Korean industry data. This category includes SGTR, LOFW, LOCV, PLOCCW, LODCA, LODCB, GTRN, and LOOP.

3) For some IEs, both the generic data and Korean industry data were used for calculating the IE frequencies. LOKV belongs to this category.

4) For ISLOCA and TLOCCW IEs, the frequencies were quantified from fault tree models developed by considering plant-specific design and operational characteristics.

5) For IEs transferred from the event trees of other initiating events, the IE frequencies were not calculated separately. SBO and ATWS belong to this category.

2.3 Interfacing System LOCA Analysis

For interfacing system loss-of-coolant accident (ISLOCA) analysis, the screening criteria for lines interfacing with the reactor coolant system (RCS) were modified from the criteria used in the previous IE analysis (cf. [14]). This study did not screen out the lines with high design pressure (> 2,000 psig), lines that can be isolated from the RCS by closing an open valve, and lines with diameters of 3/8 inch or more.

The frequency of ISLOCA was calculated by fault tree modeling. The current ISLOCA IE fault tree model was modified from the model which the authors had developed previously [14], so the ISLOCA frequency through each potential pathway was also changed.

In addition, the state of knowledge correlation (SOKC) that exists between two or more components with the same data was considered. According to the ASME/ANS PRA standard [2], the effect of the SOKC has been found to be significant particularly in calculating the ISLOCA frequency involving the rupture of multiple valves.

For ISLOCA paths with diameters less than 1 inch, the occurrence of the ISLOCA through the paths was not assumed to lead directly to core damage. In this study, CCW supply lines to and return lines from RCP high pressure coolers belong to this case. For these lines, an event tree was developed, which is similar to the event tree for small LOCA.

3. Results and Insights

Table III shows the IE frequencies for the Hanul units 3 and 4. It compares the current IE frequencies with the previous IE frequencies. Each frequency was multiplied by an assumed average criticality factor of 0.95 to obtain a frequency per reactor calendar year (/rcy) from a frequency per reactor critical year (/rcry). Some IE frequencies such as SGTR and LODCA are similar to the previous frequencies, but not a few IE frequencies are quite different from the previous ones.

Table III: Comparison of the IE Frequencies of the Hanul Units 3&4

IE Group	Previous Freq. (/rcy) [3]	Current Freq. (/rcy)
LLOCA	1.70E-4	2.89E-6
MLOCA	1.70E-4	1.62E-4
SLOCA	3.00E-3	3.49E-4
SGTR	4.50E-3	4.92E-3
ISLOCA (SCS suction lines)	1.77E-9	5.29E-9
ISLOCA (Letdown line)	N/A	4.52E-9
ISLOCA (CCW supply/return lines from RCP HP coolers)	N/A	1.76E-8
RVR	2.66E-7	3.44E-8
MSLB-IC	1.50E-3	3.49E-4
MSLB-OC		7.32E-3
LOFW	1.86E-1	4.10E-2
LOCV	1.01E-1	7.38E-2
PLOCCW	2.41E-1	4.92E-3
TLOCCW	N/A	2.12E-4
LOKVA	1.22E-3	4.31E-3
LODCA	2.44E-3	2.46E-3
LODCB	N/A	2.46E-3
LOOP (critical operation)	2.20E-2	2.36E-2
LOOP (shutdown operation)	N/A	1.74E-1*
GTRN	1.45	7.06E-1

* frequency per shutdown year; It should be multiplied by 0.05 to obtain the frequency per reactor calendar year.

The results of this study provide some insights into the IE analysis as a part of PSA for Korean nuclear power plants including the Hanul units 3 and 4. Firstly, Korean industry experience (Korean generic data) is now sufficient to estimate the frequencies for most IEs, especially for transients. It means that the Bayesian updating of generic data (in general, U.S. industry data) with Korean industry data or plant-specific data is not necessary for calculating the IE frequencies in most cases. Rather, the Bayesian updating of Korean generic data with plant-specific data (e.g., Hanul 3&4 data) is needed. The IE frequencies calculated by using only Korean industry data in this study are not much different from those obtained from the U.S. industry data (NUREG/CR-6928 [6]).

Secondly, for some IEs, there are two or more approaches to calculate the IE frequencies. For example, TLOCCW IE frequency can be obtained from fault tree modeling or plant operating experience. In this case, the frequencies obtained from different approaches need to be compared with the frequencies of other IEs and with the corresponding IE frequencies in similar plants, and then the most appropriate one should be selected.

In addition, the results of this study revealed that not only shutdown cooling suction lines but also CVCS letdown line and CCW supply/return lines from RCP high pressure coolers are important when considering the risk of ISLOCA in the Hanul units 3 and 4. This can be applied to other OPR1000 plants.

4. Conclusions

In this study, as a part of the PSA for the Hanul units 3 and 4, an initiating-event (IE) analysis was newly performed by considering the current state of knowledge and the requirements of the ASME/ANS probabilistic risk assessment (PRA) standard. In comparison with the previous IE analysis, this study performed a more systematic and detailed analysis to identify potential initiating events, and calculated the IE frequencies by using the state-of-the-art methods and the latest data.

As a result, not a few IE frequencies are quite different from the previous frequencies, which can change the major accident sequences obtained from the quantification of the PSA model. Moreover, the results of this study provide some insights into the IE analysis as a part of PSA for Korean nuclear power plants including the Hanul units 3 and 4.

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