Estimation of CDF Considering the Degradation Effect of the Condensate Storage Tank

Daegi Hahm [a*], Young-Sun Choun [a], In-Kil Choi [a], Jinsuo R. Nie [b], Joseph I. Braverman [b]
[a]*Korea Atomic Energy Research Institute, Daedeok-Daero 989-11, Yuseong-Gu, Daejeon*
[b]*Brookhaven National Laboratory, 32 Lewis RD., Upton, L.I., NY 11973-5000, USA*
[*]*Corresponding author: dhahm@kaeri.re.kr*

1. Introduction

This paper presents a part of the research results of the 5-year BNL-KAERI collaboration program, the goal of which is to develop a realistic seismic risk evaluation system which includes the consideration of aging of structures and components in nuclear power plants (NPPs). A condensate storage tank (CST) located at the Ulchin NPP of Korea was selected to demonstrate the methodology. In U.S. NRC Regulatory Guide (RG) 1.174, Rev. 2 [1], "An Approach for Using Probabilistic Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," defines the acceptable level of changes of NPP in terms of risk, i.e., the core damage frequencies (CDFs). Hence, in this paper, we performed the probabilistic safety assessments to obtain the CDFs for a series of fragility capacities of degraded CST.

2. Methods and Results

The Ulchin NPP units 5&6, where the CST is located, were selected for its PSA to estimate the CDFs for various HCLPF values and uncertainties. The scope of the PSA included internal and external PSAs. Except for the degradation of the CST, the PSA model did not include the degradation of other structures, systems, and components (SSCs). The total CDF was defined as the summation of the CDFs induced by internal events and external events such as earthquakes, fire, and flood. A brief description of the seismic PSA procedure is provided below, as the degradation of the CST was taken into account in the seismic PSA.

To facilitate the seismic PSA, many structures and components were screened out based on a detailed walkdown, their generic fragility data, and fragility analyses. More specifically, those structures and components that have a HCLPF capacity larger than 0.65g were screened out based on the guidelines. As a result, 18 seismic risk significant components were kept in the seismic PSA model after the screening, as shown in Table together with their median (Am) and HCLPF capacities and uncertainties.

The components listed in Table 1 were used for the failure mode and effect analysis (FMEA) of the selected plant. The FMEA can identify initiating events that are caused by the seismic-induced failures of the structures and components. For this study, the following seismic-induced initiating events were identified through the FMEA:

- Loss of Essential Power (LEP)
- Loss of Secondary Heat Removal (LHR)
- Loss of Component Cooling Water (LCCW)
- Small Loss of Coolant Accident (SLOCA)
- Loss of Off-site Power (LOOP)

Table 1. Capacities & Uncertainties of Critical Components

Component	Am (g)	U	R	HCLPF (g)
Offsite Power	0.30	0.22	0.20	0.15
Battery Charger	1.03	0.28	0.28	0.41
Switch	2.33	0.41	0.45	0.55
4.16kV Switchgear	1.33	0.33	0.29	0.48
Inverter	1.37	0.33	0.30	0.49
Battery Rack	1.46	0.33	0.31	0.51
480V AC Load Center	1.50	0.32	0.29	0.57
125V DC Control System	1.58	0.33	0.29	0.57
Regulating Transformer	1.30	0.33	0.30	0.46
Condensate Storage Tank	1.29	0.27	0.20	0.55
Diesel Generator	1.03	0.28	0.28	0.41
ECW Compression Tank	2.33	0.41	0.45	0.55
ECW Chiller	1.33	0.33	0.29	0.48
ECW Pump	1.37	0.33	0.30	0.49
ESW Pump	1.46	0.33	0.31	0.51
CCW Surge Tank	1.50	0.32	0.29	0.57
Instrumentation Tube	1.50	0.30	0.30	0.56
HVAC Ducting and Supports	2.06	0.32	0.41	0.62

Fig. 1 shows the event tree of the above initiating events, where PDS means the plant damage state, TR the damage to be transferred to the internal system and CD the core damage. Fig. 2 shows the components grouped by their corresponding seismic-induced initiating events. Fault trees were constructed for each seismic-induced initiating event.

From the event tree and fault trees, the Boolean equations for the seismic-induced initiating events can be derived. Finally, the initiating event frequency and the corresponding core damage frequency can be determined by solving the Boolean equations or transferring the initiating event to the internal PSA process in the case of LOOP. The computer codes PRASSE (Probabilistic Risk Assessment of Systems for Seismic Events) [2] and AIMS-PSA (Advanced Integrated Management System for PSA) [3] were used for seismic PSA and internal PSA, respectively.

The seismic-induced initiating event frequencies and CDFs for an undegraded case (baseline) are summarized in Table 2, in which the total seismic-induced CDF is identified to be 6.00E-6. According to the internal and external PSA reports for the Ulchin NPP units 5&6, the CDFs due to internal events and fire are 5.65E-6 and 2.30E-6, respectively. The flood-induced CDF is reported to be less than 1.0E-12, and is negligible in the total CDF compared to other internal or external CDFs. Therefore, the total baseline CDF was calculated to be 1.395E-5.

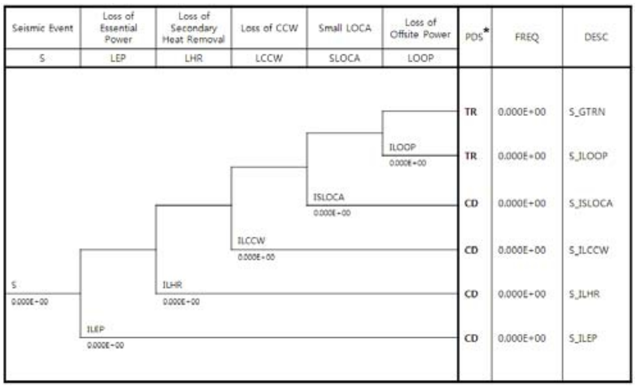

Fig. 1. Event Tree for Seismic Event

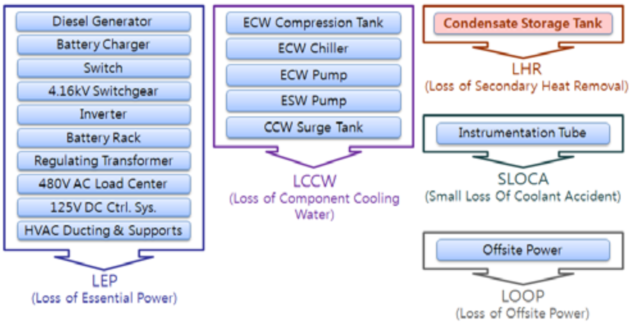

Fig. 2 Seismic-Induced Initiating Events and Their Components

The CDFs and ΔCDFs for degraded CST were estimated by the same procedure as for the baseline CDF calculation, and are shown in Table 3. for the constant uncertainty case and the varying uncertainty case. The ΔCDF values versus the baseline CDF are plotted in Fig. 3., overlapping the risk acceptance regions as prescribed in the U.S. RG 1.174, Rev. 2. The degradation acceptance criteria in terms of HCLPF is 0.422g for ΔCDF of 1.0E-6, which means the Region III.

Table 2. Seismic-Induced Initiating Event Frequencies & CDFs

Initiating Events	Frequency	Seismic-Induced CDF
LEP	2.64E-6	2.64E-6
LHR	2.65E-7	2.65E-7
LCCW	1.64E-6	1.64E-6
SLOCA	2.74E-8	2.74E-8
LOOP	4.95E-5	6.12E-7
GTRN	6.54E-4	8.16E-7
Total Seismic-Induced CDF		6.00E-6

Table 3. CDF and ΔCDF for the Cases of Constant and Varying Uncertainties

Cases	HCLPF (g)	Constant Uncertainty		Varying Uncertainty	
		CDF (E-5.0)	ΔCDF (E-7.0)	CDF (E-5.0)	ΔCDF (E-7.0)
Base case	0.549	1.395	-	1.395	-
	0.500	1.413	1.834	1.415	2.036
	0.444	1.460	6.520	1.460	6.570
	0.392	1.553	15.82	1.541	14.68
	0.340	1.724	33.00	1.640	24.58
Sliding to Overturning	0.300	1.952	55.79	1.784	39.00
	0.287	2.053	65.83	1.830	43.52
	0.235	2.752	135.8	2.154	75.98
	0.183	4.274	288.0	2.869	147.5
	0.130	8.212	681.8	4.592	319.8
Free Standing Tank	0.100	13.64	1224	6.966	557.2
Minimum case	0.078	21.87	2047	9.990	859.6

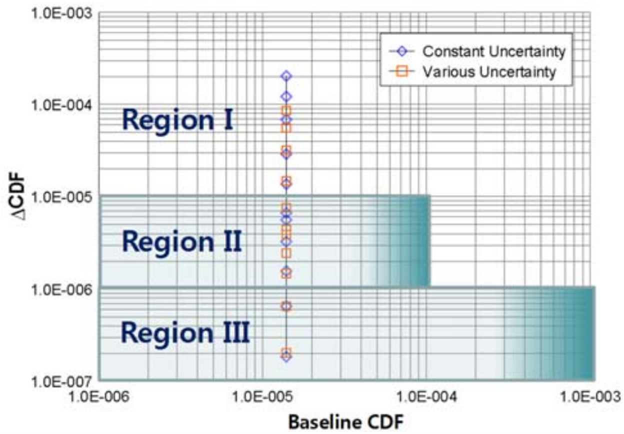

Fig. 3. ΔCDF and NRC RG 1.174 Risk Acceptance Guideline

3. Conclusions

This paper presents the results of the probabilistic safety assessments to obtain the CDFs for a series of fragility capacities of degraded CST. Based on the acceptance risk level in the U.S. RG 1.174, Rev. 2, The degradation acceptance criteria in terms of HCLPF is defined by 0.422g while the undegraded HCLPF is 0.549g.

REFERENCES

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