Plant-Specific Uncertainty Analysis for a Severe Accident Pressure Load Leading to a Late Containment Failure

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Abstract: Typical containment performance analyses for a level 2 probabilistic safety analysis (PSA) have made use of a containment event tree (CET) modeling approach, to model the containment responses by depicting the various phenomenological processes, containment conditions, and containment failure modes that can occur during severe accidents. A general approach in the quantification of the containment event tree is to use a decomposition event tree (DET) to allow a more detailed treatment of the top event. A quantification of the physical phenomena in the decomposition event tree is achieved based on the results obtained through validated code calculations or expert judgments. The phenomenological modeling in the event tree still entails a high level of uncertainty because of our incomplete understanding of reactor systems and severe accident phenomena. This paper includes an uncertainty analysis of a containment pressure behavior during severe accidents for the optimum assessment of a late containment failure model of a decomposition event tree.

Keywords: Uncertainty Analysis, PSA, MAAP Code, Containment Pressure Load.

1. INTRODUCTION

A level 2 probabilistic safety analysis (PSA) is used to assess the performance of the containment in mitigating severe accidents. The analysis includes an evaluation of the accident progression in the containment; an estimation of the timing, location, and mode of containment failure; and an estimation of the source term characteristics. Typical containment performance analyses have made use of a containment event tree (CET) modeling approach, to model the containment responses by depicting the various phenomenological processes, containment conditions, and containment failure modes that can occur during severe accidents. A level 2 PSA of an OPR-1000, which is the reference plant of this analysis, has made use of a CET modeling approach, where a general approach in the quantification of a small event tree is to use a decomposition event tree (DET) to allow a more detailed treatment of the top event. A quantification of the physical phenomena in the DET is achieved based on the results obtained by validated the code calculations or expert judgments. The phenomenological modeling in the event tree still entails a high level of uncertainty. Such uncertainty exists because of our incomplete understanding of reactor systems and severe accident phenomena.

This paper illustrates the application of a severe accident analysis code, MAAP [1], to the uncertainty evaluation of a late containment failure DET, which is one of the CET top events in the reference plant of this study. An uncertainty analysis of a containment pressure behavior during severe accidents has been performed for the optimum assessment of a late containment failure model. The MAAP code is a system level computer code capable of performing integral analyses of potential severe accident progressions in nuclear power plants, whose main purpose is to support a level 2 probabilistic safety assessment or severe accident management strategy developments. The code employs lots of user-options for supporting a sensitivity and uncertainty analysis. The present application is mainly focused on determining an estimate of the containment building pressure load caused by severe accident sequences. Key modeling parameters and phenomenological models employed for the present uncertainty analysis are closely related to in-vessel hydrogen generation, gas combustion in the containment, corium distribution in the containment after a reactor vessel failure, corium coolability in the reactor cavity, and molten-corium interaction with concrete.

2. ANALYSIS METHODOLOGY

The basic approach of this methodology is to 1) develop severe accident scenarios for which the containment pressure loads should be performed based on a level 2 PSA, 2) identify severe accident phenomena relevant to a late containment failure, 3) identify the MAAP input parameters, sensitivity coefficients, and modeling options that describe or influence the late containment failure phenomena, 4) prescribe likelihood descriptions of the potential range of these parameters, and 5) evaluate the code predictions using a number of random combinations of parameter inputs sampled from the likelihood distributions; in addition 6) the results have been summarized and displayed for the important output variables.

To quantify the uncertainties addressed in the MAAP code, a computer program, MOSAIQUE [2], has been applied, which was recently developed by the Korea Atomic Energy Research Institute. The program consists of fully-automated software to quantify the uncertainties addressed in the thermal hydraulic analysis models or codes. MOSAIQUE employs a methodology of sampling-based uncertainty analysis using thermal hydraulic or severe accident analysis codes [3][4][5]. The Korean standardized nuclear power plant, the OPR-1000, has been selected as a reference plant for this analysis.

2.1. Development of DET Scenarios for the late containment failure

A late containment failure is defined as a failure of the containment long after a reactor vessel failure. The time frame for a late containment failure begins many hours after the vessel has failed and continues to three days after accident initiation. Three days are considered enough to control the containment pressurization. The primary cause of a failure of the containment is the steam over-pressurization resulting from the loss of the containment heat removal. Containment heat removal can be achieved by the operation of recirculation sprays or fan coolers. The steam over-pressurization process is slow and it takes times to reach the containment failure pressure. The possibility of late containment failure due to a late hydrogen burn can also be considered. To evaluate the early containment failure, the total pressure inside the containment should be calculated. In addition to the base pressure and reactor coolant system (RCS) blow-down pressure, the pressurization owing to steam generation in the cavity, gas generation by molten corium-concrete interaction (MCCI), late hydrogen burn in the containment have been considered. Eventually, the probability of a containment failure and its failure mode will be calculated using the containment fragility curve, which is out of scope of this paper.

The first and second top headings of a late containment failure DET are the RCS pressure at the reactor vessel failure and the amount of corium remained in the cavity, respectively. The driving force of corium ejection out of the reactor cavity is the RCS pressure. It is known that corium can escape the reactor cavity when the RCS pressure exceeds a certain value. The higher the RCS pressure, the more corium that can be ejected out of the cavity. The third top event is the availability of secondary heat removal using a motor-driven auxiliary feedwater pump and main steam safety valves. The fourth concern of the DET is a reactor cavity condition. Three discretized regimes, which are ‘flooded’, ‘wet’, or ‘dry’, have been selected to represent the cavity condition. There is a water flow path from the containment sump level to the reactor cavity in the reference plant. This path allows the cavity to be flooded if the inventory of the refueling water tank is injected into the containment through the high pressure safety injection (HPSI) or the low pressure safety injection (LPSI) system. In the case of the HPSI or LPSI operation, the cavity condition is assigned as a ‘flood’. On the other hand, the inventory of a safety injection tank (SIT) is only injected into the cavity, and the cavity condition is allocated as ‘wet’. The fifth event is a containment pressure load due to a late hydrogen burn. The late hydrogen burn has a dependency on the cavity condition. If the cavity condition is ‘flooded’ or ‘wet’, the containment pressure load increase by the late hydrogen burn will be limited owing to the higher steam generation in the reactor cavity.

Nine scenarios were developed as DET scenarios of a late containment failure. The developed DET scenarios are shown in Fig. 1 and Table 1: three large loss of coolant (LOCA) initiated scenarios for the sequences of low reactor vessel pressure at vessel failure, three loss of offsite power (LOOP) initiated scenarios for the sequences of high reactor vessel pressure at vessel failure, and three LOOP initiated scenarios for cases of high reactor vessel pressure at vessel failure with secondary heat removal available. Three cavity flooded cases by LPSI or HPSI operation, three cavity wet cases by SIT operation, and three cavity dry cases are included.

Fig. 1: DET scenarios for uncertainty analysis of the pressure load for late containment failure

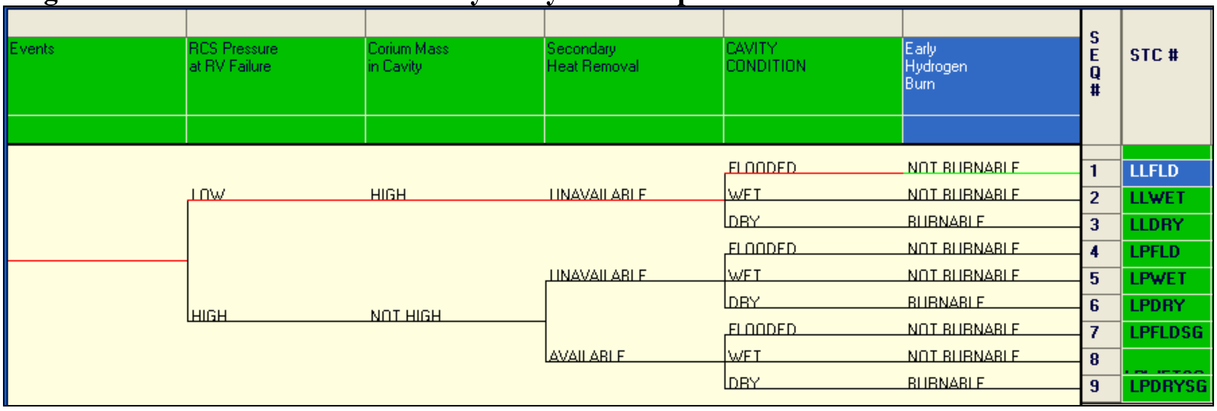

Table 1: Tabularized DET scenarios for an uncertainty analysis of the pressure load for late containment failure

RCS Pressure at RV Failure	Corium Mass in Cavity	Secondary Heat Removal	Cavity Condition	Late Hydrogen Burn	Sequence ID
Low	High	Unavailable	Flooded	Not Burnable	LLFLD
			Wet	Not Burnable	LLWET
			Dry	Burnable	LLDRY
High	Not High	Unavailable	Flooded	Not Burnable	LPFLD
			Wet	Not Burnable	LPWET
			Dry	Burnable	LPDRY
		Available	Flooded	Not Burnable	LPFLDSG
			Wet	Not Burnable	LPWETSG
			Dry	Burnable	LPDRYSG

2.2. Selection of MAAP Modeling Parameter and Sampling

In the severe accident analysis, there were uncertainties in the physical phenomena. There were also uncertainties in the MAAP phenomenological models. Users had control over the uncertainties through the so-called 'model parameters' of the MAAP program. They were either used as an input to a given physical model or to select between different physical models. This feature of the code architecture was included specifically to facilitate sensitivity or uncertainty in the analysis. In this study, input variables assigned as the model parameters to affect the pressure load of containment building during the late state of a severe accident were identified, and their uncertainty was characterized using a user specified distribution. These parameters were selected based on MAAP input parameter files.

For the present uncertainty analysis, 20 input variables were selected, which include six variables of steam and non-condensable gas generation, eight variables of in-vessel hydrogen generation, three variables of high pressure melt ejection, and three variables of hydrogen combustion in the containment. The list of variables and descriptions of the listed parameters were defined as shown in Table 2. The corresponding default values and uncertainty distributions of the parameters were defined as shown in Table 3. User assumption was given for the assigned range of modeling parameters and uncertainty distributions based on engineering judgments. To propagate these uncertain inputs through the MAAP code, they were sampled using a random sampling technique. The Monte Carlo Sampling method with a size of 200 for each scenario was used to sample the input parameter distributions, and 200 MAAP calculations were then performed.

Table 2: The list of parameters considered in the uncertainty analysis of pressure load for the late containment failure

Phenomena	MAAP Parameter	Parameter Description
Steam and Non-condensable gas generation in Cavity	HTCMCR	Downward heat transfer coefficient for convective heat transfer from molten corium to the lower crust in MCCI
	HTCMCS	Sideward heat transfer coefficient for convective heat transfer from molten corium to the side crust in MCCI
	TCNNP	Melting temperature of concrete
	FCHF	Flat plate critical heat flux Kutateladze number
	HTFB	Coefficient for film boiling heat transfer from corium to an overlying pool
	ACMPLB(1)	Floor surface area which the corium debris pool may occupy in cavity
High Pressure Melt Ejection	FKUTA	Kutateladze coefficient in the debris entrainment criterion
	FWEBER	Weber number used in the calculation of the diameter of the debris particles during the entrainment process
	ENT0C	Jet entrainment coefficient for the Ricou-Spalding correlation
In-vessel Hydrogen Generation	FAOX	Multiplier for the cladding outside surface area 새 calculate oxidation
	TCLMAX	Temperature to lead to rupture if the cladding is at this temperature for 0.01 hr. Larson-Miller parameter is calculated from TCLMAX
	LMCOL0	Collapse criteria parameters for a Larson-Miller-like functional dependence
	LMCOL1	
	LMCOL2	
	LMCOL3	
	EPSCUT	Cutoff porosity below which the flow area and the hydraulic diameter of core node are zero, i.e., the node is fully blocked
	EPSCU2	
Hydrogen Burn	TAUTO	Auto-ignition temperature for H_2 and CO burns
	XSTIA	Steam mole fraction required to inert an H_2-Air-H_2O mixture at incipient auto-ignition
	TJBRN	Temperature of H_2 jet entering a non-inerted compartment which is sufficient to cause a local burn

Table 3: The default values and uncertainty distributions of MAAP modeling parameters considered in the uncertainty analysis

MAAP Parameter	Default Value	Assigned Range [min-max]	Distribution
HTCMCR	3,500 W/m^2C	[1000, 5,000]	Triangle
HTCMCS	3,000 W/m^2C	[1000, 5,000]	Triangle
TCNNP	1,450 K	[1,450-1,750]	Triangle
FCHF	0.1	[0.02, 0.25]	log uniform
HTFB	300 W/m^2C	[100, 400]	Triangle
ACMPLB(1)	62.54 m^2	[43.78-62.54]	uniform
FKUTA	2.46	[2.46-3.7]	uniform
FWEBER	10.0	[1.0-100]	log uniform
ENT0C	0.045	[0.025-0.06]	uniform
FAOX	1.0	[1.0-2.0]	uniform
TCLMAX	2500 K	[2000-3000]	uniform
LMCOL0	50.0	[48-54]	uniform
LMCOL1	50.0	[48-54]	uniform
LMCOL2	50.0	[48-54]	uniform
LMCOL3	50.0	[48-54]	uniform
EPSCUT	0.1	[0-0.25]	Triangle
EPSCU2	0.2	[0-0.35]	Triangle
TAUTO	983 K	[750-1200]	Triangle
XSTIA	0.75	[0.55-0.75]	Triangle
TJBRN	1060 K	[900-1900]	Triangle

3. ANALYSIS RESULTS

3.1. Accident Progression Analyses of Representative Sequences

In advance of uncertainty analyses, accident progression analyses have been performed for the representative DET scenarios. The selected accident sequences are large loss of coolant (LOCA), loss of offsite power (LOOP) with auxiliary feedwater system (AFWS), and LOOP without AFWS sequences which are shown in Table 1. Each sequence has three cavity conditions: 'flood'. 'wet', and 'dry' cases. For the LLFLD scenario, none of engineered safety features (ESF) such as high pressure safety injection system (HPSIS), containment spray, or reactor containment fan cooler is available. The only available system is a low pressure safety injection system (LPSIS). For the LPFLD scenario, the only water available to cool the core is from the HPSIS. For the LOOP sequences, the injected water from four safety injection tanks (SITs) or HPSIS is available only after the system pressure decreases after the reactor vessel failure.

Complete coverage of corium behavior both in-vessel and ex-vessel, and the corresponding containment responses, can be predicted in the MAAP code analyses. The in-vessel progressions include the thermal hydraulics in the primary system, core heat up, hydrogen generation, and melt progression up to the reactor vessel breach. The ex-vessel progressions include high pressure melt ejection, direct containment heating, gas combustion phenomena, molten-corium concrete interaction and the pressure behavior in the containment atmosphere. The values of the MAAP uncertain input parameters for these scenarios are taken from the default values in Table 3. The calculation results for the timing of key events and the pressure loads in the containment are summarized in Table 4.

Table 4: Timing of key events occurrence and containment pressure load for the representive DET scenarios in OPR-1000

Sequence ID	Simulated Scenario		Timing of Key Event Occurrence (seconds)			Containment Pressure (MPa)
	Initiating Event	Safety System Availability	Steam Generator Dryout	Core Uncovery	Reactor Vessel Failure	Peak Pressure at 72 hours
LLFLD	Large Loss of Coolant Accident	LPSIS	No Dryout	< 10.0	21,170	1.384
LLWET		SIT	No Dryout	< 10.0	9,796	1.055
LLDRY		N/A	No Dryout	< 10.0	5,043	0.774
LPFLD	Loss of Offsite Power Accident	HPSIS	5,340	6,994	14,914	1.296
LPWET		SIT	5,340	6,994	14,914	1.244
LPDRY		N/A	5,340	6,994	14,914	0.812
LPFLDSG		AFWS, HPSIS	55,869	60779	75,089	0.734
LPWETSG		AFWS, SIT	54,516	59,392	73,360	1.031
LPDRYSG		AFWS	54,516	59,392	73,360	0.649

3.2. Uncertainty Analysis

The results of the 200 MAAP analyses constitute samples of the distribution of the containment pressure load related variables given the uncertainties expressed in Table 3. In this study, any dependency between parameters was not considered in the sampling process, and thus all parameters were treated as independent. The results of all 200 MAAP analyses of the uncertain code parameters for the 9 scenarios are shown in Table 5. Since this application was focused on determining an estimate of the pressure load in the containment building, the calculation results of the relevant variables are shown from Fig. 2 through Fig. 7 for the case of the LPWER scenario as examples. The figures show the calculation results of the pressure behavior in the containment building, axial concrete erosion depth in the cavity, hydrogen combustion mass in the containment, and their distributions.

Table 5: Calculation results for the pressure load related variables of the late containment failure in the uncertainty analysis

Calculation Scenario ID	Peak Pressure (MPa)		Axial Concrete Erosion (m)		H_2 Burn Mass (kg)	
	Mean	Deviation	Mean	Deviation	Mean	Deviation
LLFLD	1.428	0.037	0.463	0.396	11.2	77.0
LLWET	1.044	0.023	3.10	0.84	6.56	0.78
LLDRY	0.776	0.053	3.74	1.19	1695	183
LPFLD	1.292	0.041	0.065	0.157	108.2	129.9
LPWET	1.228	0.050	2.21	0.67	127.2	295.5
LPDRY	0.767	0.026	3.02	0.94	1500	445
LPFLDSG	0.753	0.023	0.005	0.007	86.8	115.1
LPWETSG	0.990	0.069	0.99	0.56	75.6	112.9
LPDRYSG	0.592	0.056	2.13	0.85	127.2	295.5

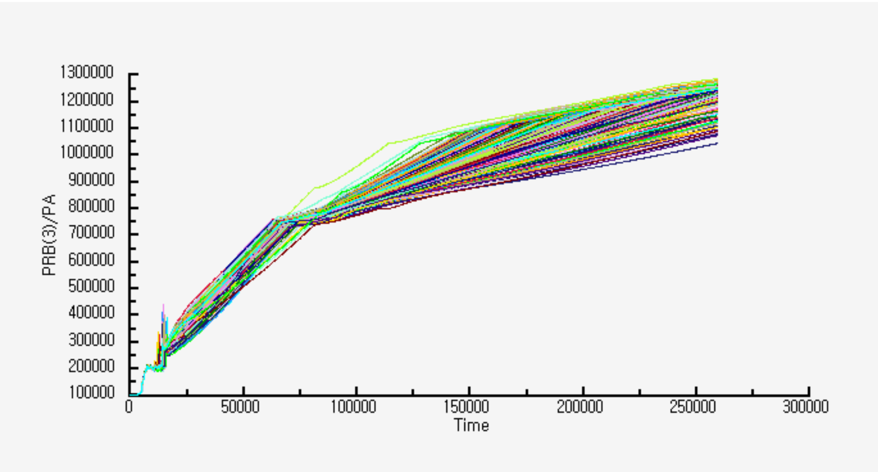

Fig. 2: Pressure behavior in the containment building (LPWET case) (time: seconds)

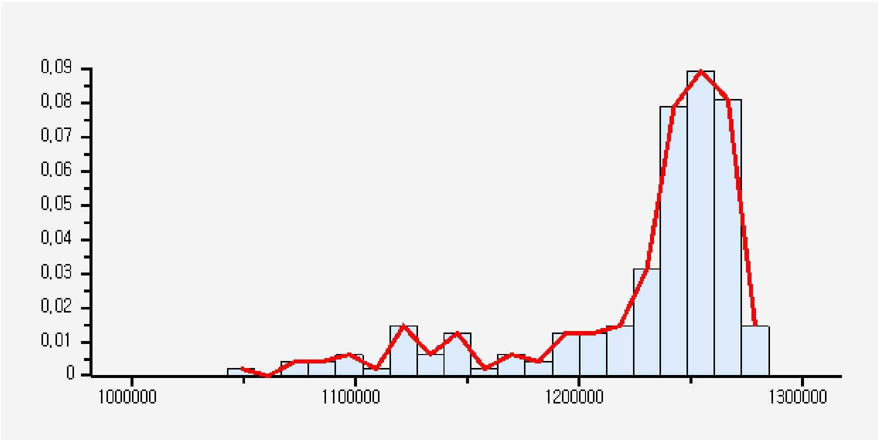

Fig. 3: Distribution of peak pressure in the containment building (LPWET case) (Mean: 1.228 MPa, Deviation: 0.050 MPa)

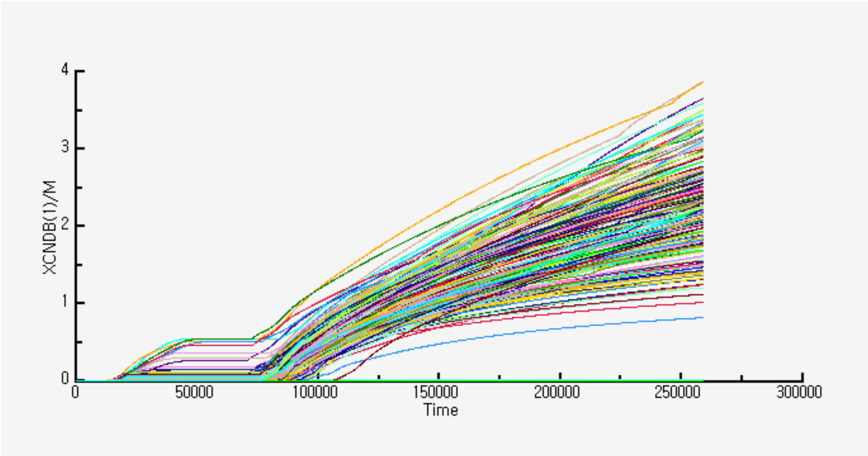

Fig. 4: Axial concrete erosion behavior in the cavity (LPWET case) (time: seconds)

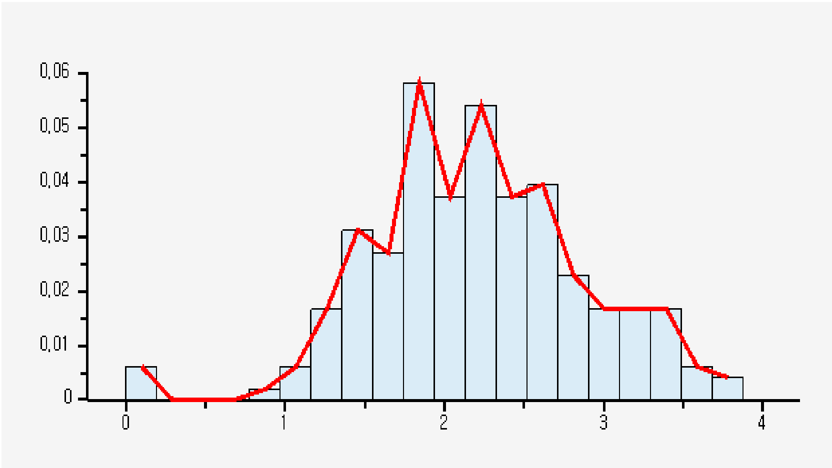

Fig. 5: Distribution of the axial concrete erosion in the cavity (LPWET case) (Mean: 2.21 m, Deviation: 0.67 m)

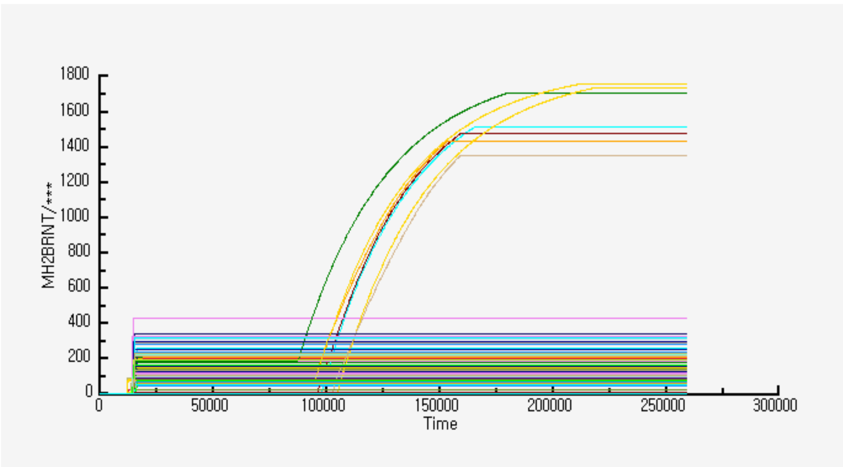

Fig. 6: Hydrogen combustion mass behavior in the containment (LPWET case) (time: seconds)

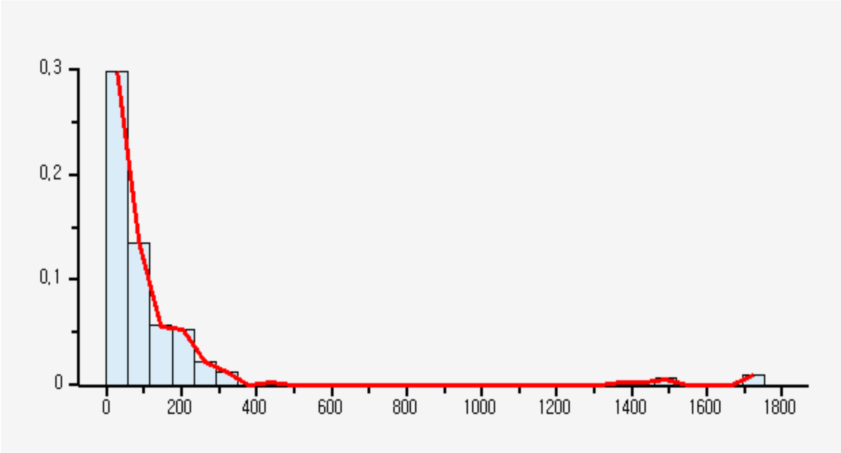

Fig. 7: Distribution of the hydrogen combustion mass in the containment (LPWET case) (Mean: 127.2 kg, Deviation: 295.5 kg)

4. SUMMARY AND CONCLUSION

The phenomenology of severe accidents is extremely complex. Severe accident evaluation methodologies are associated with large uncertainties. Thus, a quantitative evaluation of the uncertainties associated with the results of a level 2 PSA requires knowledge of the uncertainties in the severe accident phenomenology. Such epistemic uncertainties are the major source of uncertainty in the results of a level 2 PSA [6].

In this paper, a sampling-based phenomenological uncertainty analysis was performed to statistically quantify uncertainties associated with the pressure load of a containment building for a late containment failure evaluation, based on the key modeling parameters employed in the MAAP code and random samples for those parameters. Phenomenological issues surrounding the late containment failure mode are highly complex. Included are the pressurization owing to steam generation in the cavity, molten corium-concrete interaction, late hydrogen burn in the containment, and the secondary heat removal availability. The methodology and calculation results can be applied for the optimum assessment of a late containment failure model. The accident sequences considered were a loss of coolant accidents and loss of offsite accidents expected in the OPR-1000 plant. As a result, uncertainties addressed in the pressure load of the containment building were quantified as a function of time.

A realistic evaluation of the mean and variance estimates provides a more complete characterization of the risks than conservative point value estimates. Therefore, the analysis methodologies demonstrated by these phenomenological uncertainty studies can be much preferable over deterministic methods employing a conservative selection of the code parameters. This methodology provides an alternative to simple deterministic analyses and sensitivity studies for use in the containment performance analysis of a level 2 PSA, and provides insight into identify the additional research area to reduce the uncertainties associated with severe accident phenomena by an investigation of the responsible uncertain parameters, and provides a useful tool in establishing risk-informed or severe accident related regulation to the nuclear industry.

Acknowledgements

This work was supported by Nuclear Research & Development Program of the National Research Foundation of Korea (NRF) grant, funded by the Korean government, Ministry of Science, Ict & future Planning (MSIP).

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