

AN APPROACH TO EVALUATION OF UNCERTAINTIES IN LEVEL 2 PSAs*

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Abstract: This paper describes an approach to evaluation of uncertainties in Level 2 PSAs being developed at the Japan Atomic Energy Research Institute (JAERI). Uncertainty analysis would provide information on uncertainty ranges of containment function failure frequency and source terms as well as their contributors. The information on uncertainty is quite important when PSA results are used for decision making in various areas including risk informed regulation, accident management, off-site emergency planning, and the legal system for compensation of nuclear damage. Studies on uncertainty evaluation at JAERI include (1) development and improvement of computer codes, (2) assessment of uncertain parameters, and (3) development of an uncertainty analysis method for source terms. To quantify uncertainties in parameters, recent experimental and analytical research results are surveyed. Also an analytical study is being performed for evaluating uncertainties in containment failure probabilities due to energetic events such as steam explosion and direct containment heating by using a decomposition event tree method, where aleatory and epistemic uncertainties are addressed. To evaluate uncertainties in source terms, Monte Carlo simulation method is used with an integrated severe accident analysis code THALES2 developed at JAERI, where parameter values in the code are assigned by user-specified distributions

Keywords: Uncertainty analysis, PSA, Containment event tree, Source terms, Energetic event, THALES2

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1. INTRODUCTION

Probabilistic Safety Assessment (PSA) has been recognized to be one of the most promising methods that evaluate reactor safety comprehensively and systematically. As the quality of PSA increases, the use of PSA has been steadily expanding in the area of risk informed regulation, accident management (AM) measures, emergency planning and so on. Recently Japanese utilities established severe AM measures, where PSA was used to select AM measures and to evaluate their effectiveness. The Nuclear Safety Commission has started discussion on the establishment of safety goals, where PSA results are useful in providing technical basis on quantitative risks to the public.

The Japan Atomic Energy Research Institute (JAERI) has been studying PSA of Light Water Reactors (LWRs) for internal and seismic events. JAERI made the first evaluation of Level 1 and 2 PSA of a Japanese reference BWR with Mark-II type containment for internal events in 1990 (This is called the first phase PSA hereafter). The first phase PSA did not include evaluation of 1) effectiveness of AM measures, 2) source terms for potential energetic events (steam explosion, direct containment heating and so on), and 3) uncertainties in the results of the PSA. JAERI has started a program of PSA that takes accounts of the three items 1) - 3) above for a 1,100 MWe BWR with Mark-II type containment. Also a limited scope of PSA has been started for a 1,100 MWe 4-loop PWR with prestressed concrete containment vessel (PCCV), where significant accident scenarios identified so far are focused.

Since a PSA investigate a number of possible accident sequences, and they are not fully understood, there exist uncertainties in PSA results. The information on uncertainty is quite important when PSA results are used in several areas mentioned above, thus uncertainty evaluation is one of the most important issues in PSA. Three types of uncertainties are recognized [1]; they are parameter (data) uncertainty, model uncertainty, and completeness uncertainty. Parameter uncertainties are associated with the values of basic parameters such as component failure rate, containment failure pressure, and so on. The uncertainties are propagated through a model to generate the uncertainty in an end result. The end result uncertainty is usually evaluated by a Monte Carlo simulation method. Model uncertainties are associated with phenomenological models for the physical-chemical processes during severe accidents. The uncertainties are studied by a comparison of the model results with experimental data or different model results. Completeness uncertainties are associated with the scope of PSA or truncation effects. This type of uncertainty is not evaluated in the PSA scope. An additional analysis beyond the scope is necessary to evaluate the completeness uncertainties. This paper addresses parameter and model uncertainties and describes an approach to evaluation of uncertainties in Level 2 PSAs being developed at JAERI.

Studies on uncertainty evaluation at JAERI include (1) development and improvement of computer codes, (2) assessment of uncertain parameters, and (3) development of an uncertainty analysis method for source terms.

In Section 2, a framework of uncertainty analysis method in Level 2 PSA is described. In Section 3, studies on uncertainty evaluation, items (1) - (3) above, are described. Concluding remarks are given in Section 4.

2. FRAMEWORK OF UNCERTAINTY EVALUATION IN LEVEL 2 PSA

Level 2 PSA consists of the three steps of analyses, core damage frequency analysis (Level 1 PSA), accident progression analysis, and source term analysis. Figure 1 shows a framework of uncertainty analysis for Level 2 PSAs at JAERI. In core damage frequency analysis, initiating events are identified followed by a system event tree (ET) analysis and fault tree (FT) analysis, core damage sequences and plant damage states (PDS) are identified, and the core damage frequency (CDF) is estimated. Here uncertainties in component failure probabilities are addressed to evaluate the uncertainty in CDF. The identified core damage accident sequences are used as input for the following accident progression analysis.

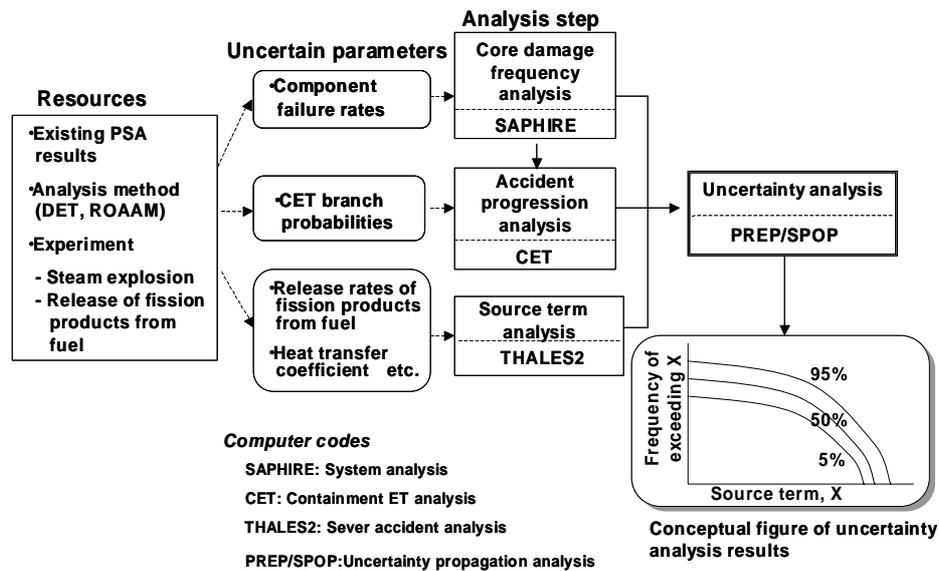


Fig. 1 Framework of uncertainty analysis for level 2 PSA

In accident progression analysis, accident scenarios resulting in potential containment function failure are identified and their occurrence probabilities are estimated. The analysis is performed with a containment event tree (CET) method for each core damage sequence or PDS. JAERI developed the accident progression stage event tree (APSET) method [2] where the accident progression are divided into several stages that are the pre-core melt stage, the core melt stage, the stage just after reactor vessel failure, and the long term progression stage. Uncertainties in CET branch probabilities are addressed to evaluate the uncertainty in containment function failure probability.

In source term analysis, thermal hydraulic and fission product (FP) transport behavior during severe accidents are analyzed for each accident scenario identified above. Uncertainties in parameters such as FP release rates from fuel and a heat transfer coefficient between molten core and water are addressed to evaluate the uncertainty in source terms.

These results including uncertainties are collected, and analyzed in a statistical method to evaluate the uncertainty in the endpoint of Level 2 PSA. For example, the relationship between an occurrence frequency of FP release to the environment and the associated source term is

illustrated in the form of a complementary cumulative distribution function (CCDF) for source term.

3. APPROACH TO UNCERTAINTY EVALUATION

3.1 Computer codes

To evaluate uncertainties in CDF, we introduced the SAPHIRE code [3] developed at U.S. Nuclear Regulatory Commission (USNRC). SAPHIRE has capability of analyzing uncertainties in CDF by Monte Carlo simulation with the basic event probabilities drawn from user-specified distributions.

To evaluate probabilities of containment function failure identified by the CET method, we developed a CET analysis code. The CET analysis code was constructed based on an object-oriented concept to flexibly cope with change of logic or data in the CET. In the CET analysis code, each object for a heading has rules and branch probabilities. Rules determine the branch probabilities depending on the status of preceding headings.

To evaluate source terms, an integrated severe accident analysis code THALES2 [4] developed at JAERI is used. THALES2 simulates the thermal-hydraulic transient in the reactor cooling system of a BWR or PWR, containment and reactor buildings with consideration of the operation of emergency core cooling system (ECCS) and other engineered safety systems together with their control logics.

The thermal hydraulics model of the code is based on a volume and junction concept. Parameters such as mass of fluid, pressure, mixture level, void fraction, and temperatures of fluid (gas and liquid) and wall are calculated in each volume. Severe accident phenomena considered in the code include the hydrogen generation by the metal-water reaction, reactor vessel lower head failure, hydrogen burning in the containment and generation of non-condensable gases by the interaction of molten core and concrete.

The models for the FP release and transport are based on those of the ART code [5] and they cover the release of FPs from fuel and the transport of FPs in the form of gas, aerosol, deposit on structure walls and floors, and solution in water. The aerosol model considers the size growth by agglomeration and condensation/evaporation, chemical adsorption of the gas species at structure surfaces, deposition of aerosol to walls and floors, removal by sprays and filters, scrubbing by water pools, and convection by liquid as well as gas flow.

To evaluate uncertainties in the endpoints of concern, the PREP/SPOP code [6,7] developed at the Joint Research Center (JRC) of Ispra is used. The PREP code is for preparation of an input sample for Monte Carlo simulations, and the SPOP code is for performing uncertainty and sensitivity analyses on the output of a user implemented model.

3.2 Assessment of uncertain parameters

This section describes assessment of uncertain parameters in the steps of core damage frequency analysis and accident progression analysis. Assessment of uncertain parameters in the step of source term analysis is described in the next section.

3.2.1 Core damage frequency analysis

Uncertain parameters in core damage frequency analysis are basic event probabilities

including component failure rates, human error rates, and common cause failure. In the first phase Level 1 PSA, we mainly utilized component failure rate data from U.S. database, because available domestic data were, then, rare. Recently component failure rates were evaluated by collecting and analyzing operational data of 49 domestic Nuclear Power Plants (NPPs) from 1982 to 1997 at Central Research Institute of Electric Power Industry (CRIEPI) [8]. The evaluated results show that component failure rates for Japanese NPPs have a tendency to be lower than those for U.S. NPPs. To reevaluate CDF and perform an uncertainty analysis for CDF, input data to SAPHIRE are made out of the component failure rates derived for Japanese NPPs.

3.2.2 Accident progression analysis

In accident progression analysis with the CET method, branch probabilities and some physical quantities are addressed. Branch probabilities include probabilities of containment function failure modes such as containment overpressure, containment overtemperature, in- or ex- vessel steam explosion, direct containment heating, hydrogen burning and containment venting. Physical quantities includes such as the flow rate of molten core for the large scale failure mode of the reactor vessel to evaluate energy generated by steam explosion.

To assign uncertainties for these parameters, 1) recent experimental and analytical studies are surveyed and reviewed, 2) THALES2 results of thermal hydraulic behavior during severe accidents are utilized, and 3) analytical methods such as a decomposition event tree (DET) method are used for evaluating containment failure probability due to energetic events including steam explosion and direct containment heating. As an examples of the assessment regarding items 3), described is containment failure probability due to ex-vessel steam explosion at a PWR based on a DET method.

(1) Containment failure probability due to ex-vessel steam explosion at a PWR based on a DET method

When molten debris is ejected from reactor vessel, there is a possibility that a steam explosion would take place if a reactor cavity is filled with water. The steam explosion would generate dynamic load into the containment, cause cavity wall failure, move the reactor vessel, cause piping tenseness, and hence cause loss of containment integrity at a penetration. The Nuclear Safety Research Association (NSRA) constructed a decomposition event tree (DET) to evaluate containment failure probability caused by ex-vessel steam explosion at an advanced PWR (APWR) plant [9]. Since physical processes resulting in containment failure due to steam explosion at a 4-loop PWR plant of concern are similar to those at the APWR, and the type of containment of the both plants is PCCV, the DET for the APWR was applied to analyze containment failure probability at the 4-loop PWR. Figure 2 shows the DET used in this analysis, and Table 1 shows headings and branch probabilities used. The DET identifies possible scenarios (paths) that result in loss of containment integrity. For each scenario, a load energy brought into containment and its occurrence probability are calculated. The load energy is compared with a critical load energy over which the containment integrity is lost to identify scenarios resulting in loss of containment integrity. Then the containment failure probability is calculated by summing up probabilities of these scenarios identified.

RV failure	RV failure mode(-)	Flow rate of melt (kg/s)	Internal energy of melt (MJ/kg)	Occurrence of triggering(-)	Melt mass in premixture (kg)	Energy conversion rate (-)	Energy generated (MJ)	
	Small scale	Small	Low	No	Small	Low		
				Yes		Medium		
						High		
						Medium	Low	
						Medium	Medium	
						High		
			Large	Low				
				Medium				
				High				
				Low				
				Medium				
				High				
		Large	High	No	Small	Low		
	Yes			Medium				
				High				
				Medium		Low		
				Medium		Medium		
				High				
		Large	High	Yes	Large	Low		
						Medium		
						High		
						Low		
						Medium		
						High		
	Large scale	Large	Low	No	Small	Low		
				Yes		Medium		
						High		
						Medium	Low	
						Medium	Medium	
						High		
		Large	High	High	Yes	Large	Low	
							Medium	
							High	
							Low	
							Medium	
							High	
	Large scale	Large	High	No	Large	Low		
				Yes		Medium		
						High		
						Low		
			High					
			High	No	Large	Low		
			Yes	Medium				
				High				

Fig. 2 Decomposition event tree for stem explosion induced containment failure (from Ref. 9)

Table 1 Heading and branch probability

Heading	Branch probability	
	Point estimate	Uncertainty
Failure mode of reactor vessel		
Small scale	0.99	1- P(Large scale) ^(a)
Large scale	0.01	LM:-4.61, LSD:1.54, LN
Flow rate of molten core		
For small scale failure		
Low (420 kg/s)	0.9	
High (650 kg/s)	0.1	
For large scale failure		
High(45980 kg/s)	1	
Internal energy of molten core		
Low (1.2 MJ/kg)	0.9	
High (1.5 MJ/kg)	0.1	
Occurrence of triggering		
No	0.9	1- P(Yes) ^(b)
Yes	0.1	LM:-2.3, LSD:0.77, LN
Melt mass in premixture		
Small	0.6	
Medium	0.39	
Large	0.01	
Energy conversion rate		
For small scale failure		
Low (0.01)	0.4	
Medium (0.02)	0.4	
High (0.025)	0.2	
For large scale failure		
Low (0.008)	0.5	
Medium (0.01)	0.4	
High (0.011)	0.1	

LM : Logarithmic mean, LSD : Logarithmic standard deviation

LN: Lognormal distribution

(a): P(Large scale) denotes occurrence probability of large scale failure mode.

(b): P(Yes) denotes occurrence probability of triggering.

The containment failure probability depends on branch probabilities, the load energy, and the critical load energy. In the NSRA study for the APWR, much effort was made to determine these parameter values by surveying research results and/or by analyses. Since some parameters were expected to depend on plants, differences in such parameters between the two plants were taken into accounts in this analysis.

Two physical quantities regarding the phenomena of steam explosion induced containment failure were reevaluated. One is the flow rate of molten core for the large scale failure mode of the reactor vessel. It is expected that whole the molten core in the lower plenum would fall down into the pedestal, and the mass of molten core is proportional to a reactor power. Then the flow rate at the 4-loop PWR was set to be smaller than that at the APWR in consideration of their power (4-loop PWR: 3,411MWt, APWR: 4,451 MWt). On the other hand, the flow rate of molten core for the small scale failure was set to be the same as that at APWR, because the feature of the failure between the two plants is similar. The other is the critical load energy

brought into the containment causing loss of containment integrity. It is supposed that the critical load energy at the 4-loop PWR is lower than that at APWR according to their powers. Also uncertainties in heading branch probabilities and in the critical load energy were addressed in this study to evaluate uncertainty in containment failure probability. Headings considered for uncertainty evaluation are “reactor vessel failure mode” and “occurrence of triggering”. In the uncertainty analysis, two types of uncertainties, aleatory and epistemic uncertainties, were addressed. Aleatory uncertainty is caused by randomness or stochastic properties accompanied with phenomena or data of concern, and epistemic uncertainty is caused by lack of our knowledge. Since the uncertainties are difficult to assess and differ from expert to expert, they are often determined by subjective judgment (analysts’ degree of belief). For the branch probabilities for reactor vessel failure mode and occurrence of triggering, epistemic uncertainties were addressed. Due to lack of our knowledge, the uncertain ranges of the branch probabilities for these headings were maximized, that is, the upper limit probability was set to be 1. In the NSRA study, the critical load energy causing containment failure was expressed by a normal distribution with a mean of 523 MJ and a standard deviation of 15 MJ for APWR based on an analysis of structural deformation of reinforced concrete. In the present study, the critical load energy was applied by taking account of the difference of reactor powers between the APWR and the 4-loop PWR. It is assumed that the critical load energy is expressed by a normal distribution with a mean of 400 MJ and a standard deviation of 15 MJ. Here this type of uncertainty is aleatory uncertainty caused by uncertainty in material strength. To assess epistemic uncertainty, the dynamic load factor that is the ratio of dynamic load to static one was investigated. As the maximum dynamic load is 1.5, it is assumed that epistemic uncertainty is expressed by a uniform distribution function for the mean of the critical load energy with a range of 200 - 600 MJ. Figure 3 shows cumulative containment failure probability as a function of the load energy brought by steam explosion with 5th, 50th and 95th percentiles. It is noted that epistemic uncertainty is larger than aleatory uncertainty.

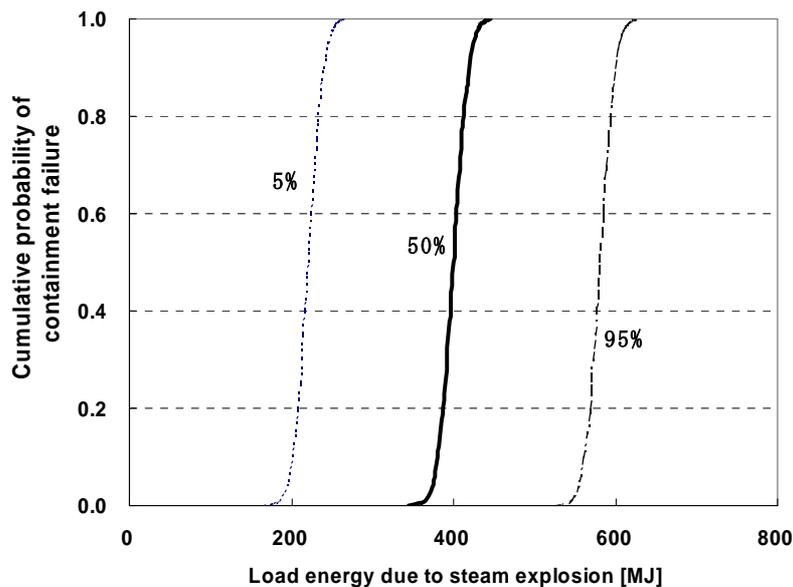


Fig. 3 Cumulative probability of containment failure as a function of the load energy

Figure 4 shows probability distribution of the load energy generated by steam explosion for various scenarios when point estimate values of branch probabilities are used. It is found that there are two groups of energy. One is in the range of less than 100 MJ, and the other is in the range of 300 – 500 MJ. The lower energy group corresponds to the scenarios lead from small scale failures of the reactor vessel, and the higher one from large scale failures of the reactor vessel. This is because as the scale of the failure is larger, larger amount of molten core is ejected from the reactor vessel and more strongly contributes to the steam explosion, and vice versa. Comparing the distribution of the load energy in Figure 4 with the containment failure probability as a function of the load energy in Figure 3, it is noticed that scenarios contributing containment failure is only the higher energy group even when the lowest critical load energy is assigned.

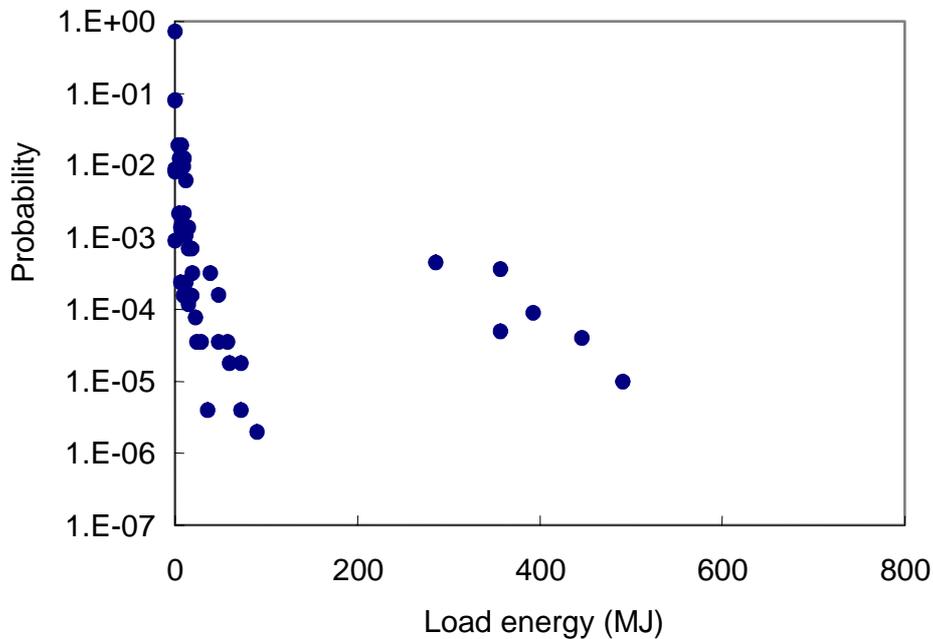


Fig. 4 Distribution of the load energy

A cumulative distribution function (CDF) of the containment failure probability is shown in Figure 5, where contribution from aleatory and epistemic uncertainties in the critical load energy is illustrated separately. The uncertainty of containment failure probability is in the range of 0 - 0.006 and a mean of 0.0015. It is found that epistemic uncertainty dominantly contributes to the total uncertainty.

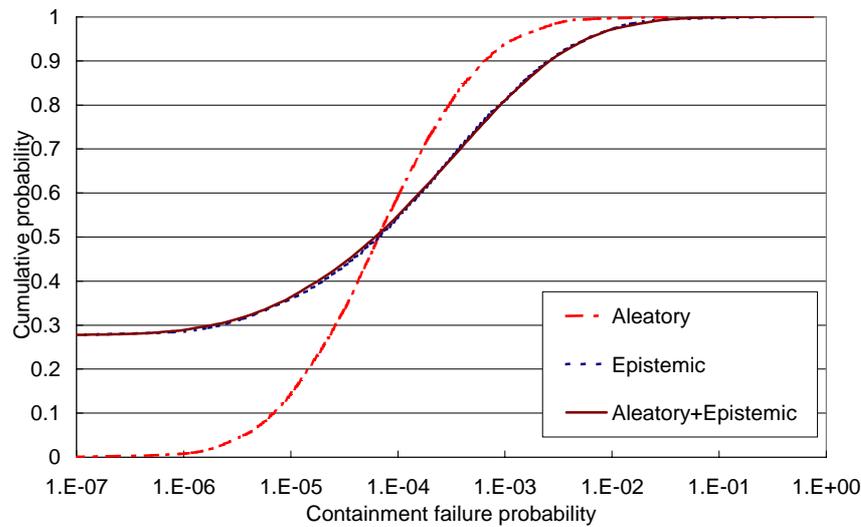


Fig. 5 Uncertainty analysis results for containment failure probability due to ex-vessel steam explosion.

3.3 Uncertainty analysis method for source terms

3.3.1 Uncertainty analysis method

Uncertainty evaluation of source terms for a severe accident sequence caused by uncertain parameters through a model is estimated using a Monte Carlo method. The method requires a large number of calculations by changing values of uncertain parameters. Since it often takes a long computation time for so many calculations with a large severe accident analysis code, USNRC developed a parametric simulation code XSOR in the NUREG-1150 study [10]. The parametric code includes several parameters such as fraction of initial inventory of species i released from the fuel prior to vessel failure, fraction of material released from the fuel that is released from the vessel, and so on. Since these parameters depend on accident sequences and times, it is difficult to assign reasonable and consistent values to them in consideration of uncertainties. In the NUREG-1150 study, experts' opinions (degree of belief) were elicited and their results were integrated to give probability density functions for the parameters. Then the method used by USNRC requires much effort to assign values to the uncertain parameters by eliciting experts' opinions. Also it is not easy to verify the parametric code, because XSOR analyzes a set of sequences characterized by some accident conditions, and hence the XSOR results may not be directly compared with those by another severe accident analysis code that is designed to analyze one definite sequence.

As computer technology has been developed, it would be possible to use a severe accident analysis code for a large number of calculations. Thus, an approach presented in this paper is utilization of the severe accident analysis code THALES2 with Monte Carlo simulation, where parameter values in the code are assigned by user-specified distributions. Nevertheless it is not practical to analyze the all core damage accident sequences identified in Level 1 PSA. Hence, many accident sequences identified in Level 1 PSA are classified into several groups of accident sequences by taking account of similarity of accident progression and source terms. Table 2 shows an example of the groups of accident sequences classified for the BWR plant with

Mark-II type containment. In practical calculations, the groups including risk dominant accident sequence of which occurrence frequencies and source terms are relatively higher and larger, respectively, will be analyzed preferentially.

Table 2 Classification of sequences for uncertainty analysis (BWR Mark-II)

Sequence Group	RPV Pressure	Sequences subgroup	Representative sequences
Loss of containment heat removal function with high pressure core injection available	High ~ Medium	Transient or SB-LOCA	TW
	Medium ~ Low	MB- ~ LB-LOCA (Reactor is depressurized by LOCA)	S1W
	Low	Transient with reactor depressurized by failure of SRV reclose	TPQU1W
Loss of containment heat removal function with low pressure core injection available	Low	Transient or SB-~LB-LOCA (Reactor is depressurized by ADS or failure of SRV reclose or LOCA)	S1UV1W
Loss of coolant injection	High ~ Medium	Transient or SB-LOCA	TQUX
	Medium ~ Low	MB- ~ LB-LOCA (Reactor is depressurized by LOCA)	S1UX
	Low	Transient or SB-LOCA (Reactor is depressurized by ADS or failure of SRV reclose or LOCA)	TQUV
	Low	MB-~LB-LOCA (Reactor is depressurized by ADS or LOCA)	S1UV
Station blackout	High ~ Medium	Transient or SB-LOCA	TB
	Low	Transient with reactor is depressurized by failure of SRV reclose	TBP
Anticipated transient without scram (ATWS)	High ~ Medium	Transient or SB-LOCA	TC
	Low	MB-~LB-LOCA (Reactor is depressurized by LOCA)	S1C

RPV: Reactor Pressure Vessel, SRV: Safety Relief Valve, ADS: Automatic Depressurization System, SB-LOCA: Small Break Loss Of Coolant Accident(LOCA), MB-LOCA: Medium Break LOCA, LB-LOCA: Large Break LOCA

Sequence Identifier: T(Transient), C(failure of scram), S1(Medium Break LOCA), P(Loss of SRV Reclose), Q(Loss of Feedwater), U(Loss of all High Pressure Core Injection), U1(Loss of Reactor Core Isolation Cooling System), X(Loss of Reactor Depressurization), V(Loss of all Low Pressure Core Injection), V1(Loss of Low Pressure Core Spray), W(Loss of Heat Removal Function)

This approach has the following features and advantage. Uncertain parameters considered in

THALES2 are rather fundamental, and less dependent on accident sequences and times. Then it would be easier to assess the uncertain parameters, because they are more closely related to data obtained from individual experimental and analytical studies.

3.3.2 Assessment of uncertain parameters

Table 3 shows an example of uncertain items and associated parameters in THALES2 for the BWR plant with Mark-II type containment. These uncertain parameters and their ranges are assessed by surveying recent experimental and analytical studies as well as by our experiences on analyzing severe accidents with THALES2.

Table 3 An example of uncertain items and associated parameters in THALES2 (BWR Mark-II)

Uncertain items	Associated parameters
Release of FPs from fuel (in-vessel)	- Release rate of FPs (for volatile FPs)
Coolability of molten core (in-vessel)	- Heat transfer coefficient between molten core and coolant - Fraction of fragmented molten core in FCI - Average particle size of molten core droplets in FCI - Heat transfer coefficient between molten core and lower head wall
Deposition of FPs (in-vessel)	- Deposition rate of FPs to the wall - Deposition rate of FPs to the floor
Deposition of FPs (ex-vessel)	- Deposition rate of FPs to the wall - Deposition rate of FPs to the floor
Release of FPs from molten core (ex-vessel)	- Release rate of FPs (for volatile FPs)
Coolability of molten core (ex-vessel)	- Heat transfer coefficient between molten core and coolant
Integrity of containment	- Failure pressure by overpressure - Size of rupture
Pool scrubbing	- Size of bubble - Rising velocity of bubble
LOCA	- Break size in LOCA

FP: Fission Product, FCI: Fuel Coolant Interaction

An example of the assessment regarding coolability of molten core in fuel coolant interaction (FCI) in reactor vessel is described in the following. In a core degradation phase, debris is gradually fallen down to lower part and accumulated on the core support plate. At the timing of core support plate failure debris is poured into the water in the lower plenum, fragmented and settled down to the bottom. The factors influencing the coolability of molten core are heat transfer coefficient and heat transfer surface area between molten core and water.

(1) Heat transfer coefficient between molten core and water

Powers et al. [11] estimated that the boil heat flux between molten core and water was in the range of 0.16 – 1.6 MW/m² with a lognormal distribution. Several tests [12-18] on coolability of molten core during the molten core concrete interaction (MCCI) showed that the heat flux between molten core and water was in the range of 0.2 – 0.8 MW/m². Considering the above results, we estimate that the heat flux is in the range of 0.16 – 1.6 MW/m². To obtain the

uncertainty in the heat transfer coefficient (HTC) that is a input parameter to THALES2, we performed a sensitivity analysis by changing the value of HTC. The analysis showed that the uncertainty in the heat flux is reproduced by using the value of HTC from 100 to 600 MW/m²K. Hence we estimate that the uncertainty in HTC is in the range of 100 – 600 MW/m²K.

(2) Heat transfer surface area between molten core and water

The heat transfer surface area is determined by two input parameters in THALES2. They are the fraction of fragmented mass of debris and the average size of fragmented particles at the timing of fuel coolant interaction (FCI). To determine the ranges of these uncertain parameters, FCI tests carried out so far are surveyed. Among the tests, the FARO test at JRC of Ispra was selected, where the molten core including UO₂, ZrO₂, and Zr was used. The results of the FARO test for the saturated water condition are shown in Table 4 [11].

Table 4 The FARO test results (From Ref. 19)

Test case	L-06	L-08	L-11	L-14	L-19	L-20
Composition of melt	A	A	B	A	A	A
Mass of melt [kg]	18	44	151	125	157	96
Temperature of melt [K]	2923	3023	2823	3123	3073	3173
Water subcooling [K]	0	12	2	0	1	0
Fragmentated rate [-]	0.67	0.68	1.0	0.84	0.49	0.78
Median particle size [mm]	4.5	3.8	3.5	4.8	3.7	4.4

Composition of melt (A:80wt%UO₂-20wt%ZrO₂, B:77wt%UO₂-19wt%ZrO₂-4wt%Zr)

The range of fragmented rate for molten core is from 0.49 to 1.0. In the FARO tests, the whole molten core is injected into the water, however, during severe accidents at light water reactors the whole part of core would not melt as was realized at the FARO test conditions. Then it is anticipated that the fraction of fragmented mass of core debris during severe accidents is lower than the FARO results. Assuming the range of the fraction of molten core to the whole is from 0.5 to 0.75, the range of fragmented rate of debris is set to from 0.25 to 0.75.

The range of average particle size is determined by the FARO test results and COTELS test result. The FARO test shows that the range is from 3.5 to 4.8 mm as is shown in Table 4 [19]. The COTELS results [20] showed the mass median particle size was in the range of 5 - 7 mm. Thus we estimate that the range of average particle size is from 3.5 to 7 mm.

4. CONCLUDING REMARKS

One of the most important issues in quantifying uncertainties in Level 2 PSA is to assess uncertain parameters. To quantify uncertainties in parameters, experimental and analytical research results are surveyed. Also an analytical method is used to evaluate uncertainties in containment failure probabilities due to energetic events such as steam explosion and direct containment heating, where aleatory and epistemic uncertainties are addressed. The uncertainty analysis method for source terms is direct use of the severe accident analysis code THALES2 with Monte Carlo simulation.

In the framework of the uncertainty analysis method in this paper, uncertainties in containment function failure probabilities and the associated source term will be estimated by

using the uncertain parameter values assessed and the analysis method presented. The uncertainty will be repeatedly evaluated and revised as perspective and our knowledge on uncertain parameters are increased. In 2005, uncertainties in Level 2 PSA results of the 1,100 MWe BWR with Mark-II type containment will be estimated. The output will include uncertainties in containment function failure probability and source terms, and contributors to the uncertainties.

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