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Use of MAAP in Generating Accident Source Term Parameters

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Abstract

The parametric model method determines the accident source term which is presented by a set of source term parameters. In this method, the cumulative distribution of each source term parameter should be derived for its uncertainty analysis. This paper introduces a method of generating the parameters in the form of cumulative distribution using MAAP version 4.0. In MAAP, there are model parameters which could incorporate uncertain physical and/or chemical phenomena. In general, the model parameters do not have a point value but a range. In this paper, considering that, the input values of model parameters influencing each parameter are sampled using LHS. Then, the computation results are shown in cumulative distribution form.

For a case study, the CDFs of FCOR and FVES of Kori Unit 1 are derived. The target scenarios for the computation are the ones whose initial events are large LOCA, small LOCA and transient, respectively. It is found that the computed CDF' s in this study are consistent to those of NUREG-1150 and the use of MAAP is proven to be adequate in assessing the parameters of the severe accident source term.

1. Introduction

The risk from operating a nuclear power plant is unique in a sense that it is caused by the influences of radioactive materials. To quantify the risk, the source term as well as occurrence probabilities of accident scenarios which would release radioactive materials should be identified. The source term is determined by the factors like inventory, energy, time and location of radioactive nuclide releases when the accident occurs. Based upon various previous studies, the severe accidents whose occurrence probabilities are very small but whose radiological consequences are very serious are known to have a decisive influence on the overall risk of the plant.

Although many studies on severe accidents have been proceeded and are under way, the exact physical phenomena of their processes are hard to define. Therefore, the uncertainties resulted from the lack of knowledge on the physical process and the incompleteness of modeling severe accidents should be somehow adequately reflected in the source term. The Parametric Model Method proposed is one of the methods that could be used to present such uncertainties. This method is considered very effective even though it requires a large computation time.

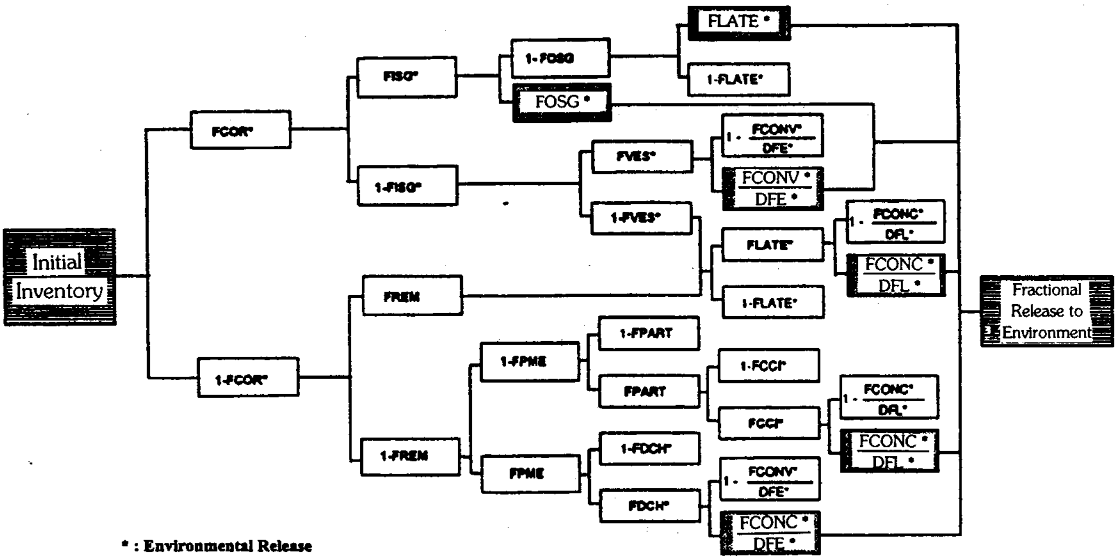

Fig. 1. Structure of Source Term Parameters.

To obtain the plant specific source term in the parametric model method, a database for the parameter should be established beforehand, which is actually the release fraction of nuclides in accordance with the corresponding accident phenomena and processes. In this paper, MAAP 4.0 has been used to generate the database for the parameter. As basic parameters, two important parameters are chosen; FCOR and FVES, which are the fraction of nuclide in the initial core inventory that is released from the fuel to the vessel before vessel failure, and the fraction of nuclide released from the fuel in the vessel that is released from the vessel at or before vessel failure, respectively. First, through an extensive literature search, the factors affecting the parameters are identified, and the model parameters which model such factors in MAAP code are selected. By changing the values for the model parameters, the two parameters are computed for various cases of accident scenario and two resultant sets of parameters are created which subsequently form two cumulative distribution functions(CDF' s).

For a case study, CDF' s of FCOR and FVES of Kori Unit 1 are derived. The input file of MAAP code requires the plant characteristics of Kori Unit 1 in details, and the target accident scenarios chosen for the computation are AHF, S2HF and TMLB′ whose initiating events are large LOCA, small LOCA and transient, respectively.

2. Parametric Model Method

2.1. Introduction

The parametric model method is a method which classifies the transport of fission products into several principal steps as the accident processes and calculates the source term for each step by introducing the parameter, which is defined as the release fraction of each step.[1] Figure 1 shows the structure of parameters, in which the shadowed parameters represent the source term that finally leak to the environment. Each parameter is described in Table 1.

Table 1. PWR-Related Parameters.

PARAMETER	MEANING
FCOR(i) *	fraction of i in the initial core inventory that is released from the fuel to the vessel before the vessel fails
FISG(i)	fraction of i of FCOR(i) that is transported to S/G
FOSG(i)	fraction of i of FISG(i) that is released to environments
FVES(i)	fraction of i released from the fuel in the vessel that is released from the vessel at, or before, vessel failure
FCONV	fraction of a radionuclide released from the containment to environments without decontamination at or before vessel breach
DFE	early decontamination factor
FREM	fraction of core that remains in vessel after vessel fails
FPART	fraction of core that takes part in CCI
FPME	fraction of core that is spouted by high-pressure melt
FDCH(i)	fraction of the inventory of i of FPME that is released to containment as a result of pressure-driven melt expulsion
FCCI(i)	fraction of i of FPART that is released from CCI
FCONC	fraction of a radionuclide released from the containment to environments without decontamination after vessel breach
DFL	late decontamination factor
FLATE(i)	fraction of i remaining in the RCS (or SG) that is revaporized and released from the RCS (or SG) during and after vessel breach

* i represents radionuclide group i

The accident source term is given by

$$ST(i) = ST_i(i) + ST_h(i) + ST_c(i) + ST_r(i) + ST_s(i) \quad (1)$$

where $ST_i(i)$ is the amount of radionuclide i that is released to the environment through the boundary of RCS without deposition after it is released from the core to the vessel before the vessel fails,

$ST_h(i)$ is the amount of radionuclide i that is released to the environments from the core spouting with high pressure,

$ST_c(i)$ is the amount of radionuclide i that is released to the environments by core/concrete interaction,

$ST_r(i)$ is the amount of radionuclide i that is released to the environments by revaporization from the deposition to RCS or remaining in the vessel after vessel breach, and $ST_s(i)$ is the amount of radionuclide i that is released to the environments by revaporization from the deposition to secondary S/G during S/G tube rupture.[2][3]

Each term of Eq. (1) can be represented by a combination of parameters as follows :

$$ST_i(i) = FCOR(i) * [FISG(i) * FOSG(i) + (1-FISG(i)) * FVES(i) * FCONV/DFE] \quad (2)$$

Table 2. Radionuclide Groups.

GROUP	Radionuclides	Principal nuclide
Noble Gases	Xenon(Xe), Kripton(Kr)	Noble Gas
Halogens	Iodine(I), Bronium(Br)	Iodine
Alkali Metals	Cesium(Cs), Rubidium(Rb)	Cesium
Tellurium Group	Tellurium(Te), Selenium(Se), Antimony(Sb)	Tellurium
Barium	Barium(Ba)	Barium
Strontium	Strontium(Sr)	Strontium
Noble Metals	Ruthenium(Ru), Molybdenium(Mo), Palladium(Pd),Rhodium(Rh), Technetium(Tc)	Ruthenium
Lanthanides	Lanthanum(La),Neodymium(Nb), Niobium(Nb), Europium(Eu),Yttrium(Y), Praseodymium(Pm), Samarium(Sm), Zirconium(Zr)	Lanthanum
Cerium Group	Cerium(Ce), Neptunium(Np), Plutonium(Pu)	Cerium

$$ST_h(i) = (1-FCOR(i)) * (1-FREM) * FPME * FDCH(i) * RM\&FCONV/DFE \quad (3)$$

$$ST_c(i) = (1-FCOR(i)) * (1\text{-}FREM) * (1\text{-}FPME) * FPART * FCCI(i) * FCONC(i)/DFL \quad (4)$$

$$ST_r(i) = [FCOR(i) * (1\text{-}FISG(i)) * (1-FVES(i)) + (1\text{-}FCOR(i)) * FREM] * FLATE(i) * FCONC(i)/DFL \quad (5)$$

$$ST_s(i) = FCOR(i) * FISG(i) * (1\text{-}FOSG(i)) * FLATE(i) \quad (6)$$

It can be seen that this method suggests a very simple way of computing the accident source term including its related uncertainties. In particular, the uncertainties can be very easily analyzed once the nature of each parameter is known with its implicated uncertainties. In this paper, the parameter has been represented as a distribution function rather than a unique numerical value. The demerit of the method is that it requires a huge amount of established data including plant specific characteristics of the individual plant in order to construct a distribution function.

2.2. Database

The parameters are normally determined by an expert panel based upon the importance estimated by NUREG-1150 draft version and the interest of Reactor Safety Committee.[1] The special feature of this determination procedure is that through a series of conferences, several groups of experts analyze the results of various experiments, TMI materials and various accident codes, synthetically combine them, and they derive a CDF for each parameter, which is supposed to adequately reflect the collective uncertainties of the parameter. To derive a CDF, each expert individually estimates the distribution of the parameter based upon his and/or her personal experiences and knowledges, and these distributions are taken an average to obtain the final CDF. NUREG-1150 defines the CDF with nine reliability levels (min, 1%, 5%, 25%, 50%, 75%, 95%, 99%, max). And radionuclides are categorized to nine groups as presented in Table 2.

The database presented in the NUREG-1150 is made of opinions of various experts and tries to envelop all the PWR plants. Hence, its application would have limitations, and it is not so justifiable to apply the same database to an individual plant.[2] NUREG/CR-4551 which supports the technical basis of NUREG-1150 divides the parameters into two; general plant parameters not affected by structures and characteristics of a plant, and plant-specific parameters significantly affected by those of a plant.[3]

There are FCOR, FVES and FLATE in general plant parameters. The zirconium oxidation is considered as the most important factor for FCOR and it is divided into two cases; 'lower than 50% oxidation', and 'higher than 50% oxidation'. The pressure of RCS is considered as the most important factor for FVES and it is divided into three cases; 'higher than 2000 psia', 'between 200 and 2000 psia', and 'lower than 200 psia'. The number of openings of the reactor vessel is considered as the most important factor for FLATE and it is divided into two cases; 'one opening after vessel failure', and 'two openings after vessel failure'. And in the two-opening case, the cooling water at the bottom of the containment could inhibit the revaporization to provide assistant heat sink to the primary system. Therefore, it is divided into two cases in addition; 'dry containment', and 'wet containment'.

There are FDCH, FCCI, FCONC and FCONV in plant specific parameters. The pressure of RCS is considered as the most important factor for FDCH and it is divided into two cases; '17MPa', and '7MPa'. The existence of cavity water and the amount of non-oxidized zirconium in corium are considered as the important factors for FCCI and it is divided into four cases by compounding 'dry cavity' and 'wet cavity' with 'low zirconium oxidation' and 'high zirconium oxidation'. And Te, Sr, La, Ce and Ba are important radionuclides in the core/concrete interactions. FCONV and FCONC are divided into four cases according to the combination of time and size of the containment failure by compounding 'early failure' and 'late failure' with 'leak size' and 'rupture size'.

There are FREM, FPART, FPME, FISG, FOSG, DFL and DFE which are not treated in NUREG-1150. It is difficult to generalize these parameters since these are highly depending on the accident scenarios and structural characteristics of a plant. Hence, these should be calculated for each accident case.

3. Generation of Parameters

3.1. Phenomenological Uncertainties of FCOR and FVES

For FCOR, the uncertain physical and chemical phenomena are as follows: 1) prediction of fuel temperature and extent of local oxidation(e.g; TCLRUP, FAOX); 2) fuel, structure, and aerosol chemistry; 3) condensed phase transport; 4) boundary layer transport; 5) coolant velocity effects; and 6) fuel geometry(e.g; EPSCUT, EPSCU2, TEU).[3] For FVES, the uncertain physical and chemical phenomena are: 1) chemical interactions between radionuclides and structures; 2) residence time; 3) carrier gas and structure temperatures governed by core blockage formation and steam flow rate(e.g; EPSCUT, EPSCU2, FFRICR, FFRICX); and 4) ratio of structural surface area to gas volume(FACT).[3] Model parameters account for these phenomena in MAAP modeling. They are presented in Tables 3 and 4. Here, the maximum and minimum values of model parameter are what are recommended by MAAP user's guide.[4] Each model parameter

Table 3. Model Parameters Used for FCOR Computation.

NAME	meaning	minimum	maximum	distribution function
EPSCUT	cutoff porosity of intact node	0.000	0.250	Uniform
EPSCU2	cutoff porosity of collapsed core node	0.000	0.350	Uniform
TEU	core node eutectic melting temp.	2100	2800	Uniform
TCLRUP	the temp. at which cladding fails	1000.	2300.	Uniform
FAOX	multiplier for cladding outside surface area	1.00	2.00	Uniform
FZORUP	Zr oxidation fraction to prevent cladding rupture	0.300	1.00	Uniform

Table 4. Model Parameters Used for FVES Computation.

NAME	meaning	minimum	maximum	distribution function
EPSCUT	cutoff porosity of intact node	0.000	0.250	Uniform
EPSCU2	cutoff porosity of collapsed core node	0.000	0.350	Uniform
FACT	multiplier to reduce the hydraulic diameter and flow area	0.100	1.00	Uniform
FFRICR	friction coeff. for axial gas flow between core and upper plenum	0.05	1.00	Loguniform
FFRICX	gas cross-flow friction coefficient in core	0.250	0.450	Uniform

is assumed to have a uniform distribution except for FFRICR. because the case that FFRICR has a small value is much important.

3.2. LHS of Model Parameter

To make out the input data sets of selected model parameters for MAAP computation, LHS (Latin Hypercube Sampling) method is used.[5] The CDF of each model parameter is divided into the same interval ranges and they are compounded together for sampling. All model parameters are assumed to be mutually independent. Each model parameter has a default value in MAAP parameter file. And the value could be changed for computation by using a statement, PARAMETER CHANGE in the input deck. Here, the input deck is the one in which the user of MAAP could assign some essential and additional computation conditions.[4]

3.3. Cumulative Distribution Function

Input data sets are successively substituted in the input deck, and FCOR and FVES are computed for each accident scenario. The results of FCOR are obtained for two cases; high and low Zr oxidation. And the results of FVES are obtained for two cases; high and low RCS pressure. For each of the cases, the largest value is determined as 100% value and 0 is adopted as 0% value. And the remained values are arranged within the range of (0%, 100%) according to their magnitudes. Finally a CDF is obtained by using interpolation.

4. Results

For a case study, the above method is applied to

Table 5. Result ; FCOR, High Zr Oxidation Case.

Group	Cumulative Probability						
	0.050	0.250	0.500	0.750	0.950	0.990	1.000
Noble	99.510	100.000	100.000	100.000	100.000	100.000	100.000
I	97.300	97.550	99.300	99.800	00.000	100.000	100.000
Cs	97.300	97.550	99.300	99.800	100.000	100.000	100.000
Te	.928	2.085	2.440	2.520	2.740	2.794	2.810
Ba	1.931	12.586	17.900	25.715	45.900	48.454	49.200
Sr	.247	1.188	1.610	2.284	5.820	5.875	5.890
Ru	8.892	40.200	50.000	66.750	76.700	77.945	78.300
La	.028	.627	.685	3.270	16.200	21.149	22.800
Ce	.121	2.483	4.630	4.223	65.200	66.056	66.300

Table 6. Result ; FCOR, Low Zr Oxidation Case.

Group	Cumulative Probability						
	0.050	0.250	0.500	0.750	0.950	0.990	1.000
Noble	31.480	100.000	100.000	100.000	100.000	100.000	100.000
I	30.053	97.600	99.600	99.850	100.000	100.000	100.000
Cs	30.053	97.600	99.600	99.850	100.000	100.000	100.000
Te	.265	1.500	1.740	2.045	2.312	2.326	2.330
Ba	.491	3.602	6.560	11.657	16.123	20.448	21.700
Sr	.077	.403	.658	1.085	1.495	1.962	2.100
Ru	3.538	17.693	23.700	31.690	41.520	44.675	45.500
La	.000	.008	.099	.982	3.850	4.562	4.760
Ce	.000	.022	1.250	6.822	14.130	19.621	21.300

Kori Unit 1 to construct CDF' s of FCOR and FVES, which are considered as the important basic accident source term parameters.[6] The target scenarios for the computation are AHF, S2HF and TMLB' whose initial events are large LOCA, small LOCA and transient, respectively. They are typical severe accident scenarios whose occurrence probabilities are comparatively large and hazards are very serious.[7] The results are shown in Tables 5 through 8 and those of NUREG-1150 are shown in Table 9 through 12. It is found that the distribution functions obtained in this study are roughly consistent to those of NUREG-1150, and there would be negligible differences. However, the distribution functions of Te FCOR are conspicuously different from those of NUREG-1150. It is our opinion that it is because this study uses 0% for the value of FTEREL, by a recommendation of MAAP user' s guide, which is the model parameter of MAAP code determining the chemical compounding of Te and zircaloy.[4] The study of NUREG-1150 assumes that the release fractions for most of radionuclides are 100% at 100% values of CDF' s. Therefore, for high probability regions, the results of NUREG-1150 have many differences from those of this study. But generally, the values of high probability regions are not used in the parametric model method. In overall, the results of this study show that this method is adequate in assessing the severe accident source term parameters of Kori Unit 1. And the more scenarios considered, the more reasonable the results would be.

Table 7. Result ; FVES, High Pressure Case.

Group	Cumulative Probability						
	0.050	0.250	0.500	0.750	0.950	0.990	1.000
Noble	.001	99.900	100.000	100.000	100.000	100.000	100.000
I	.001	38.250	39.900	41.100	43.845	42.120	46.100
Cs	.001	39.350	40.000	41.449	43.955	42.384	46.000
Te	.001	32.698	34.000	42.553	48.374	47.111	50.000
Ba	.000	14.491	20.600	38.647	42.031	41.043	43.300
Sr	.000	5.842	22.500	39.647	43.629	42.563	45.000
Ru	.000	17.362	21.300	27.556	36.884	36.014	38.000
La	.000	9.862	16.900	37.099	44.746	43.021	47.000
Ce	.000	7.501	15.100	40.750	62.042	47.558	86.500

Table 8. Result ; FVES, Low Pressure Case.

Group	Cumulative Probability						
	0.050	0.250	0.500	0.750	0.950	0.990	1.000
Noble	31.537	99.900	100.000	100.000	100.000	100.000	100.000
I	11.850	40.396	47.900	49.897	57.822	60.192	60.800
Cs	10.893	37.926	48.100	50.049	56.018	60.676	61.900
Te	15.967	51.296	97.600	99.300	100.000	100.000	100.000
Ba	11.085	35.193	46.200	82.096	92.778	94.073	94.400
Sr	11.309	37.002	77.800	82.448	92.469	93.692	94.000
Ru	10.797	33.888	39.600	78.305	90.244	92.760	93.400
La	9.895	31.432	65.600	75.268	81.479	90.494	92.900
Ce	9.959	32.002	48.200	70.123	83.721	89.102	90.500

Table 9. NUREG-1150 ; FCOR, High Zr Oxidation Case.

Group	Cumulative Probability						
	0.050	0.250	0.500	0.750	0.950	0.990	1.000
Noble	41.968	80.323	92.049	99.905	99.983	99.998	100.000
I	26.438	55.733	75.048	95.512	100.000	100.000	100.000
Cs	17.423	41.747	61.568	88.761	100.000	100.000	100.000
Te	1.798	9.748	33.084	59.090	91.364	98.636	100.000
Ba	0.118	0.419	0.858	3.006	52.448	100.000	100.000
Sr	0.025	0.211	0.639	1.764	51.659	100.000	100.000
Ru	0.000	0.005	0.456	1.988	8.088	14.033	26.699
La	0.000	0.002	0.010	0.118	2.143	9.979	11.054
Ce	0.000	0.002	0.015	0.303	8.540	50.951	100.000

Table 10. NUREG-1150 ; FCOR, Low Zr Oxidation Case.

Group	Cumulative Probability						
	0.050	0.250	0.500	0.750	0.950	0.990	1.000
Noble	17.590	60.476	90.000	99.905	99.983	99.998	100.000
I	8.394	37.105	69.470	91.035	100.000	100.000	100.000
Cs	6.703	30.309	58.536	83.007	100.000	100.000	100.000
Te	1.265	7.579	19.595	46.087	88.729	98.220	100.000
Ba	0.022	0.174	0.645	2.744	52.448	100.000	100.000
Sr	0.015	0.076	0.402	1.336	51.659	100.000	100.000
Ru	0.000	0.005	0.204	1.228	5.812	14.033	26.699
La	0.000	0.002	0.010	0.095	2.141	9.979	11.054
Ce	0.000	0.002	0.015	0.249	8.540	50.951	100.000

Table 11. NUREG-1150 ; FVES, High Pressure Case.

Group	Cumulative Probability						
	0.050	0.250	0.500	0.750	0.950	0.990	1.000
Noble	100.000	100.000	100.000	100.000	100.000	100.000	100.000
I	1.088	19.547	41.040	60.764	89.121	99.219	100.000
Cs	0.895	13.082	29.469	58.548	89.121	99.219	100.000
Te	0.804	11.815	24.843	42.903	88.806	99.219	100.000
Ba	0.858	12.576	23.813	37.186	86.977	99.219	100.000
Sr	0.858	12.576	23.813	37.186	86.977	99.219	100.000
Ru	0.858	12.576	23.813	37.186	86.977	99.219	100.000
La	0.858	12.576	23.813	37.186	86.977	99.219	100.000
Ce	0.858	12.576	23.813	37.186	86.977	99.219	100.000

Table 12. NUREG-1150 ; FVES, Low Pressure Case.

Group	Cumulative Probability						
	0.050	0.250	0.500	0.750	0.950	0.990	1.000
Noble	100.000	100.000	100.000	100.000	100.000	100.000	100.000
I	11.502	31.079	51.631	86.524	99.314	99.902	100.000
Cs	7.199	20.402	40.150	86.524	99.314	99.902	100.000
Te	4.041	16.682	33.295	66.713	99.167	99.833	100.000
Ba	4.041	16.682	33.295	61.773	99.167	99.833	100.000
Sr	4.041	16.682	33.295	61.773	99.167	99.833	100.000
Ru	4.041	16.682	33.295	61.773	99.167	99.833	100.000
La	4.041	16.682	33.295	61.773	99.167	99.833	100.000
Ce	4.041	16.682	33.295	61.773	99.167	99.833	100.000

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