

Uncertainty Analysis on Containment Failure Frequency for a Japanese PWR Plant

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ABSTRACT

JNES performed, as a part of level 2 PSA projects, the uncertainty analysis of containment failure frequencies for a Japanese 4-loop PWR plant with a pre-stressed concrete containment. Probability distribution functions were determined for five parameters of the severe accident phenomena: (1) over-temperature failure of primary system, (2) over-temperature failure of steam generator tube, (3) oxidization rate of zirconium in reactor vessel, (4) discharge rate of melted core debris at reactor vessel failure, and (5) containment load. In parallel, uncertainty probability distributions of in-vessel and ex-vessel steam explosion, direct containment heating, and debris coolability in the reactor cavity were estimated by the ROAAM method. These probability distributions were taken into account in the containment event tree, the probability propagation calculation was performed, and the uncertainty of containment failure frequency was estimated including the uncertainty of level 1PSA.

The Latin Hypercube sampling method was applied to perform uncertainty analysis with a sample size of 200 to each probability distribution function using the PREP/SPOP code, which was developed as a Monte Carlo simulations program.

The calculated results showed that the uncertainty band of each containment failure modes leading to late containment failure was not large, such as containment bypass, and late over-pressurization excluding a basemat melt-through mode. The uncertainty bands of early containment failure modes such as in-vessel and ex-vessel steam explosions estimated by the ROAAM method were relatively large, but they did not contribute much to the total containment failure frequency.

1. Introduction

With a primary objective of estimating containment performance, the level 2 PSA by uncertainty estimate was executed for a typical Japanese 1,100 MWe PWR plant. In the level 2PSA, it is necessary to estimate the phenomenological uncertainty associated with phenomena such as steam explosions, direct containment heating, and debris cooling. The evaluation methodology of probability distributions by the ROAAM method applying experiment results for simulated severe accident phenomena, and analysis results using the MELCOR code [1-2], and computerized fluid dynamic codes was established. Quantification of CET was carried out considering the phenomenological probability distributions. The uncertainty widths of such phenomena and the total containment failure frequency were quantitatively clarified taking account of the core damage frequency uncertainty results for obtained previously.

2. Outline of the Reference PWR Plant

The reference plant is a Japanese 4-loop PWR plant. The design characteristics are as follows:

- (1) core thermal power of 3,411MW,
- (2) pre-stressed concrete containment with design pressure of 439kPa and volume of 73,700 m³,
- (3) two-train safety injection system and three auxiliary feed water pumps,
- (4) pumps of the containment spray system and the safety injection system take suction from the in-containment, re-circulation sump, and the water resource is necessary to switch from the refueling water storage pit.

2.1 Mitigation Measures for Severe Accident Management

The AM mitigation measures [3] are shown in Figure 2.1. These AMs were considered (1) the natural convection cooling by containment cooling units for normal operation, (2) fire water injection into the containment, (3) the forced depressurization of primary system by pressurizer PORVs, (4) the restoration of containment spray system, and (5) water injection into the primary system by charging pumps. In addition to those countermeasures, cooling down and re-circulation as the SGTR measurement was considered in the present study.

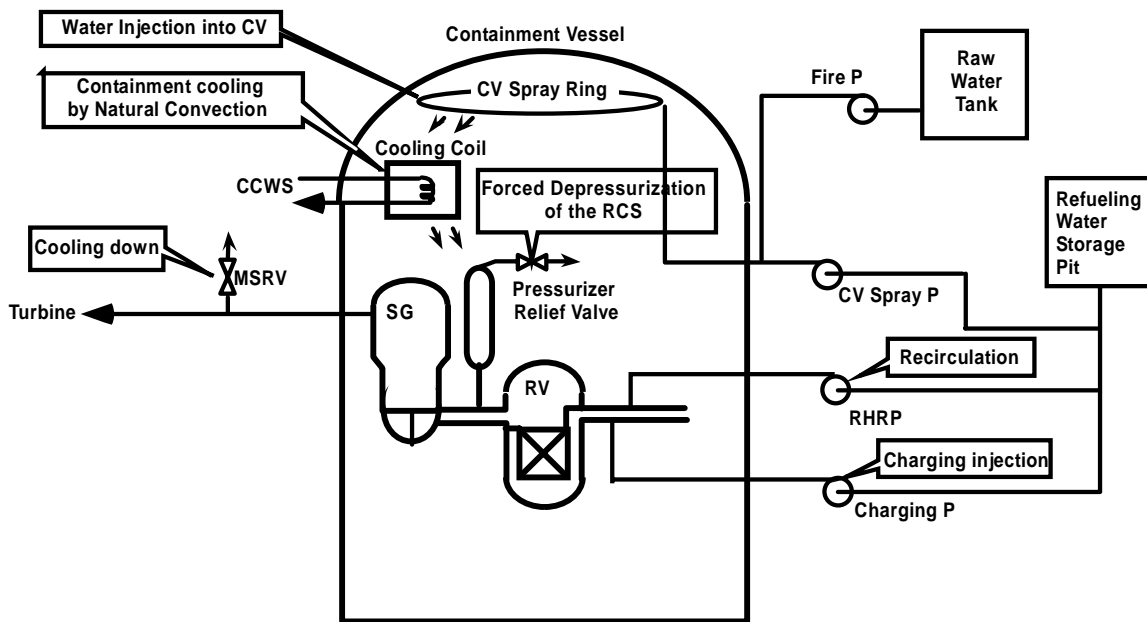


Figure 2.1 Accident Management Measures

3. Level 1 - Level 2 Interface

3.1 Plant Damage States

The primary objective of plant damage states (PDS) analysis is to combine event sequences from the level 1 accident analysis that result in similar severe accident progression, containment response, and fission product release to the environment. The PDSs provide initial and boundary conditions to perform level 2PSA with the systematic scheme.

Table 3.1 shows PDSs of the PWR Plant in the present study. The accident sequences obtained from the level 1 PSA were grouped into thirteen (16) PDSs by considering the similarities of accident progression, and containment response.

Table 3.1 PDSs of a Japanese PWR Plant

Symbol	Accident type	Timing of core damage	Containment response
AEC	Large and medium break LOCA	Early core damage	CV spray on
AE	Large and medium break LOCA	Early core damage	CV spray off
ALC	Large and medium break LOCA	Late core damage	CV spray on
AL	Large and medium break LOCA	Late core damage	CV spray off
SEC	Small break LOCA	Early core damage	CV spray on
SE	Small break LOCA	Early core damage	CV spray off
SE'	SBO & pump seal LOCA	Early core damage	CV spray off
SE''	CCWS failure & pump Seal LOCA	Early core damage	CV spray off
SLC	Small break LOCA	Late core damage	CV spray on
SL	Small break LOCA	Late core damage	CV spray off
TEC	Transient	Early core damage	CV spray on
TE	Transient	Early core damage	CV spray off
TE'	Transient & SBO	Early core damage	CV spray off
P	Containment failure prior to core damage		CV spray off
G	SGTR		
V	Interfacing system LOCA		

3.2 Plant Damage States Analysis and Quantification

Using the WinNUPRA code [4], 1,000 sampling calculation was performed for minimal cut sets of each PDS. The core damage frequency as an average value was obtained to be 8.8×10^{-8} /RY. The uncertainty width of total core damage frequency based 95% value and 5% value was obtained to be two digits

The detailed uncertainty results of PDSs were summarized in Appendix A.

4. Construction of Containment Event Trees

4.1 Containment Event Trees Definition and Analysis

The containment event trees (CET) provide a systematic approach for evaluation of accident sequences that lead to containment failure in coping with severe accident. These are used to characterize the progression of severe accident and containment failures modes that lead to fission product releases beyond the containment boundary. The CET structure and nodal questions address all of the relevant issues important to severe accident progression, containment response, failure, and source terms. Accident recovery and/or management actions must remain consistent between the level 1 PRA and the CET analysis.

4.2 CET for the PWR plant

For the PWR plant, the CETs with the AMs were developed to trace the interdependent physico-chemical processes influencing severe accident progression in the reactor system and the containment as shown in Figure 4.1 through Figure 4.4. A large CET approach was selected to present accident progression with twenty seven top events that are called headings. Seventeen headings have probability distribution. Six headings (BM1, BM2, CM1, CM2, DM1 and DM2) deal with the mitigation AMs of the severe accident. The heading (B1, C4, C5 and D1) concern with four severe accident phenomena and are treated with the ROAAM method [5]. The same split probability as the point-estimate evaluation

was used for ten top events that did not consider the probability distribution.

The end points of CETs that were relevant to the integrity and retention capability of the containment were attributed to containment failure modes shown in Table 4.1.

Table 4.1 Containment Failure Modes

Mode	Definition
α	An in-vessel steam explosion generates a missile of upper head top which causes a containment failure.
β	Loss of containment isolation leads to large leakage early in the accident.
γ	Hydrogen produced in core prior to reactor vessel failure burns rapidly in containment to cause a containment failure.
γ'	Hydrogen released to the containment at the time of reactor vessel failure burns to cause a containment failure.
γ''	Failure by gamma or γ' mode does not occur and combustibles accumulated late after reactor vessel failure burn to cause a containment failure.
δ	Steam and non-condensable gases accumulate by core-coolant and core-concrete interactions and containment fails by overpressurization.
ϵ	Containment fails by besemat melt-through due to core-concrete interaction.
η	Containment fails by steam explosion in reactor cavity.
θ	Containment fails by overpressurization due to a loss of containment engineered safety features (containment failure prior to core damage).
σ	Core debris is ejected from the reactor vessel at high velocity and containment fails by direct containment heating.
ξ	Radioactivities are released from the ruptured tube of steam generator tube to the environment through the secondary system bypassing the containment.
ξ'	Radioactivities are released to the environment by the temperature induced SGTR.
ν	Radioactivities are released from the break in primary coolant system to the environment through the auxiliary building bypassing the containment.

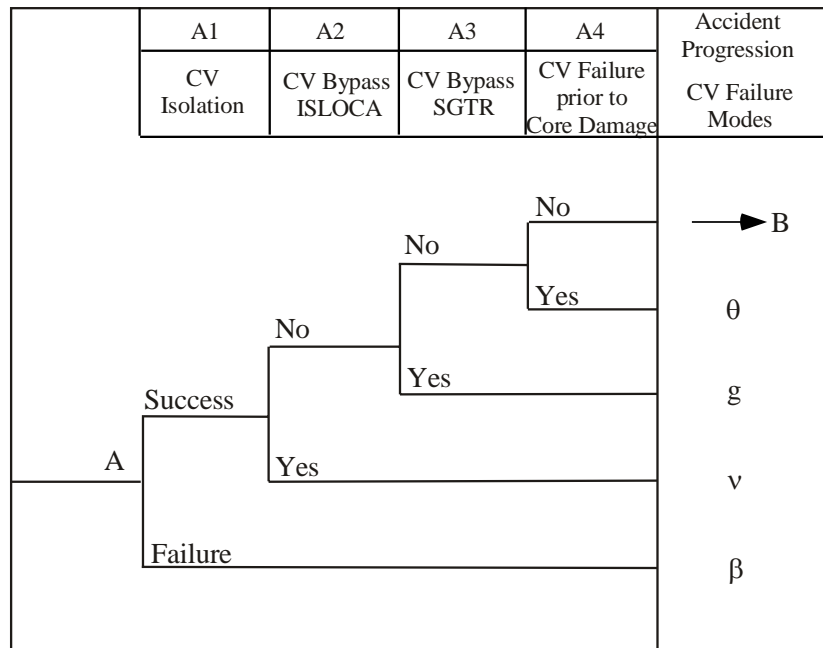


Figure 4.1 CET for a period from the accident initiation to the initiation of core damage (Japanese PWR plant)

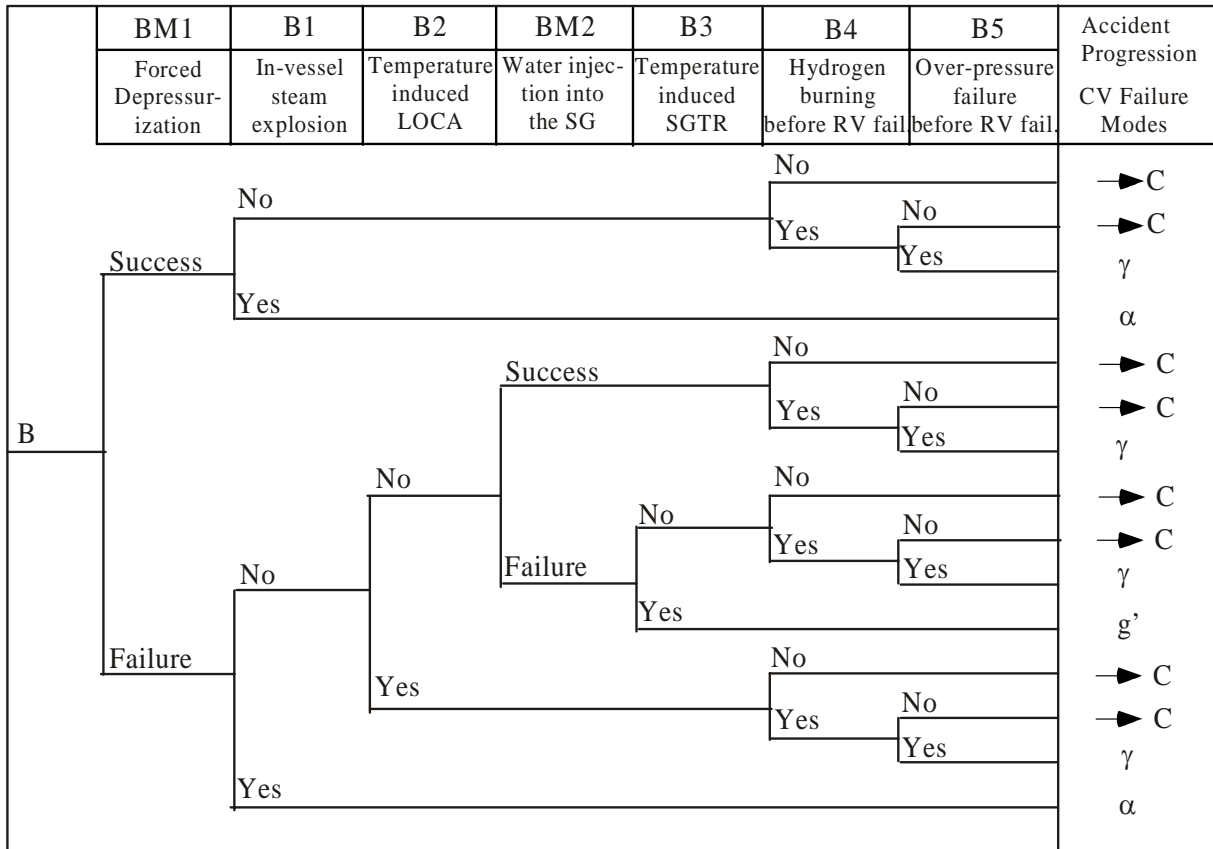


Figure 4.2 CET for a period from the initiation of core damage to the reactor vessel failure (Japanese PWR plant)

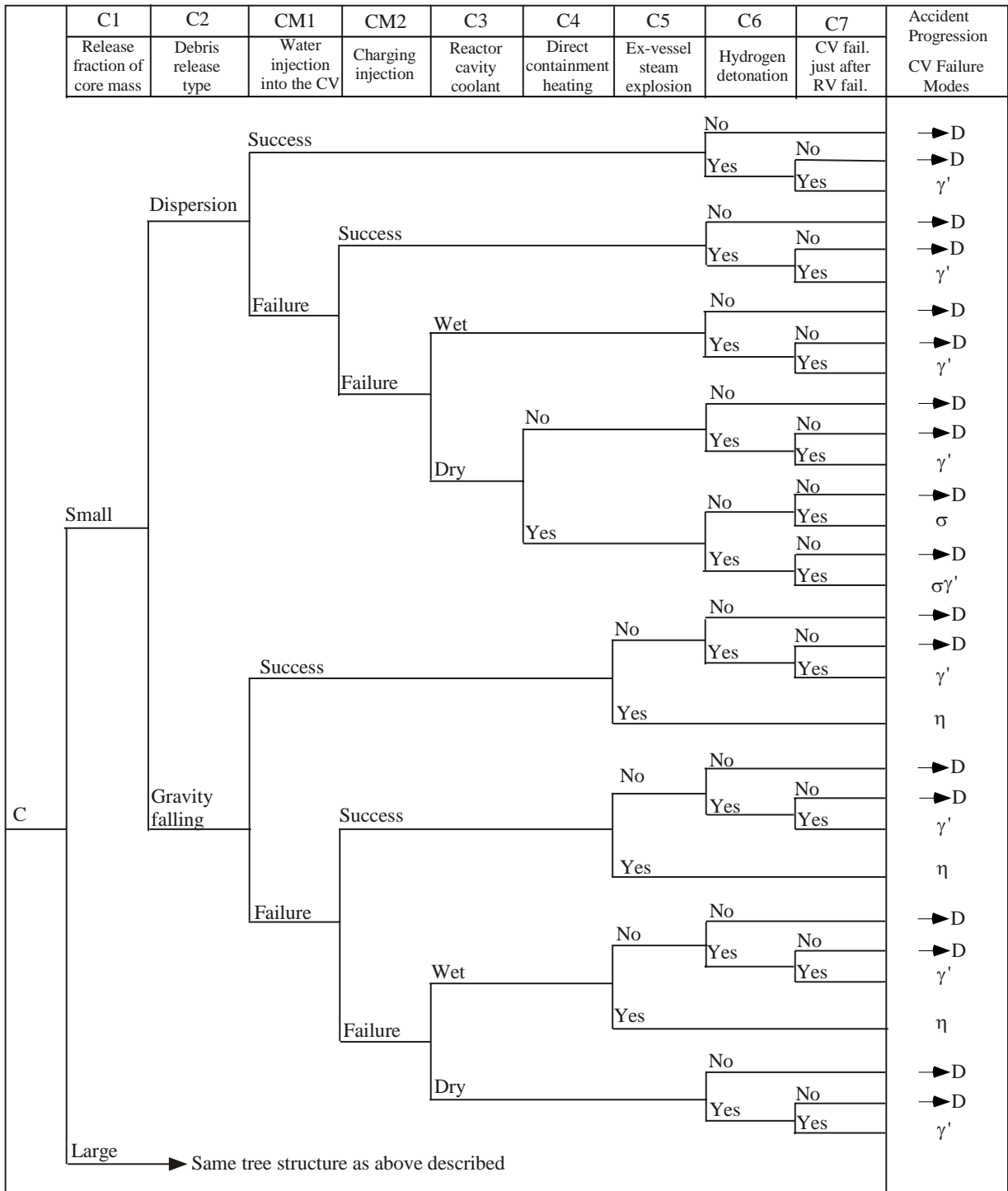


Figure 4.3 CET for an early phase after the reactor vessel failure (Japanese PWR plant)

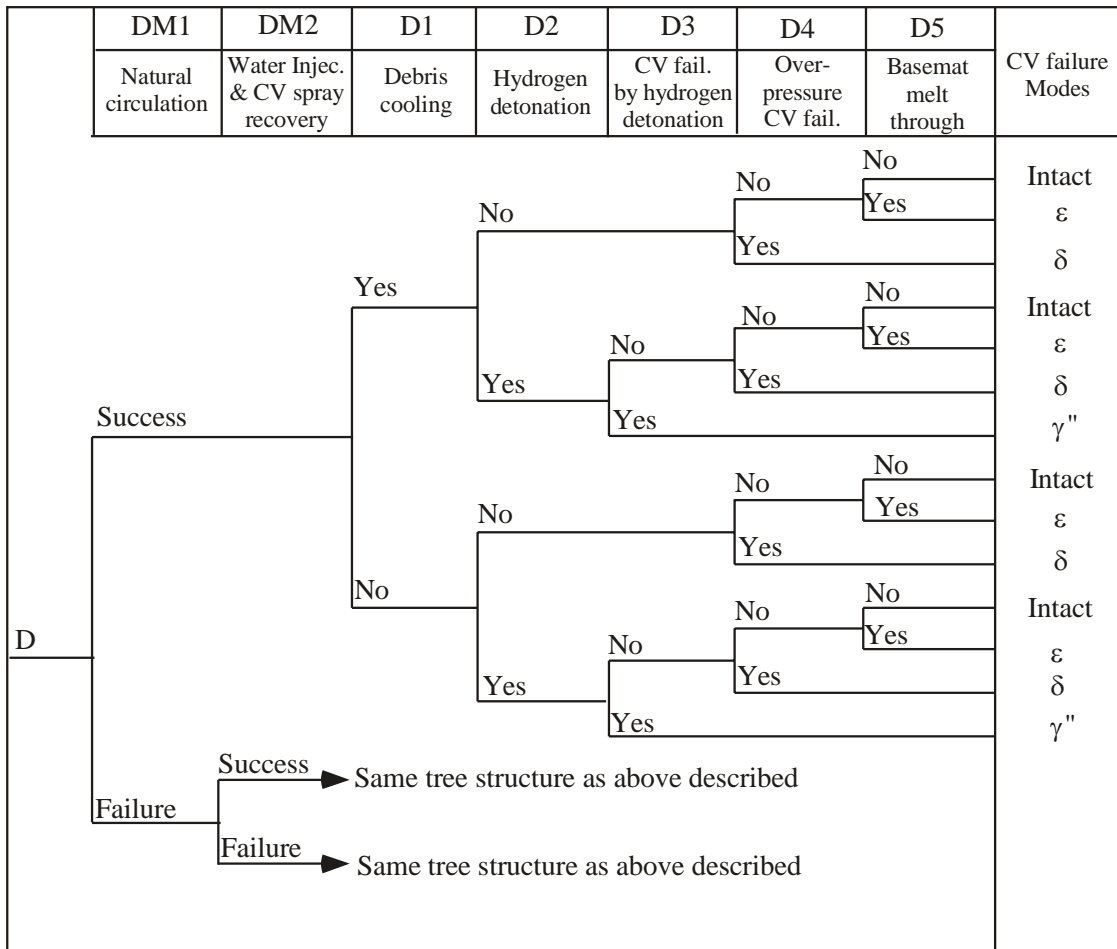


Figure 4.4 CET for a long-term period after the reactor vessel failure (Japanese PWR plant)

5. Uncertainty Analysis of Containment Failure Frequency

By using probability distributions of headings concerning mitigation AMs, and probability distributions obtained by the ROAAM method of headings corresponding to in-vessel steam explosion, ex-vessel steam explosion, direct containment heating and cooling of debris in reactor cavity, uncertainty distributions of containment failure frequency and release categories were obtained by means of the uncertainty propagation analysis of the containment event tree.

5.1 Uncertainty probabilitis of the CET headings for the PWR plant

Branching probabilities of headings with uncertainty are given in Table 5.1. Regarding headings of BM1, BM2, CM1, CM2, DM1, and DM2 concerning mitigation AMs, the uncertainty distributions were calculated in consideration of the uncertainty of equipment failures and the human factor. For headings of the containment event tree concerning in-vessel steam explosion (B1), direct containment heating (C4), ex-vessel steam explosion (C5), and ex-vessel debris cooling (D1), probability distributions were calculated using the ROAAM method. And for headings concerning the temperature induced SGTR (B3), and the hydrogen burning (B4), and the release fraction of core mass (C1), probability distributions were determined by the accident progression analysis results by the MELCOR code for the plant with a method similar to the Zion plant evaluation [6] in NUREG-1150 [7]

The phenomenological event tree shown in Figure 5.1 was developed in order to compute the probability distribution of the containment failure (α mode failure) caused by in-vessel steam explosion. The probability distribution of α mode failure was calculated by 200 samplings by the PREP/SPOP code [8] for the probability distributions of each box which constitutes the phenomenological event tree. The probability distribution of α mode failure calculated is shown in Figure 5.2. The average value was estimated to be 6×10^{-6} . The containment failure probability distribution by the ex-vessel steam explosion was calculated using the ROAAM method. Figure 5.3 shows phenomenological event tree to analyze probability distribution for ex-vessel steam explosion. The probability distribution is shown in Figure 5.4. The average value was obtained to be 2×10^{-4} .

Table 5.1 CET Heading Probability with Uncertainty (1/2)

CET Heading	Method	PDSs	Average	5%	50%	95%
A1: Containment Isolation	Unavailability Calculation	SE'	0.008	0.004	0.007	0.016
		Other PDSs excluding SE', G, V	0.004	0.002	0.003	0.008
BM1: Forced depressurization	Unavailability Calculation	TEC	0.45	0.27	0.35	0.64
		TE	0.54	0.39	0.46	0.70
		SEC	0.37	0.37	0.37	0.37
		SE	0.31	0.31	0.31	0.31
		SLC,SL	0.21	0.21	0.21	0.21
		Other PDSs	1.0	1.0	1.0	1.0
B1: In-vessel steam explosion	ROAAM Analysis	All PDSs excluding P,G,V	6E-06	7E-09	3E-08	1E-06
B2: Temperature induced LOCA	NUREG-1150 Estimation	TEC,TE,TE'	0.37	0.0	0.93	1.0
		Other PDSs	1.0	1.0	1.0	1.0
BM2: Water injection into the SG	Unavailability Calculation	TEC,TE	0.65	0.60	0.62	0.72
		Other PDSs	1.0	1.0	1.0	1.0
B3: Temperature Induced SGTR	Unavailability Calculation	TEC,TE,TE'	0.21	0.0	0.014	0.60
		Other PDSs	1.0	1.0	1.0	1.0
B4: Hydrogen burning before RV fail.	NUREG-1150 Estimation	AEC,AE,ALC,AL	3E-02	1E-04	1E-04	0.1
		SEC,SE,SE',SE''	1E-04	1E-04	1E-04	1E-04
		SLC,SL	1E-04	1E-04	1E-04	1E-04
		TEC,TE,TE'	3E-02	1E-04	1E-04	0.1
C1: Release fraction of core debris mass	NUREG-1150 Estimation	All PDSs excluding P,G,V	0.35	0.0	0.75	1.0
CM1: Water injection into the CV	Unavailability Calculation	SE',TE'	0.60	0.48	0.53	0.73
		Other PDSs excluding P,G,V	0.54	0.42	0.47	0.67
CM2: Charging Injection	Unavailability Calculation	SE',TE'	0.24	0.12	0.17	0.38
		All PDSs excluding P,G,V	0.14	0.022	0.073	0.28
C4: Pressure Load by Direct Containment Heating	ROAAM Analysis	Debris dispersal, Non hydrogen Burning	0.43 MPa	0.20M Pa	0.43M Pa	0.65M Pa
		Debris dispersal, With hydrogen burning	0.46 MPa	0.25 MPa	0.48 MPa	0.70 MPa

Table 5.1 CET Heading Probability with Uncertainty (2/2)

CET Heading	Method	PDSs	Average	5%	50%	95%
C5: Containment Failure by Ex-vessel steam explosion	ROAAM Analysis	All PDSs excluding P,G,V	2E-04	0.0	0.0	1E-03
C7: Containment Failure pressure	Containment Failure Distribution	All PDSs excluding P,G,V	1.2 MPa	0.52 MPa	1.2 MPa	2.0 MPa
DM1: Natural circulation	Unavailability Calculation	SE',TE'	0.042	0.035	0.038	0.051
		Other PDSs excluding P, G, V	0.015	0.008	0.011	0.024
DM2: Water injection & CV spray recovery	Unavailability Calculation	SE',TE'	0.13	0.065	0.081	0.17
		SE''	0.076	0.005	0.025	0.11
		Other PDSs excluding P, G, V	0.072	0.007	0.023	0.11
D1: Debris cooling	ROAAM Analysis	Low pressure sequences, with water injection	0.30	0.005	0.11	0.90
		Low pressure sequences, without water injection	0.67	0.019	0.87	0.99
		High pressure sequences, debris dispersal with water injection	0.02	5E-04	0.006	0.068
		High pressure sequences, debris dispersal without water injection	0.09	0.001	0.033	0.54
		High pressure sequences, non debris dispersal with water injection	0.09	0.001	0.01	0.57
		High pressure sequences, non debris dispersal without water injection	0.21	0.002	0.11	0.94

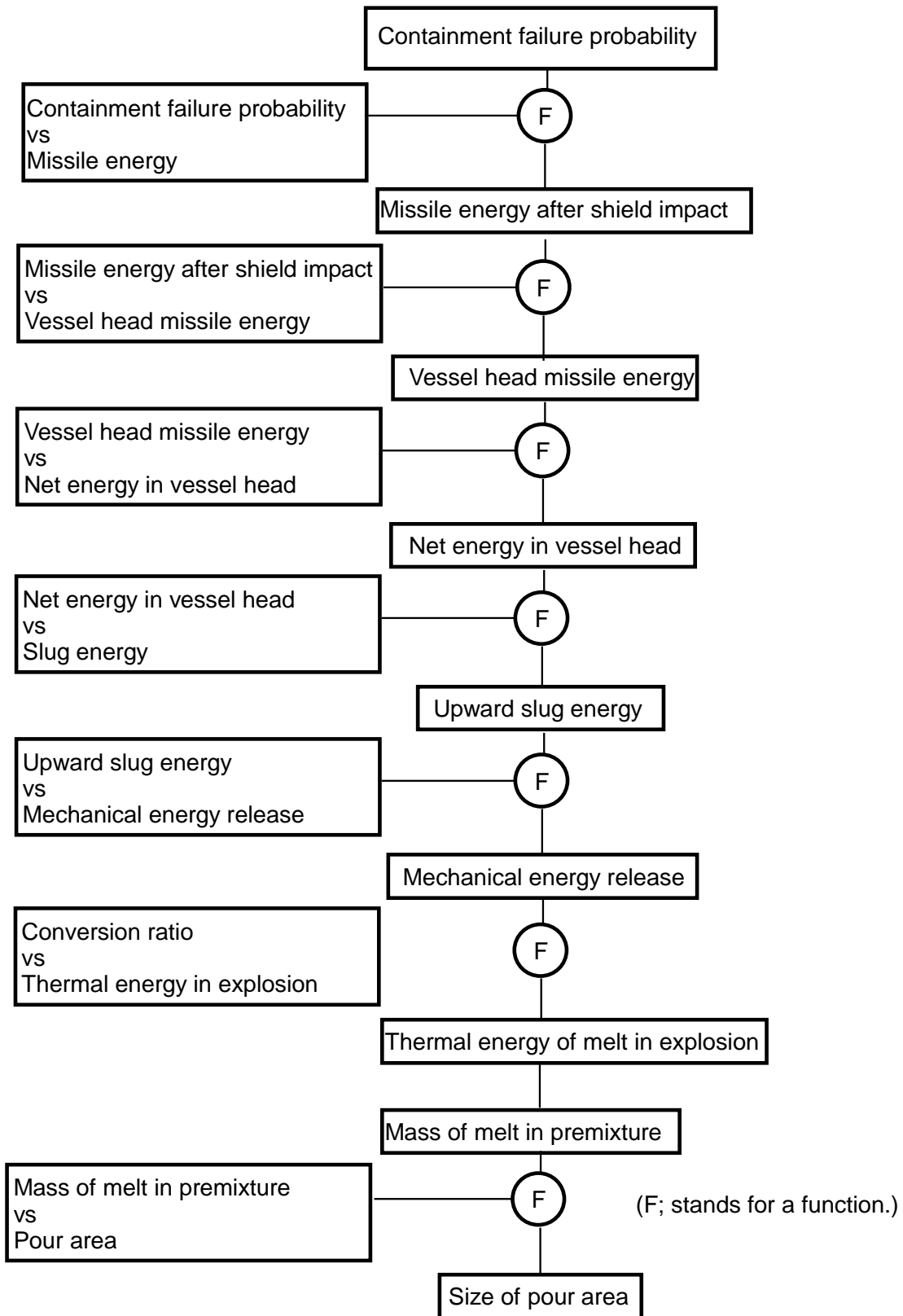


Figure 5.1 Phenomenological Event Tree of ROAAM Method for the In-vessel Steam Explosion

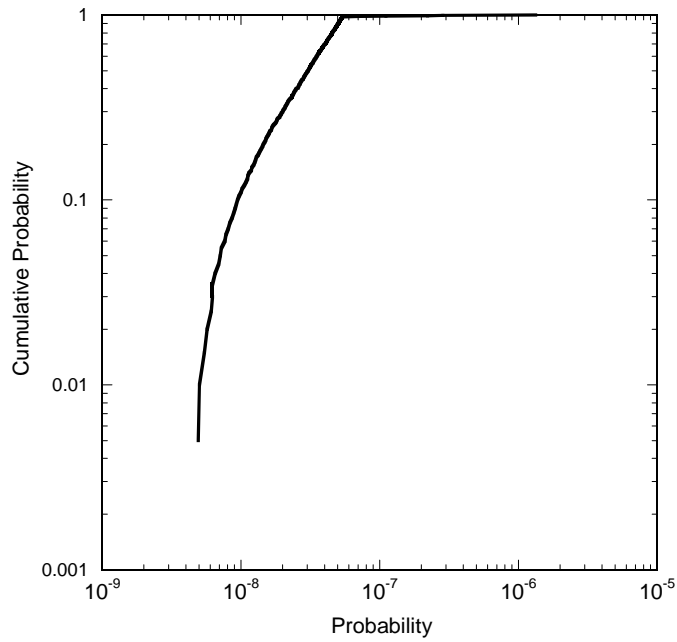


Figure 5.2 Containment Failure Probability of the In-vessel Steam Explosion Obtained by ROAAM

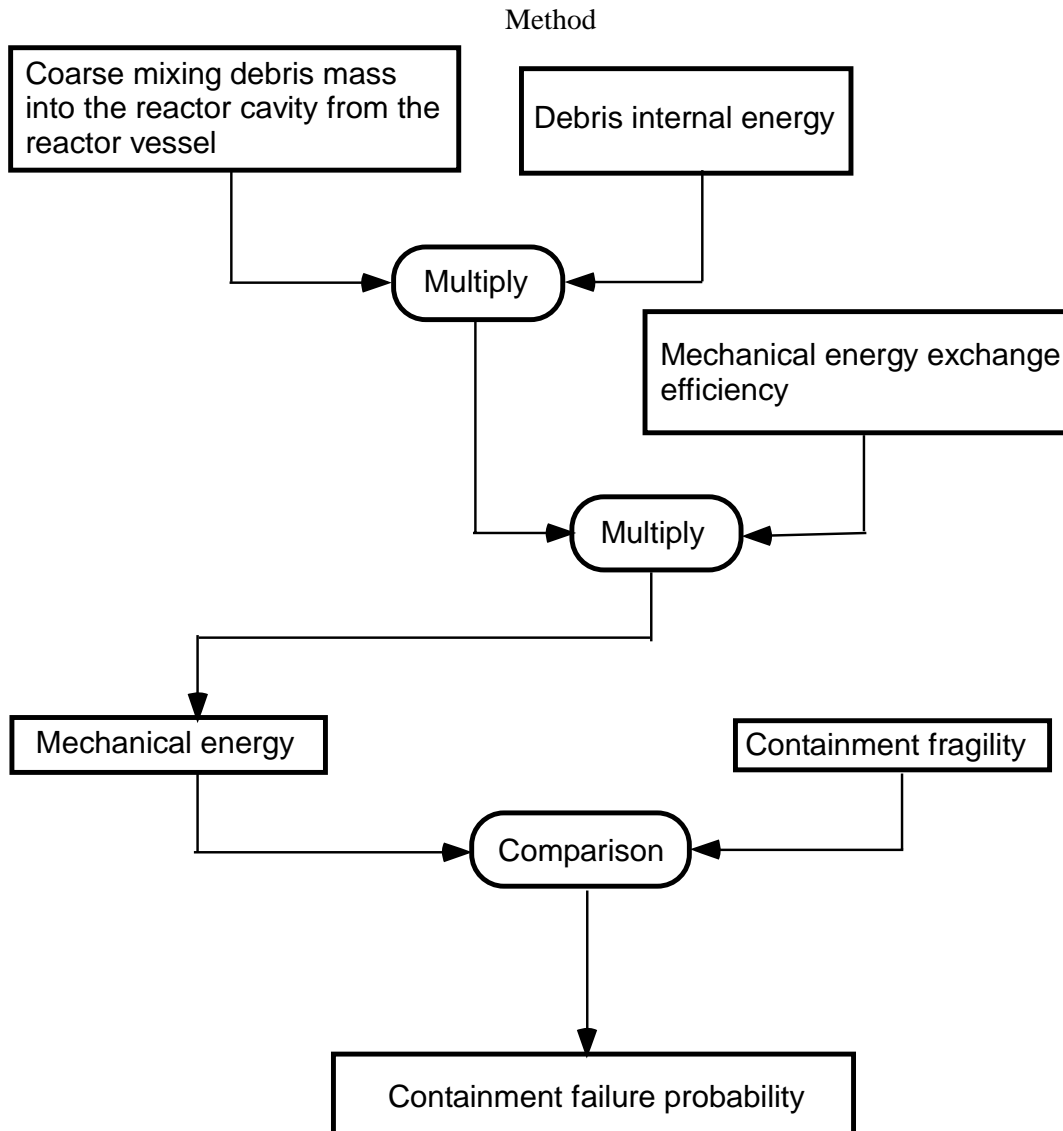


Figure 5.3 Phenomenological Event Tree of ROAAM Method for the Ex-vessel Steam Explosion

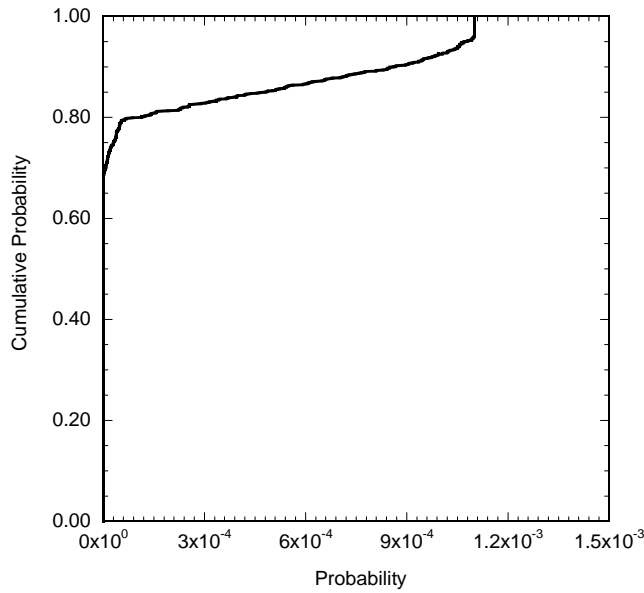


Figure 5.4 Containment Failure Probability of the Ex-vessel Steam Explosion Obtained by ROAAM Method

5.2 Uncertainty Analysis Results of the CET for the PWR plant

5.2.1 Containment Failure Frequency

Uncertainty values of the containment failure frequency for each containment failure mode were obtained to be 1.0×10^{-8} /RY using the PREP/SPOP code. The probability distributions of the containment failure frequency are shown in Fig. 5.3 for each containment failure mode. The uncertainty bands of the containment failure frequency by such phenomena as in-vessel steam explosion (α), direct containment heating (σ), the ex-vessel steam explosion (η), and the hydrogen burning (γ) predicted by the ROAAM method extended three or more figures.

The detailed uncertainty results of containment failure frequency were summarized in Appendix B.

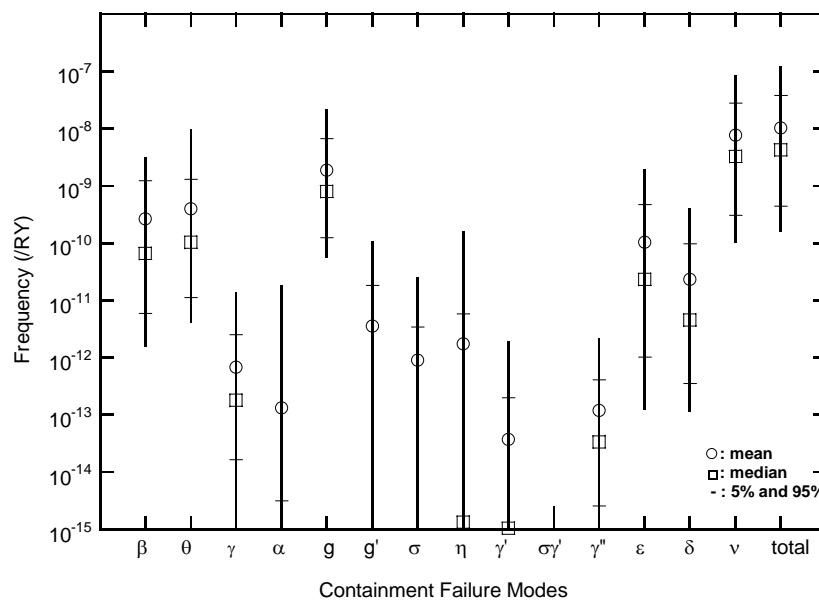


Figure 5.3 Containment Failure Frequency (Uncertainty estimation)

5.2.2 Release Categories

The release categories are obtained by combining PDS in Table 3.1 with containment failure modes in Table 4.1. The containment failure frequencies for release categories are shown in Table 5.2. The release category V-v in which fission products bypass a containment boundary by an IS-LOCA node became dominant to the containment failure frequency. The contribution fraction occupies about 75% of the total frequency. The release category G-g of a containment bypass by SGTR contributed about 18%, and discharge category P-θ which causes core damage after containment failure became about 4%. Discharge categories such as TEC-β and SL-β which are associated with the containment isolation failure estimated to be about 2%. Discharge categories such as SE"-δ and SL-δ which cause a late containment failure by overpressure were about 1%.

The detail uncertainty results of release categories were summarized in Appendix B.

Table 5.2 Frequency for Release Categories

Rank	Release Category	Frequency/(RY) (Average)	Definition
1	V-v	7.7E-09	IS-LOCA+Containment bypass
2	G-g	1.9E-09	SGTR+ECCS failure+Containment bypass
3	P-θ	4.0E-10	Containment failure prior to core damage
4	TEC-β	1.3E-10	Transient+ Early core melt+CV isolation failure
5	SL-ε	5.8E-11	Small LOCA+ Late core melt+CV spray failure+CV melt-through
6	SL-β	4.6E-11	Large LOCA+ Late core melt+CV spray failure+CV melt-through
7	SLC-β	2.5E-11	Large LOCA+ Late core melt+CV spray failure+CV isolation failure
8	AEC-β	2.4E-11	Small LOCA+ Early core melt+CV spray failure+CV melt-through
9	AL-ε	2.2E-11	Large LOCA+ Late core melt+CV isolation failure
10	AL-β	1.6E-11	CCWS failure+RCP seal LOCA+CV overpressure failure
11	SE-ε	9.8E-12	Small LOCA+ Late core melt+CV spray failure+CV overpressure failure
12	ALC-β	9.7E-12	Small LOCA+ Early core melt+CV spray failure+CV isolation failure
13	SE"-δ	6.6E-12	SBO+RCP seal LOCA+CV isolation failure
14	SL-δ	5.8E-12	Transient+ Early core melt+CV melt-through
15	SE-β	4.0E-12	Small LOCA+ Early core melt+CV spray failure+CV overpressure failure

6. Source Terms Uncertainty Analysis

6.1 Utilization of Source Term Evaluation Equations

For source term uncertainty analysis, the source term evaluation equations in the XSOR code [9] which were provided for the Zion plant in NUREG-1150 were applied. The source term equations in the XSOR consist of thirteen source term parameters shown in Table 6.1. The source term evaluation equations in the XSOR code express the source terms by combination of source term parameters.

The uncertainty distribution of each source term parameter was determined for each FP group consisting of nine groups, Xe, I, Cs, Te, Sr, Rn, La, Ce, and Ba.

Table 6.1 Source Term Parameter Definitions

Parameter	Definition
FCOR(i)	Fraction of initial inventory of nuclide group i release from the fuel in-vessel
FISG(i)	Fraction of fuel release transported to steam generator in an accident
FOSG(i)	Fraction of FISG released from steam generator to the environment
FVES(i)	Fraction of fuel release transported to the containment
FCONV	Containment transport fraction for releases prior to or at vessel breach
DFE	Decontamination factor of spray for in-vessel releases
FREM	Fraction of core remained in vessel after vessel breach (value fixed, 0,05)
FCCI(i)	Fractional release of nuclide group i from corium during molten core-concrete interactions
FCONC(i)	Containment transport fraction for ex-vessel release
DFL	Decontamination factor for the overlying (if any) water pool of CCI release
FPART	Conditional fraction of core involved in high pressure melt ejection accident
VDF	Decontamination factor of auxiliary building pool water in an IS-LOCA sequence
FLATE(i)	Fractional releases of material deposited in RCS due to revaporization rate

6.2 Uncertainty Propagation Analysis for each Release Category

Uncertainty probability distributions of thirteen source term parameters in Table 6.1 were obtained by carrying out 200 sampling calculation by the PREP/SPOP code. In this calculation, the conditions of a source term parameter shown in Table 6.2 for typical six release categories were chosen. The release fraction (average) to environment for each release category calculated by the uncertainty propagation analysis using conditions in Table 6.2 is shown in Table 6.3.

The release fraction of environment corresponding to the release category has been obtained with the MELCOR code for the PWR plant. The above-mentioned probability distributions of the source term were compensated by the calculation result obtained by the MELCOR code.

And the analytical source term corresponding to the release category (SL-ε) of basemat melt-through was obtained by multiplying the soil decontamination factor (5/1000) used in the Zion plant evaluation of NUREG-1150 for the environmental release fraction by the containment overpressure failure obtained by the MELCOR analysis result for the same accident sequence.

The width of the uncertainty by the 5% value and 95% value for FP groups of I and Cs was in the range of a single figure to double figures for a release category V-v of IS-LOCA sequence.

The detailed uncertainty results of source term were summarized in Appendix C.

Table 6.2 Conditional selections of source term parameters for each release category

Release Category	V-v	G-g	P-θ	TEC-β	SL-ε	SE''-δ
FCOR	Low Zr Oxidation				High Zr Oxidation	
FISG	No	SGTR	No			
FOSG	No	Secondary system break	No			
FVES	IS-LOCA	SGTR	Low Pressure	System Set Point	High or Intermediate Pressure	
FCONV	IS-LOCA	No	Early Rupture	Early Leak, Wet Containment	Late Rupture	
DEF	No			High Pressure	No	
FCCI	Low Zr Oxidation, No Water	High Zr Oxidation, No Water	Low Zr Oxidation, No Water		High Zr Oxidation, No Water	
FCONC	IS-LOCA	Early Rupture	Late Rupture	Early Leak, Wet Containment	Late Rupture	
FLATE	Two Holes in RCS			One Hole in RCS	Two Holes in RCS	

Table 6.3 The calculated source term (average) in each release category

Release Category	FP Group								
	Xe	I	Cs	Te	Sr	Rn	La	Ce	Ba
V-v	1.0E+00	4.8E-01	1.9E-01	4.6E-01	3.1E-01	3.7E-02	2.0E-01	3.9E-02	1.2E-01
G-g	1.0E+00	6.3E-01	2.4E-01	6.4E-01	4.1E-01	5.0E-02	2.6E-01	4.4E-02	1.6E-01
P-θ	1.0E+00	5.0E-01	1.9E-01	5.1E-01	3.4E-01	3.9E-02	2.1E-01	3.9E-02	1.3E-01
TEC-β	1.0E+00	4.0E-02	1.2E-02	2.0E-02	1.4E-02	2.4E-03	1.2E-02	1.9E-03	9.5E-03
SL-ε	5.0E-03	1.6E-04	1.6E-07	5.5E-07	3.5E-07	3.6E-08	1.7E-07	2.0E-08	1.1E-07
SE''-δ	1.0E+00	3.2E-02	1.9E-05	3.7E-05	3.1E-05	3.4E-06	2.0E-05	5.6E-06	1.3E-05

7. Conclusions

The present study addressed establishment of the uncertainty evaluation method including the ROAAM method on the containment failure frequency during full power operation for a Japanese 4-loop PWR plant with pre-stressed concrete containment. The average probabilities of containment failure of in-vessel and ex vessel steam explosions calculated by ROAAM method were obtained to be 6×10^{-6} and 2×10^{-4} . The calculated result showed that the average values of total containment failure frequency was obtained to be 1.0×10^{-8} /RY, and containment failure frequency has an uncertainty width of double figures similar to the uncertainty width of core damage frequency.

A source term uncertainty analysis has been performed by typical release categories based NUREG-1150 methodology. The uncertainty width of FP groups of I and Cs for a release category of IS-LOCA sequence, which was dominant, was in the range of single figure to double figures.

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Appendix A Plant Damage States Quantification for a Japanese PWR plant

Table A.1 shows the results of occurrence frequencies and uncertainty bands for each PDS. Figure A.1 shows the fraction of each PDS for the PWR plant and the total core damage frequency. The contribution fraction of TEC in PDSs was the highest and estimated to be about 44%.

Table A.1 Uncertainty Result of each PDS

PDS	Average	5%	50%	95%
AEC	7.0E-09	2.4E-10	2.1E-09	2.7E-08
AE	2.8E-10	1.3E-11	1.1E-10	1.0E-09
ALC	4.4E-09	9.8E-11	8.4E-10	1.1E-08
AL	4.3E-09	2.3E-10	1.6E-09	1.5E-08
SEC	5.9E-10	1.3E-11	1.4E-10	2.1E-09
SE	1.2E-09	3.4E-11	3.6E-10	4.0E-09
SE'	7.3E-10	7.9E-13	4.3E-11	2.3E-09
SE''	3.2E-10	1.8E-13	1.6E-11	1.3E-09
SLC	7.1E-09	2.0E-10	2.2E-09	2.7E-08
SL	1.3E-08	4.2E-10	4.0E-09	5.0E-08
TEC	3.9E-08	8.3E-10	8.1E-09	1.4E-07
TE	2.1E-10	2.4E-11	1.1E-10	7.0E-10
TE'	1.4E-10	2.4E-13	9.3E-12	5.1E-10
P	4.5E-10	1.1E-11	1.1E-10	1.4E-09
G	1.9E-09	1.2E-10	8.1E-10	7.3E-09
V	8.0E-09	3.0E-10	3.3E-09	3.0E-08
Total	8.8E-08	2.5E-09	2.4E-08	3.2E-07

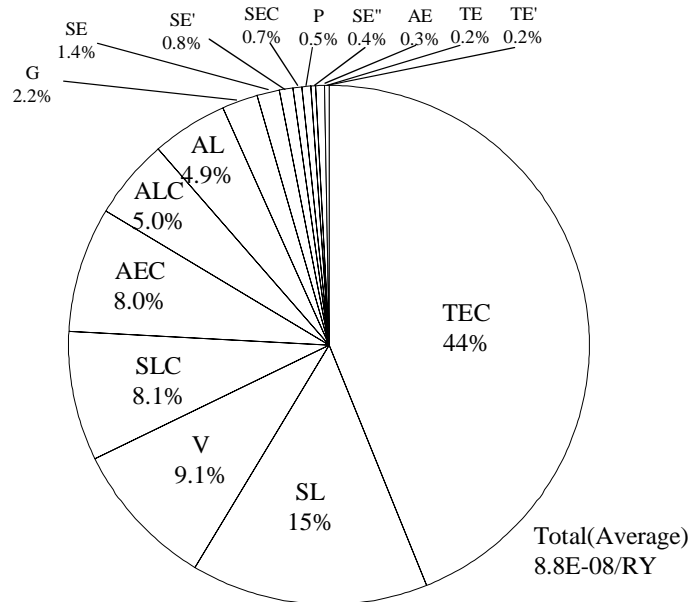


Figure A.1 The core damage fraction by each PDS for a Japanese PWR Plant (With AM)

Appendix B Uncertainty Analysis Results of the CET for a Japanese PWR plant

B.1 Containment Failure Frequency

The average value of total containment failure frequency was estimated to be 1.0×10^{-8} /RY. The core damage frequency is about 8.8×10^{-8} /RY obtained from the level 1PSA. Therefore, the conditional probability of containment failure is about 0.13. For uncertainty bands, 95% value and 5% value of containment failure were estimated to be about 3.9×10^{-8} /RY and about 4.6×10^{-10} /RY respectively. And the calculated results show that the uncertainty band of containment failure frequency was in the range similar to the uncertainty band of core damage frequency.

B.2 Containment Failure Modes

The containment failure fraction measured with average frequencies are shown in Figure B.1 for each containment failure mode. A dominant failure mode of containment failure was a containment bypass (ν) from an interface system LOCA sequence in which a pipe of the residual heat removal system breaks by primary system pressure loading. The other dominant containment failure modes were SGTR (g), basemat melt-through (ϵ), and late overpressure failure (δ). The contributions from energetic phenomena, such as steam explosions (α and η), hydrogen burning (γ), and direct containment heating (σ) were less than 0.1%.

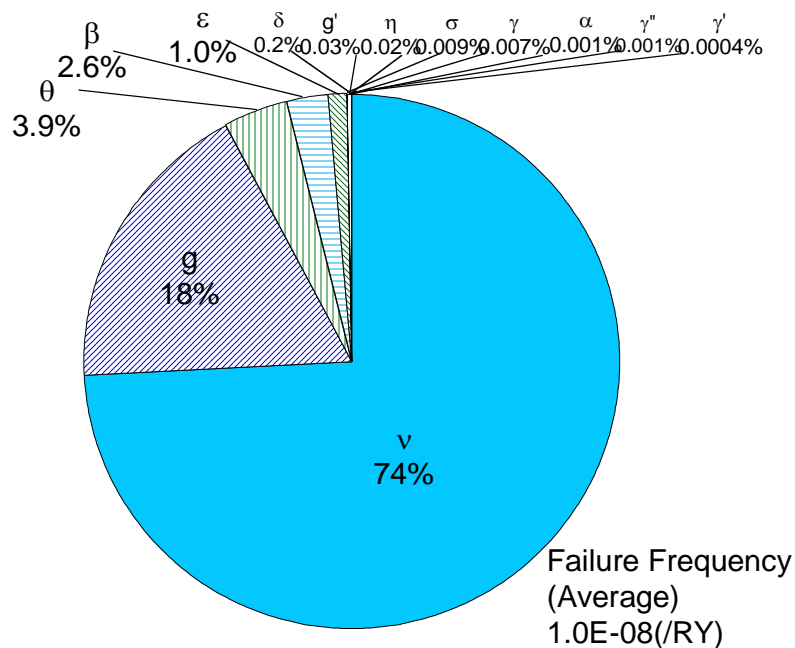


Figure B.1 Containment Failure Fraction (with AM)

B.3 Release Categories

The containment failure frequencies for release categories are shown in Figure B.2. Release category V-v became dominant in terms of containment failure frequency, and frequency, and the frequency of release category was estimated to be about 7.7×10^{-9} /RY (average).

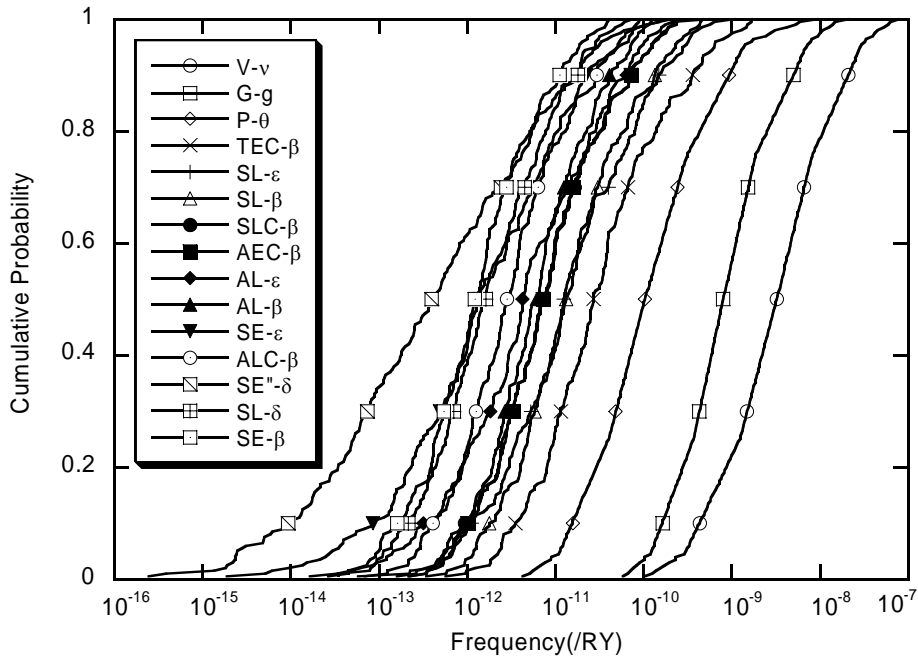


Figure B.2 Frequency of Release Categories (Uncertainty estimation)

Appendix C Results of Source Term Uncertainty

C.1 Results of Source Term Uncertainty for Release Category V-v

Table C.1 shows the release fraction of FP of core inventory to the environment for release category V-v which was compensated by the MELCOR code from the original NUREG-1150 method. Release fractions of high volatility, I, Cs, and Te, were estimated to be 24% to 42% of the initial inventory. As for FP groups with low volatility of Sr and Ba, release fractions to the environment were estimated to be 1.1% to 4.9% of the initial inventory.

The uncertainty width for all FPs group is shown in Figure C.1. The width of the uncertainty by the 5% value and 95% value for FP groups of I and Cs was in the range of a single figure to double figures. There were about seven figures of uncertainty width of volatile high Te group. The uncertainty width for volatile high Te group became about the seven figures. While the physicochemical phenomenon of Te group is uncertain, it shows what is not fully understood. Ba group of low volatile FP has an uncertainty width of about double figures. However, low volatile Sr group of FP has an uncertainty width of about six figures. Moreover, the uncertainty width of FP groups of Ru, La, and Ce was in the range of six to twelve figures, and its uncertainty width of Ru group is exceedingly broad.

The uncertainty distribution for all FP groups is shown in Figure C.2 It should be noted that the cumulative probability curves of Ba, La, and Ce groups have shifted to lower values because of the compensation by the MELCOR calculation coupled with the original NUREG-1150 method.

Table C.1 Compensated Source Term of Release Category V-v

Uncertainty Band	FP Group								
	Xe	I	Cs	Te	Sr	Rn	La	Ce	Ba
5%	1.0E+00	2.3E-02	9.5E-02	8.8E-08	2.0E-07	9.5E-14	2.6E-10	1.2E-06	2.6E-03
50%	1.0E+00	3.5E-01	3.6E-01	6.5E-03	2.5E-03	4.2E-08	1.1E-06	8.6E-05	1.1E-02
95%	1.0E+00	9.1E-01	8.8E-01	3.0E-01	2.9E-01	2.0E-01	2.2E-01	8.3E-01	8.4E-01

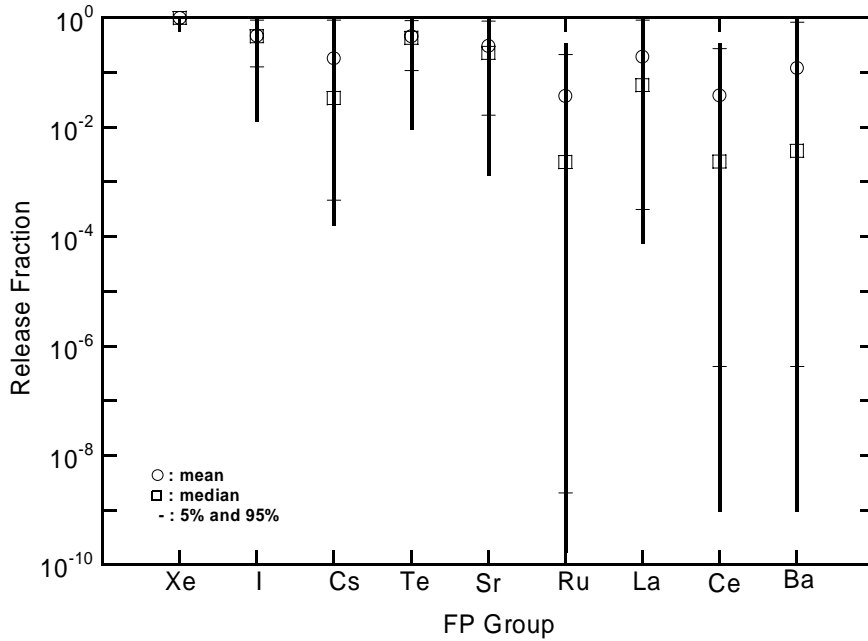


Figure C.1 Release Fraction Distribution of Release Category V-v

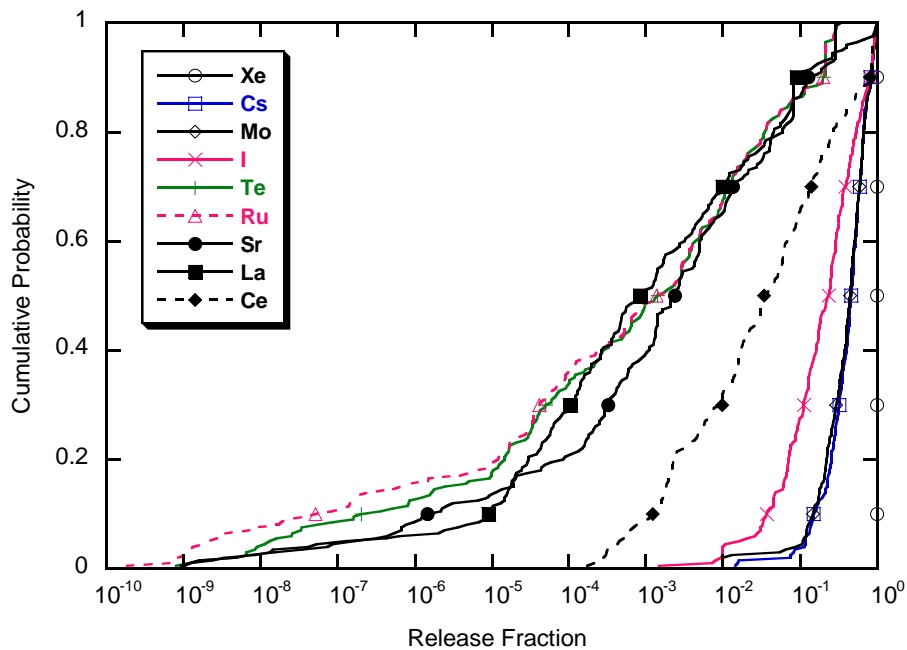


Figure C.2 Probability Distribution of FP Release Fraction for Release Category V-v

C.2 Comparison about a Release Fraction

The ratio of the average value of the probability distributions obtained for five release categories from the calculation result calculated by the MELCOR code is shown in Figure C.3. In the categories where FPs are released to environment at early stage of accidents the ratio of FP group of I, Te, and Cs is over 1.0. The FP release fractions of the release categories such as containment bypass of V-v, G-g, and TEC- β , are underestimated by the propagation analysis. On the other hand, the FP release fraction rate of the release categories such as containment overpressure failure of P- θ and SE''- δ , a ratio is less than 1.0 and the propagation analysis has overestimated the release fraction. Generally, in the five

release categories, the release fraction for FP group of Ru, La, Ce, and Ba obtained by the propagation analysis is 10 or more times larger than the result obtained from the MELCOR analysis.

Moreover, the comparison of the uncertainty width of I group among six release categories is shown in Figure C.4. The average value and 50% value for the release category of V-v, G-g and TEC- β were about 0.1, while for the release category of P- θ , SL- ϵ and SE''- δ are about 0.01. The reason of this difference depends on the fact that the various physicochemical phenomena increase uncertainties along the progression until containment failure.

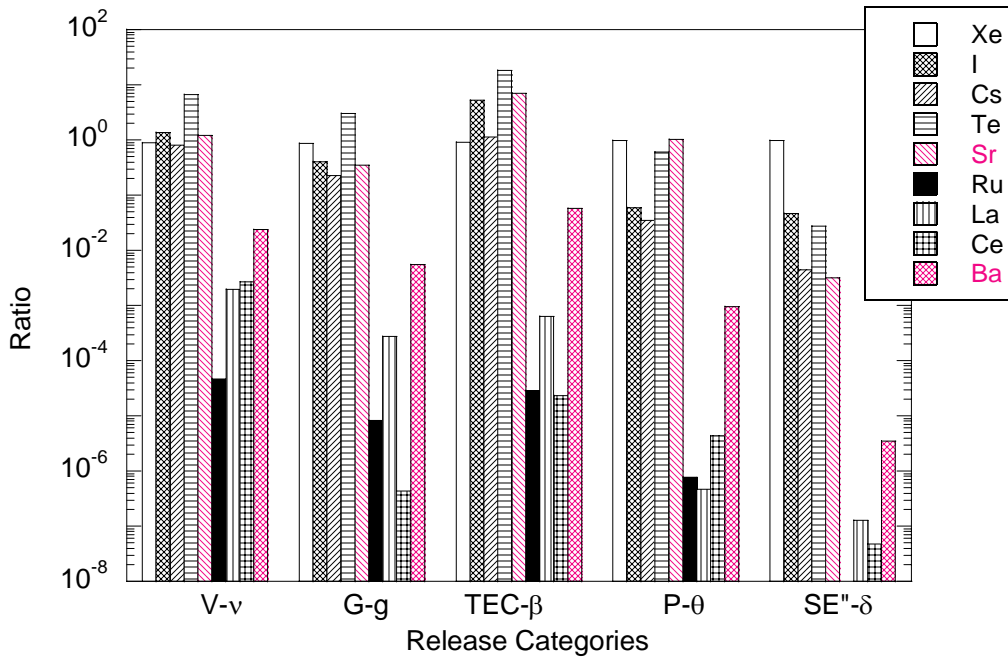


Figure C.3 Ratio of Release Fractions obtained by NUREG-1150 Methodology and MELCOR Calculation

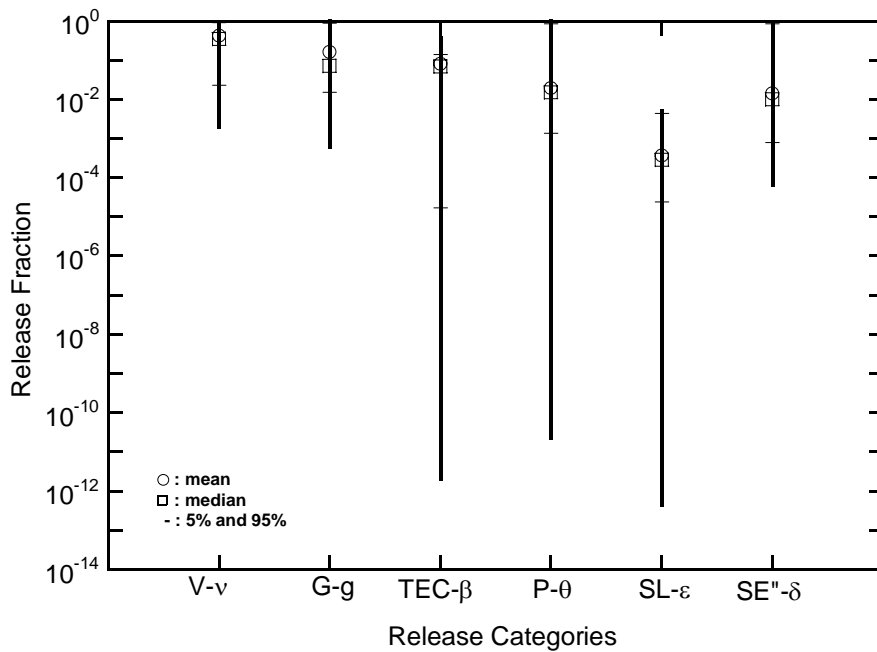


Figure C.4 Release Fraction of FP group I for Six Release Categories