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Uncertainty Analysis on Containment Failure Frequency for a Japanese PWR Plant

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Contents

1. Introduction

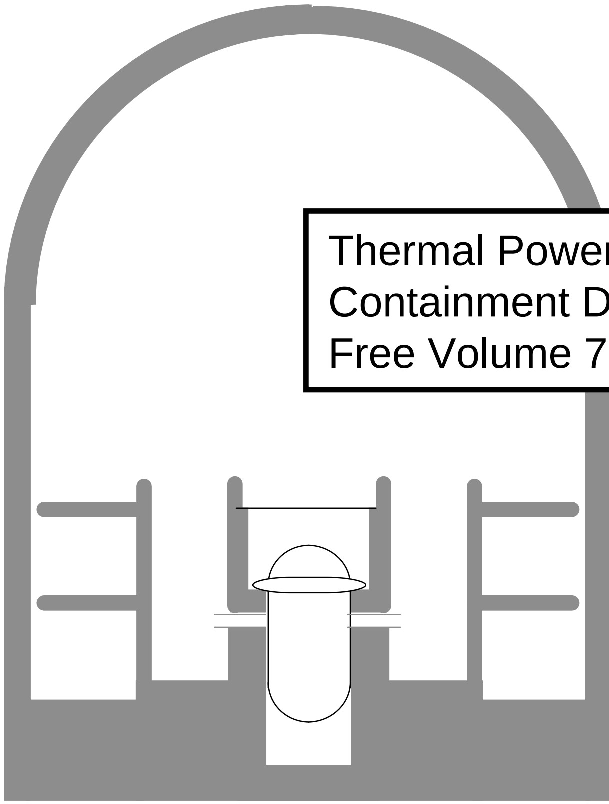
(1) With a primary objective of estimating containment performance, the level 2 PSA by uncertainty estimate was executed for a typical Japanese 1,100 MWe PWR plant.

(2) In the level 2 PSA, it is necessary to estimate the phenomenological uncertainty associated with phenomena such as steam explosions, direct containment heating, and debris cooling.

(3) The evaluation methodology of probability distributions by the ROAAM method applying experiment results for simulated severe accident phenomena.

(4) Quantification of Containment Event Trees was carried out considering the phenomenological probability distributions.

2. Outline of the Reference PWR Plant

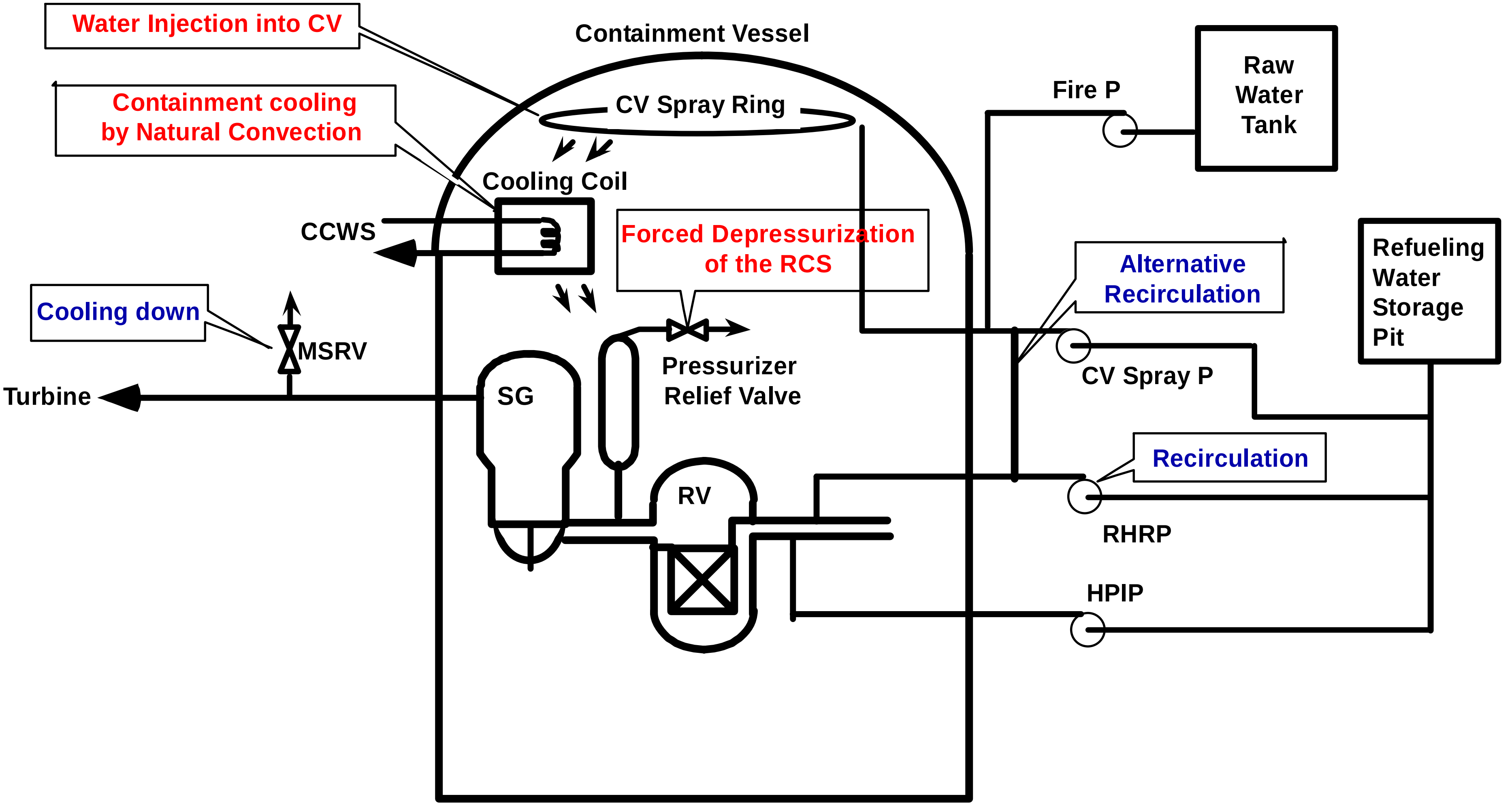


Thermal Power 3,411 MWt
Containment Design Pressure 493kPa
Free Volume 73,700m^3

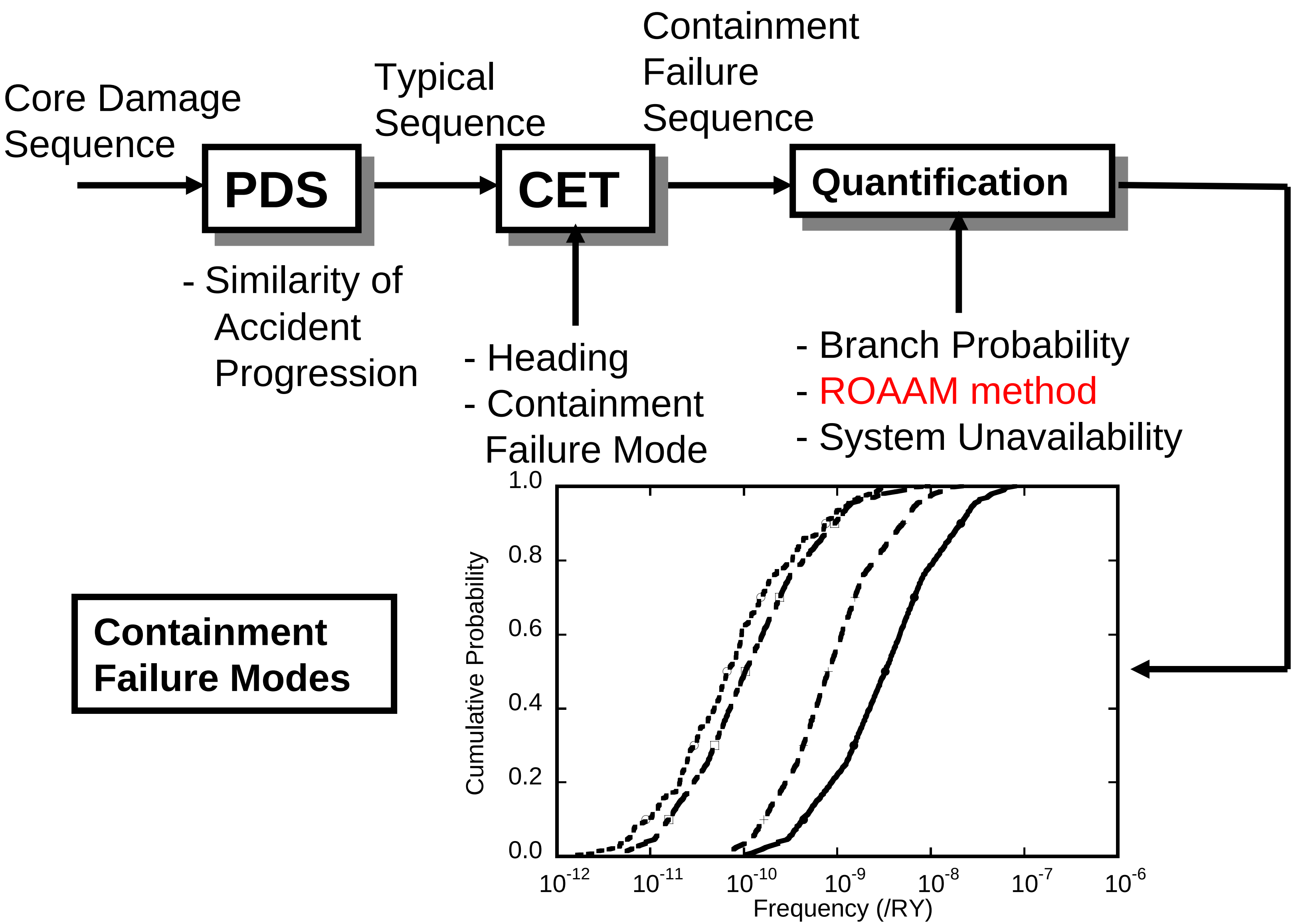
High Pressure Safety Injection System : 2 Train
Low Pressure Safety Injection System : 2 Train
Containment Spray System : 2 Train
Auxiliary feed water : 3 pumps
Re-circulation mode change of ECCS : Automatic
Component cooling water system : Train isolation with motor-operated valves

4-Loop PWR with a Pre-stressed Concrete Containment

Accident Management Counter-measures



Uncertainty Analysis Flow of Containment Failure Frequency



3. Level 1 - Level 2 Interface

PDSs of a Japanese PWR Plant

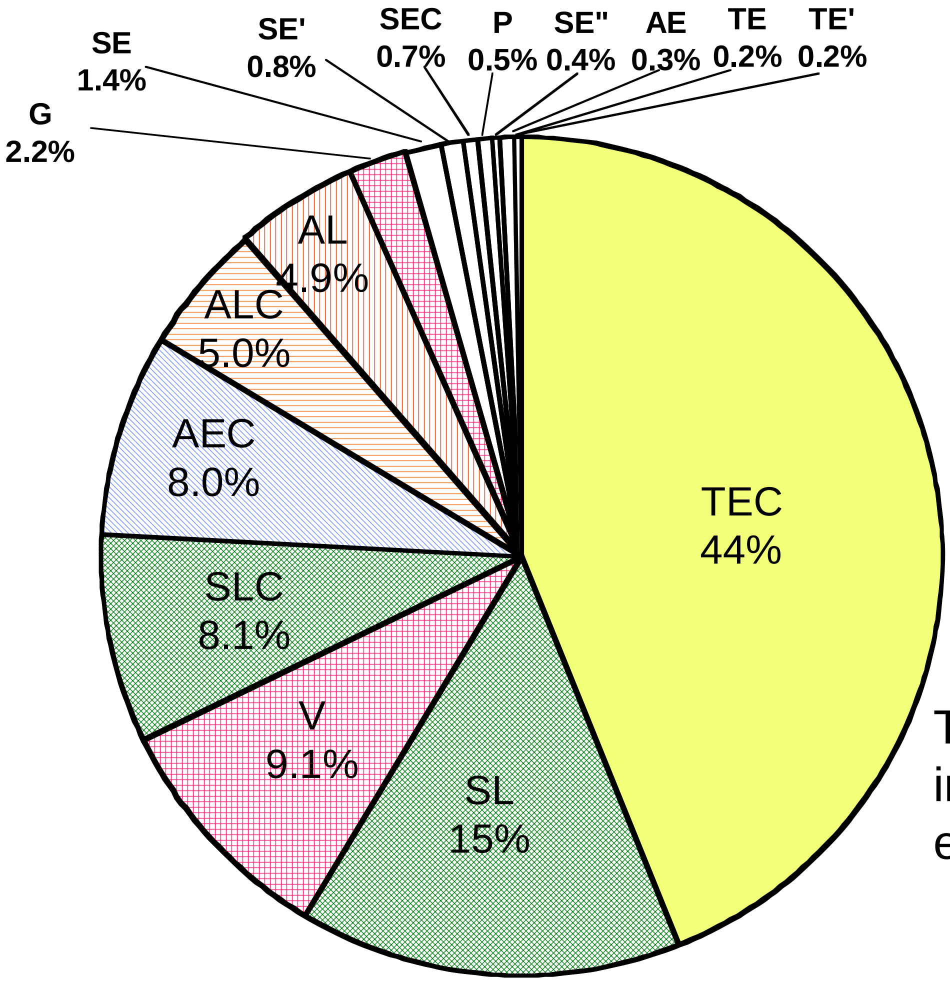
PDS	Average	5%	50%	95%
TEC	3.9E-08	8.3E-10	8.1E-09	1.4E-07
SL	1.3E-08	4.2E-10	4.0E-09	5.0E-08
V	8.0E-09	3.0E-10	3.3E-09	3.0E-08
Total	8.8E-08	2.5E-09	2.4E-08	3.2E-07

Using the WinNUPRA code, 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 double figures.

AE :Large&Medium LOCA/Early Core Damage /Without CV Spray
AEC:Large&Medium LOCA/Early Core Damage /With CV Spray
AL :Large&Medium LOCA/Late Core Damage /Without CV Spray
ALC:Large&Medium LOCA/Late Core damage /With CV Spray
SE :Small LOCA/ Early Core Damage /Without CV Spray
SE' :SBO/RC Pump Seal LOCA
SE" :CCWS Failure/RC Pump Seal LOCA
SEC:Small LOCA/Early Core Damage /With CV Spray
SL :Small LOCA/Late Core Damage /Without CV Spray
SLC:Small LOCA/Late Core Damage /With CV Spray
TE :Transient/Early Core Damage /Without CV Spray
TE' :SBO
TEC:Transient/Early Core Damage/With CV Spray
G :SGTR
P :Containment Failure before Core Damage
V :Interface-System LOCA

Core Damage Frequencies with AM (4 Loop Plant)



The contribution fraction of TEC in PDSs was the highest and estimated to be about 44%.

Average Core Damage Frequency 8.8E-08/RY

4. Construction of Containment Event Trees

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.

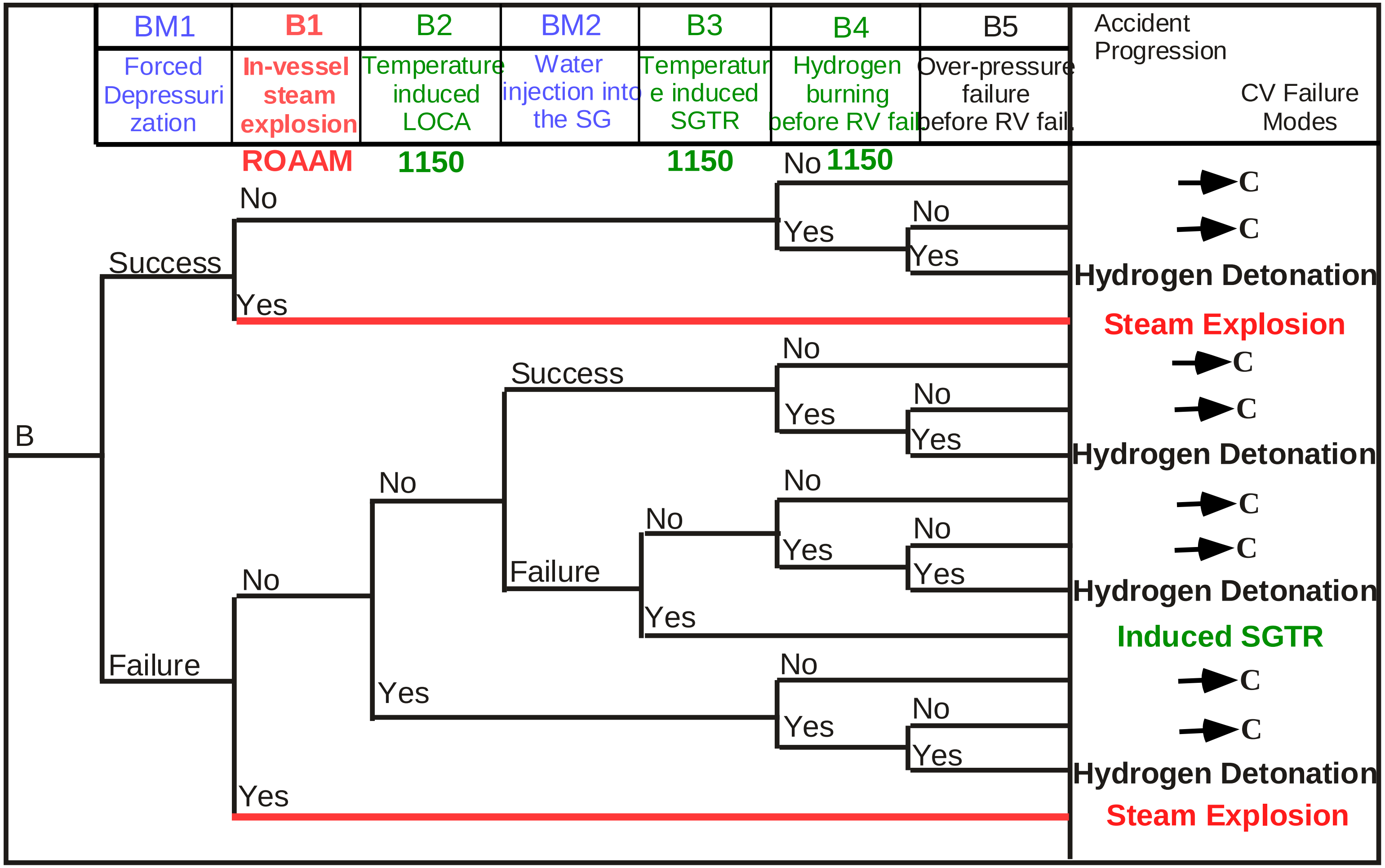
- The CET structure and nodal questions address all of the relevant issues important to severe accident progression, containment response, failure, and source terms.

CET 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.

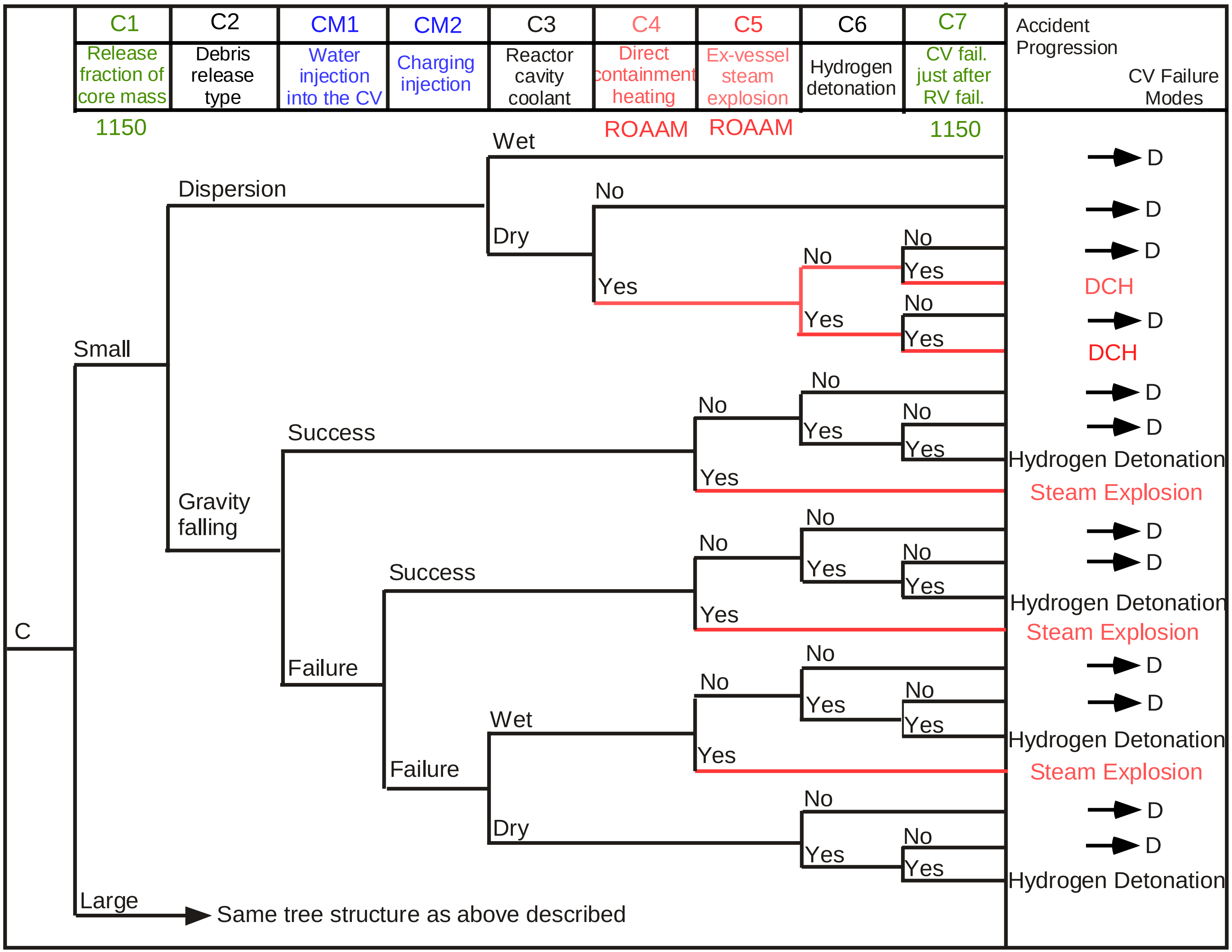
- The heading (B1, C4, C5 and D1) concern with four severe accident phenomena and are treated with the ROAAM method.

CET for a period from the initiation of core damage to the reactor vessel failure



- For headings of B2, B3, and B4, probability distributions were determined with a method similar to the Zion plant evaluation in NUREG-1150.
- The end points of CETs that were relevant to the integrity and retention capability of the containment were attributed to containment failure modes.

CET for an early phase after the reactor vessel failure



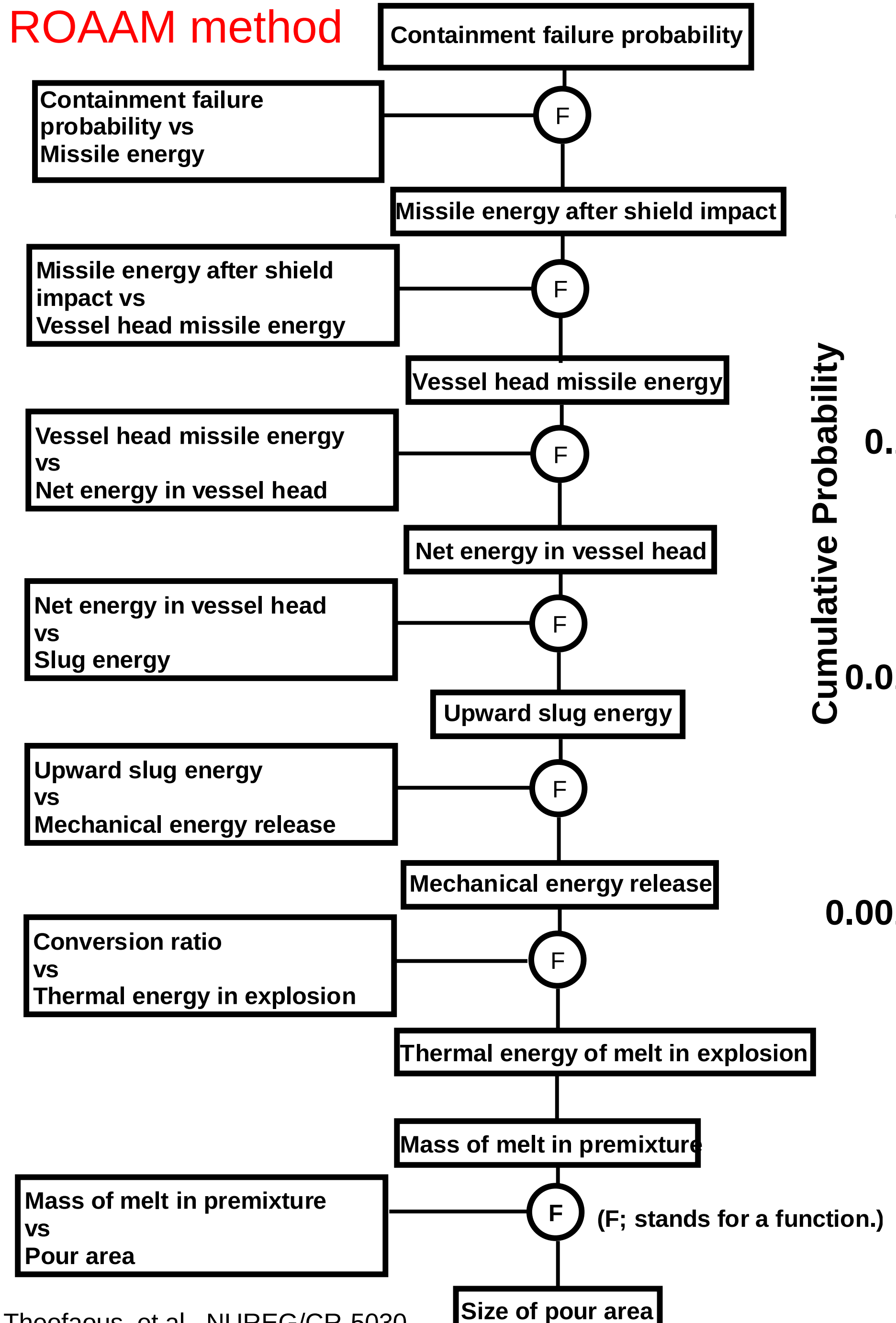
- The same split probability as the point-estimate evaluation was used for top events that did not consider the probability distribution.

5. Uncertainty Analysis of Containment Failure Frequency

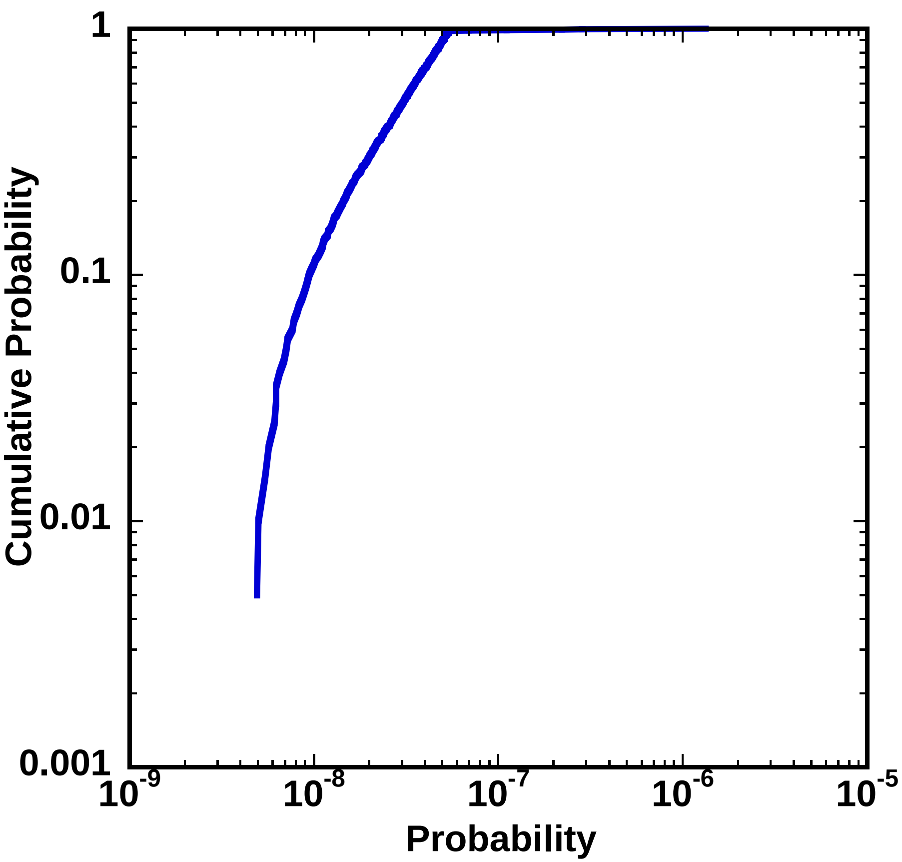
1. By using probability distributions of headings concerning mitigation AMs, and probability distributions obtained by the ROAAM method of headings, uncertainty distributions of containment failure frequency and release categories were obtained by means of the uncertainty propagation analysis of the containment event tree.

2. The probability distribution of a mode failure was calculated by 200 samplings by the PREP/SPOP code for the probability distributions of each box which constitutes the phenomenological event tree.

ROAAM method



In-Vessel Steam Explosion

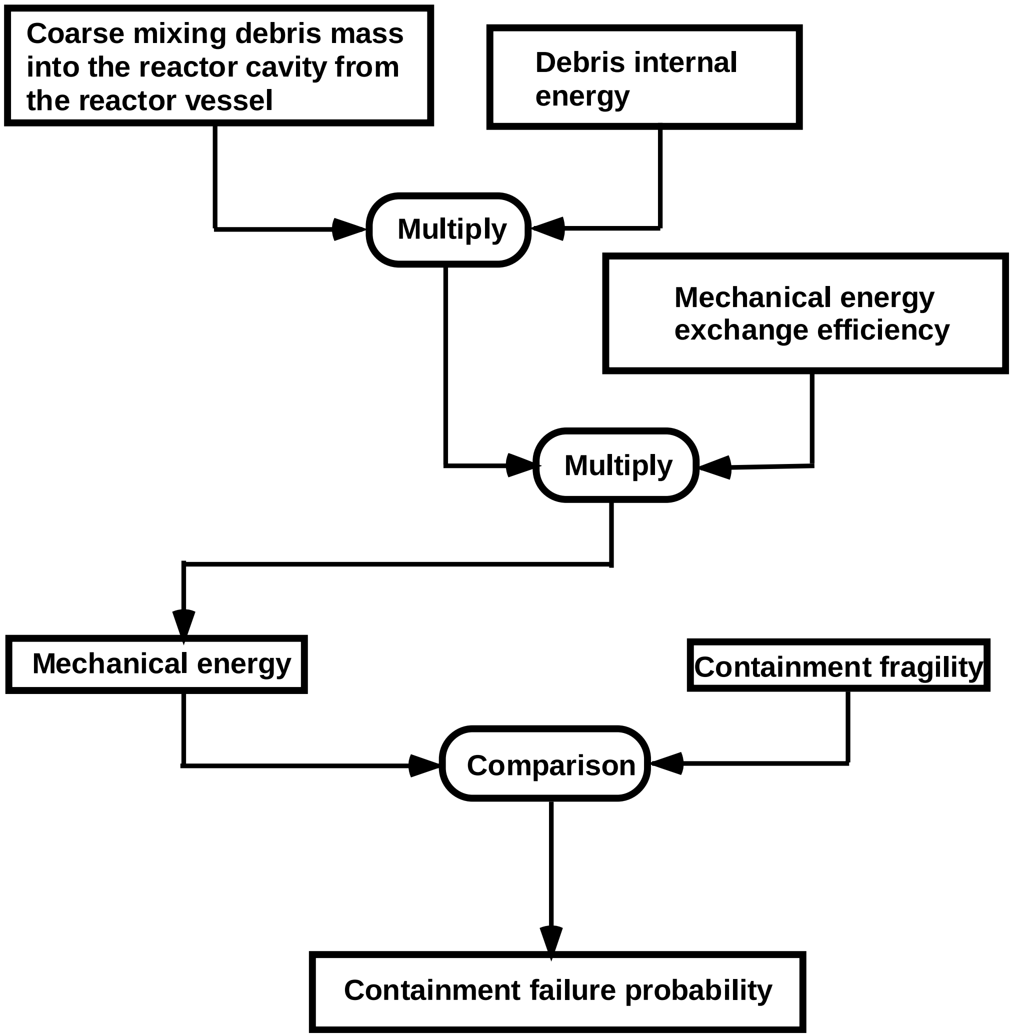


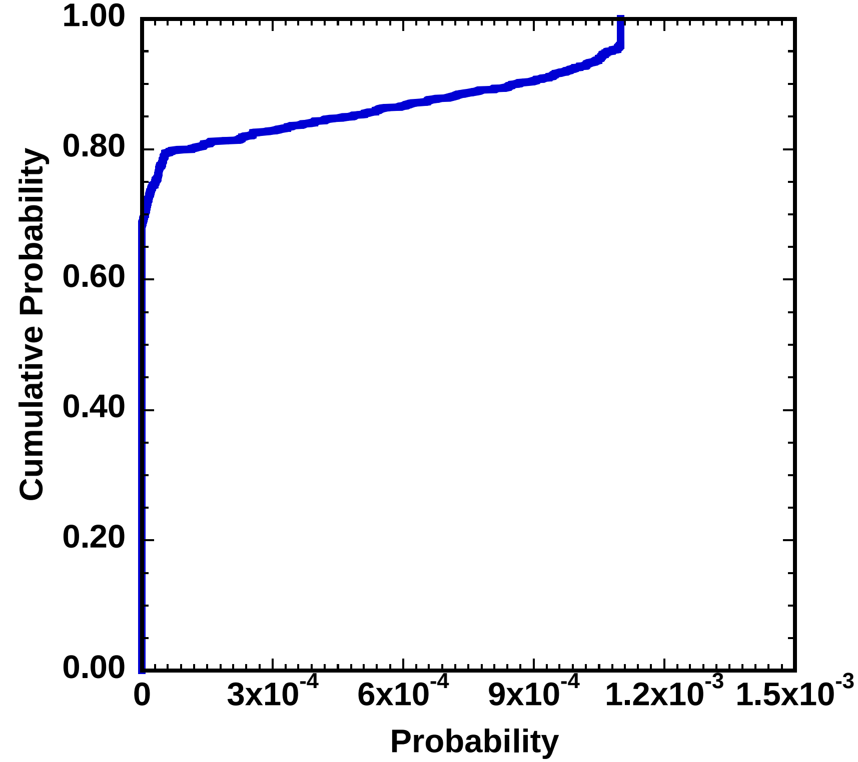
Average : 6E-06
5% value : 7E-09
50% value : 3E-08
95% value : 1E-06

T. G. Theofaous, et al., NUREG/CR-5030

Ex-Vessel Steam Explosion

ROAAM method



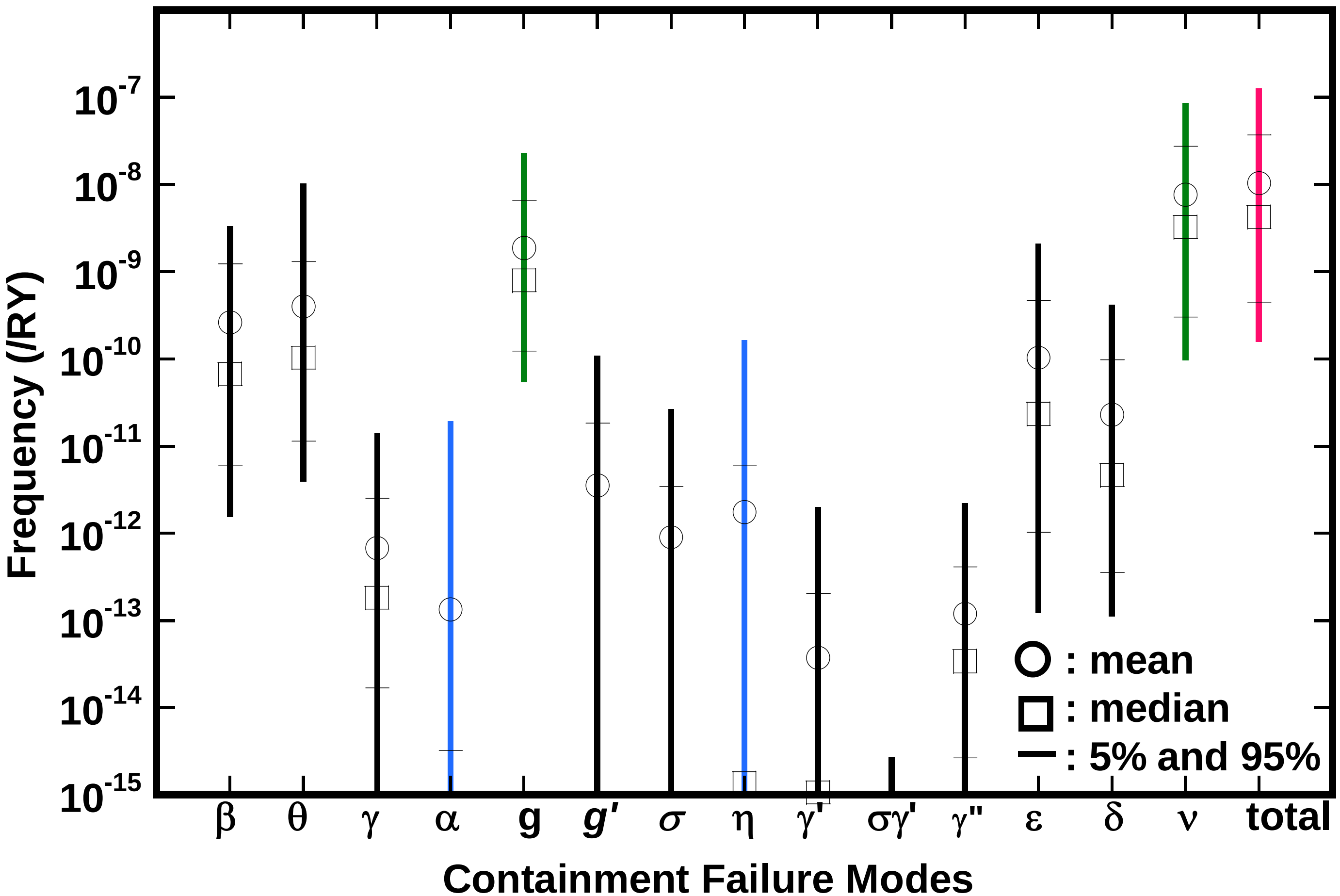


Average : 2E-04
5% value : 0.0
50% value : 0.0
95% value : 1E-03

Containment Failure Modes

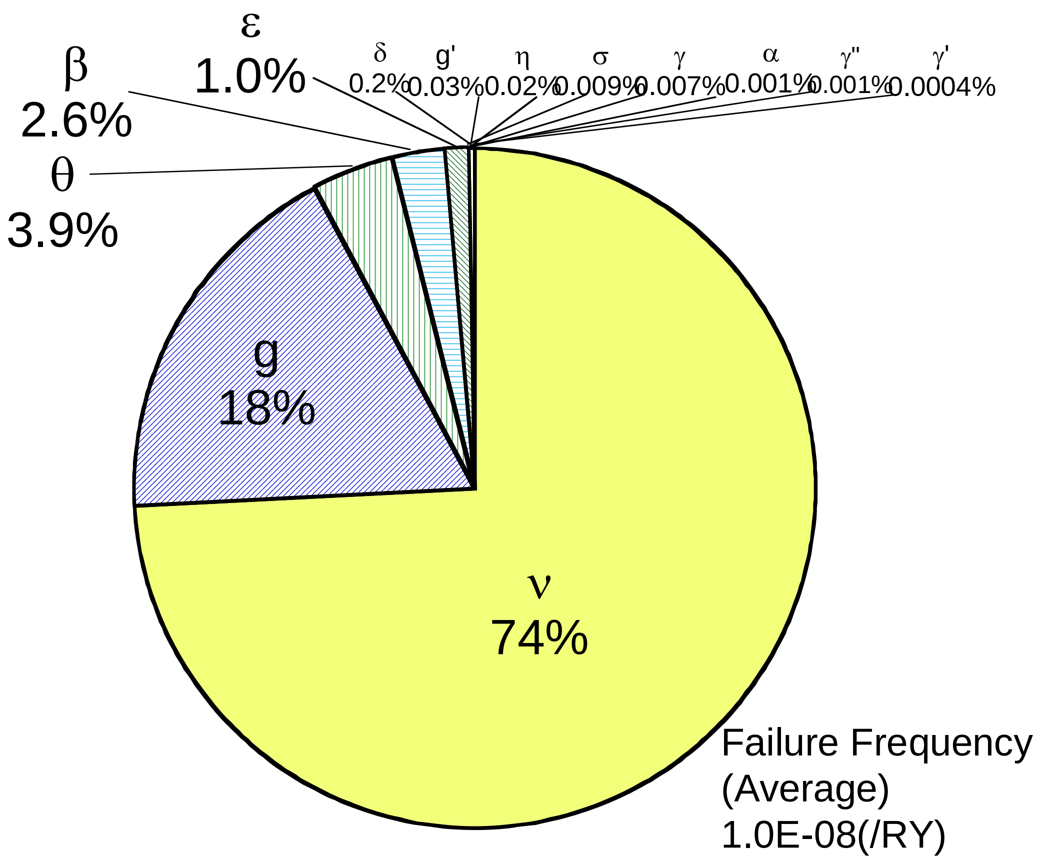
Mode	Definition
α	In-vessel steam explosion
β	Loss of containment isolation
γ	Hydrogen combustion prior to reactor vessel failure
γ ’	Hydrogen combustion at reactor vessel failure
γ “	Hydrogen combustion late after reactor vessel failure
δ	Over-pressure failure by Steam and non-condensable gases accumulation
ε	Basemat melt-through
η	Ex-vessel steam explosion
θ	Over-pressure failure prior to the core damage
σ	Direct containment heating
g	SGTR
g’	Temperature induced-SGTR
ν	Interface system LOCA

Probability distribution of containment failure frequency



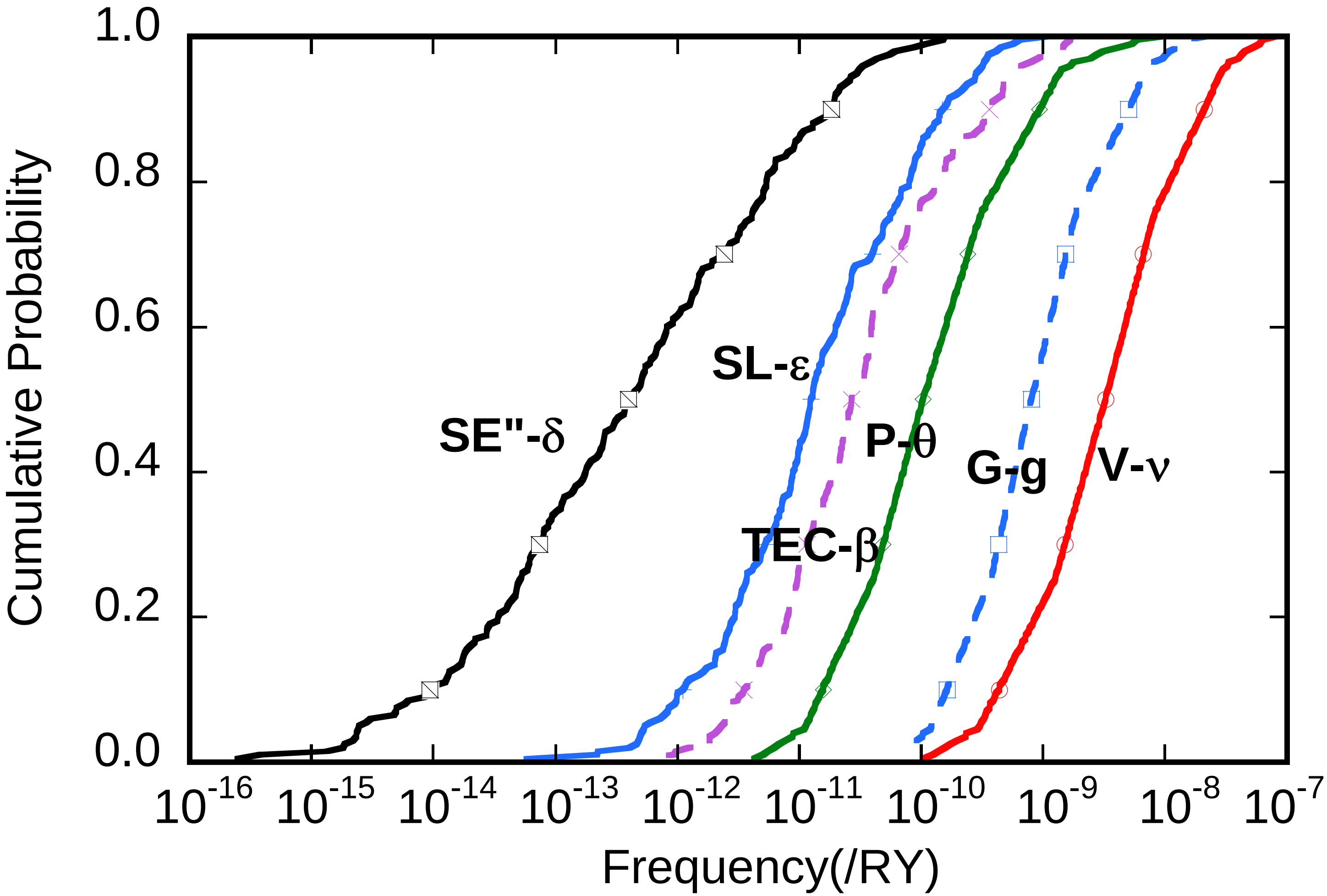
- An average value of the containment failure frequency distribution were obtained to be 1.0x10-8 /RY.
- The uncertainty bands of total containment failure frequency predicted by the ROAAM method extended three or more figures.
- The uncertainty band of containment failure frequency was in the range similar to the uncertainty band of core damage frequency.

Containment Failure Fraction with AM (4Loop Plant)



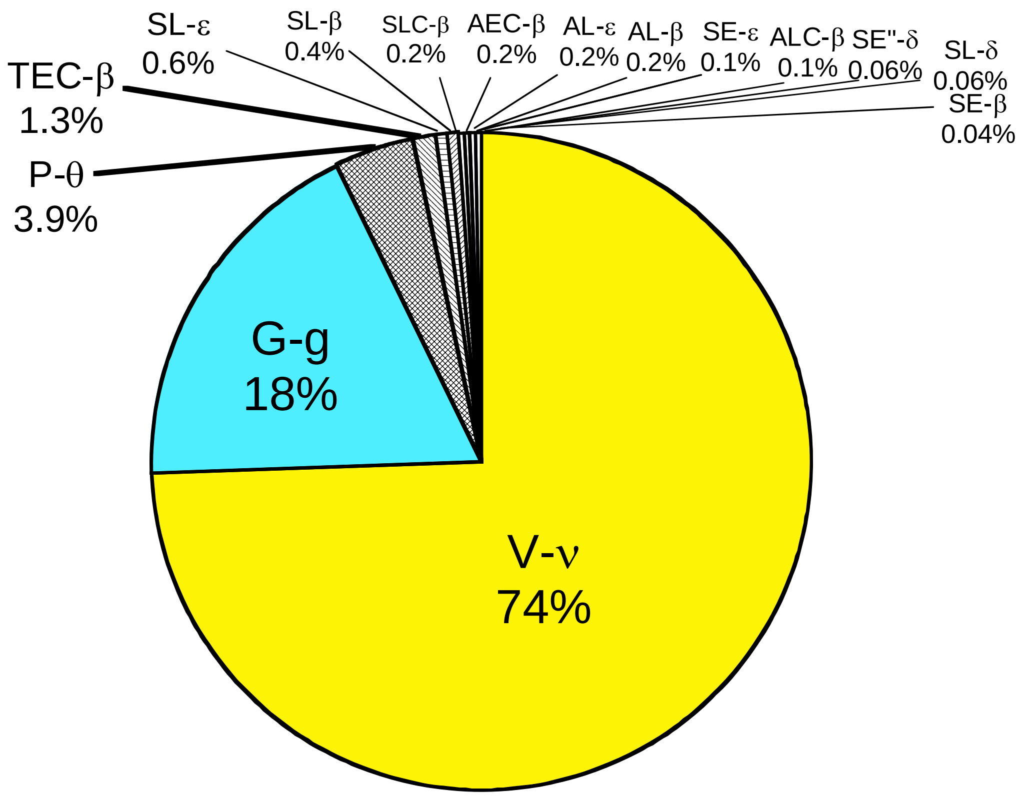
- A dominant failure mode of containment failure was a containment bypass (ν) from an interface system LOCA sequence in which a pipe of the residual heat removal system breaks by primary system pressure loading.
- The other dominant containment failure modes were SGTR (g), basemat melt-through (ε), and 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%.

Uncertainty for Release Categories with AM (4Loop Plant)



- The release categories are obtained by combining PDS with containment failure modes.
- The release category V-ν in which fission products bypass a containment boundary by an IS-LOCA node became dominant to the containment failure frequency, and the frequency of release category was estimated to be about 7.7×10^{-9}/RY (average).

Fraction for Release Categories with AM (4Loop Plant)



- The release category V-ν contributed 74% of the total frequency.
- The release category G-g of a containment bypass by SGTR contributed about 18%.
- The release category P-θ which causes core damage after containment failure became about 3.9%.
- Release categories such as TEC-β and SL-β which are associated with the containment isolation failure estimated to be about 2%.
- Release categories such as SE"-δ and SL-δ which cause a late containment failure by overpressure were about 1%.

6. Source Terms Uncertainty Analysis

Utilization of Source Term Evaluation Equations

- For source term uncertainty analysis, the source term evaluation equations in the XSOR code which were provided for the Zion plant in NUREG-1150 were applied.
- The source term equations in the XSOR consist of source term parameters.

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
FCCI(i)	Fractional release of nuclide group i from corium during molten core-concrete interactions
FCONC(i)	Containment transport fraction for ex-vessel release
FLATE(i)	Fractional releases of material deposited in RCS due to revaporization rate

Uncertainty Propagation Analysis for each Release Category (1)

- Uncertainty probability distributions of thirteen source term parameters were obtained by carrying out 200 sampling calculation by the PREP/SPOP code.
- In this calculation, the conditions of a source term parameter for release categories were chosen.

Release Category	FCOR	FISG	FOSG	FVES	FCONV	DEF	FCCI	FCONC	FLATE
V-ν	Low Zr Oxidation	No	No	IS-LOCA	IS-LOCA	No	Low Zr Oxidation, No Water	IS-LOCA	Two Holes in RCS

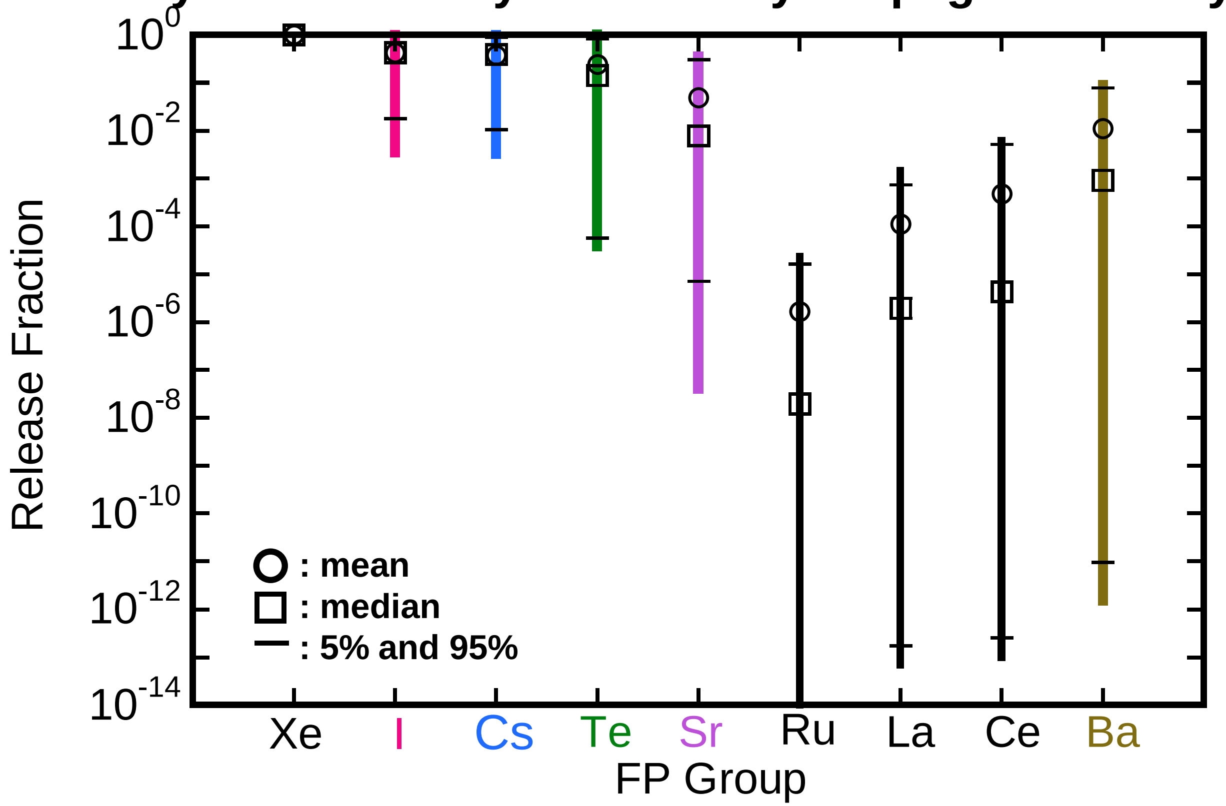
- The release fraction (average) to environment for each release category was calculated by the uncertainty propagation analysis.

Release Category	FP Group								
	Xe	I	Cs	Te	Sr	Rn	La	Ce	Ba
V-ν	1.0E-00	4.2E-01	3.8E-01	2.4E-01	4.8E-02	1.7E-06	1.1E-04	4.8E-04	1.1E-02

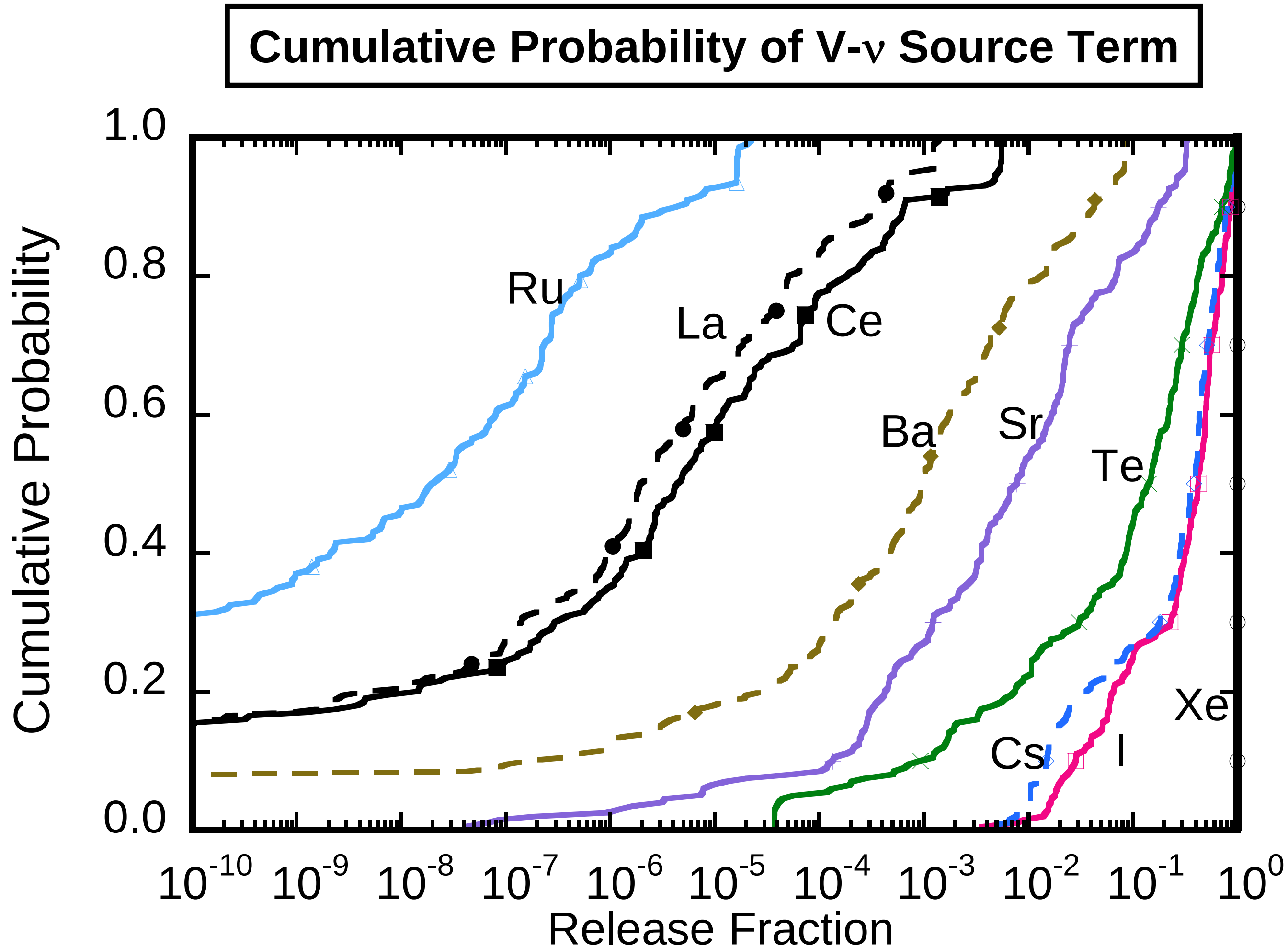
Uncertainty Propagation Analysis for each Release Category (2)

- 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.

Analysis Results by Uncertainty Propagation Analysis



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



- The cumulative probability curves of Ce, La, and Rn groups have shifted to lower values because of the compensation by the MELCOR calculation coupled with the original NUREG-1150 method.

7. Conclusion

Containment Failure Frequency Evaluation

(1) 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}.

(2) The calculated result showed that the average value of total containment failure frequency was obtained to be 1.0×10^{-8} /RY.

(3) The containment failure frequency has an uncertainty width of double figures similar to the uncertainty width of core damage frequency.

Source Terms Analysis

(1) A source term uncertainty analysis has been performed by typical release categories based NUREG-1150 methodology.

(2) 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.