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Probabilistic Safety Assessment (PSA) of Nuclear Power Plants

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DOCUMENTATION FOR THE LECTURE ON PSA OF NPP

1. Slides (M. Mazzini)
2. Chapter 6 of LECTURES OF NUCLEAR SAFETY, 2014 (M. Mazzini)
3. INSAG-6: PROBABILISTIC SAFETY ASSESSMENT, IAEA, Vienna, 1992
4. Procedures for Conducting Probabilistic Safety Assessment of NPP (Level 1). IAEA Safety Series N. 50-P-4, IAEA, Vienna, 1992
5. Procedures for Conducting Probabilistic Safety Assessment of NPP (Level 2). IAEA Safety Series N. 50-P-8, IAEA, Vienna, 1995

CONTENT

- **Introduction**
- **PSA Levels**
- **Level 1 PSA Objectives and Use**
- **Level 2 PSA Objectives and Use**
- **Examples of PSA Results**

INTRODUCTION

The Rasmussen Report had a training role in the development of this fundamental part of the nuclear safety analysis.

At present this is a structured discipline, as shown by the literature cited as documentation for these lectures.

DEFINITION OF PSA LEVELS

- **LEVEL 1 PSA: Evaluation of the Core Damage Frequency (CDF)**
- **LEVEL 2 PSA: Evaluation of the Spectrum of Releases in Function of Cumulative Complementary Distribution Frequency (CCDF)**
- **LEVEL 3 PSA: Evaluation of the Spectrum of Consequences on Population and of Economic Damages in Function of CCDF**

Types of PSA

PSA can be done a posteriori or a priori

- The first type, **termed a posteriori analysis**, refers to existing plants with operating histories. It is based on information from past operating experience, normally plant specific data. Generic data are used where plant specific data are lacking.
- The second type, **termed a priori analysis**, relates to a plant with no operating history. Here generic databases or models provide the basic information for the probabilistic study.

LEVEL 1 PSA OBJECTIVE AND USE

***Objective:* Evaluation of the Plant Safety Level**

Use of PSA:

- **Compliance with Safety Goals and Standards**
- **Identification of Effective Areas of Improvement**
- **Assistance in Plant Operation**

PSA OBJECTIVES AND USES

Effective Areas of Improvement:

- **Dominant accident sequences**
- **Critical systems, components and operator actions**
- **Evaluation of new safety issues**
- **Design modifications and decisions for back-fitting**

PSA OBJECTIVES AND USES

Assistance to Plant Operation:

- Optimization of Technical Specification and Operating Procedures
- Safety related maintenance
- Evaluation of operating experience (living PSA)
- Accident management (training and procedures)

LEVEL 1 PSA MAIN PHASES

- **Identification of the Sources of Radioactive Releases and Accident Initiators**

- **Modelling of the Accident Sequences (including Data Assessment and Parameters Estimation)**

- **Quantification of the Accident Sequences**

Radioactive Release Sources and Accident Initiators

- Determination of plant operating states
- Definition of core damage states
- Selection and grouping of initiating events
- Definition of safety functions and related safety systems

Modelling of the Accident Sequences

Construction of the Event Trees for all groups of initiating events.

On the basis of the frequency of each initiating event and of unavailability of pertinent protection and safety systems, one evaluates the probability of occurrence of the various accident sequences

(iterative application of a cut-off $10^{-10} – 10^{-12}$ y^{-1}).

The sum of all the accident frequencies gives the CDF.

SIZEWELL B LEVEL 1 PSA RESULTS

INITIAL EVENT	FREQUENCY (PER R.Y.) OF CORE MELTING	PERCENT WEIGHT
LARGE L.O.C.A.	0.182E-06	15.47
MEDIUM L.O.C.A.	0.250E-06	21.94
SMALL L.O.C.A.	0.390E-06	33.10
STEAM GENERATOR TUBE RUPTURE	0.191E-07	1.60
SECONDARY SIDE BREAK INSIDE CONTAINMENT	0.232E-07	1.97
SECONDARY SIDE BREAK OUTSIDE CONTAINMENT	0.353E-07	3.00
LOSS OF MAIN FEEDWATER FLOW	0.157E-07	1.33
CLOSURE OF ONE MAIN STEAM ISOLATION VALVE	0.571E-10	<0.01
LOSS OF REACTOR COOLANT SYSTEM FLOW	0.811E-10	<0.01
CORE POWER EXCURSION	0.511E-11	<0.01
TURBINE TRIP	0.836E-09	0.07
SPURIOUS SAFETY INJECTION	0.144E-09	0.01
REACTOR TRIP	0.854E-09	≈0.07
ANTCIPATED TRANSIENT WITHOUT SCRAM	0.141E-06	11.99
LOSS OF OFFSITE POWER / TURBINE TRIP	0.767E-08	0.65
INTERFACING SYSTEM L.O.C.A.	0.237E-08	0.20
L.O.C.A. BEYOND CAPACITY OF E.C.C.S.	0.100E-06	8.50
TOTAL	0.117E-05	≈100.00

Quantification of the Accident Sequences

Determination of the frequency of each type of plant damage state.

Several degree of core damage might be considered, on the basis of the resulting release from the core.

One can also perform a selection of core damage states on the basis of their frequency of occurrence or grouping some on the basis of similarity in release characteristics.

LEVEL 2 PSA OBJECTIVES AND USES (1/2)

- ***Insights into SA progression and containment performance***
- ***Identification of plant specific vulnerabilities***
- ***Quantification of radioactivity releases to the environment***

LEVEL 2 PSA OBJECTIVES AND USES (2/2)

- ***Demonstration of compliance with quantitative Safety Standards***
- ***Basis for off-site emergency planning***
- ***Basis for development of plant specific AM strategies***
- ***Basis for Level 3 PSA***

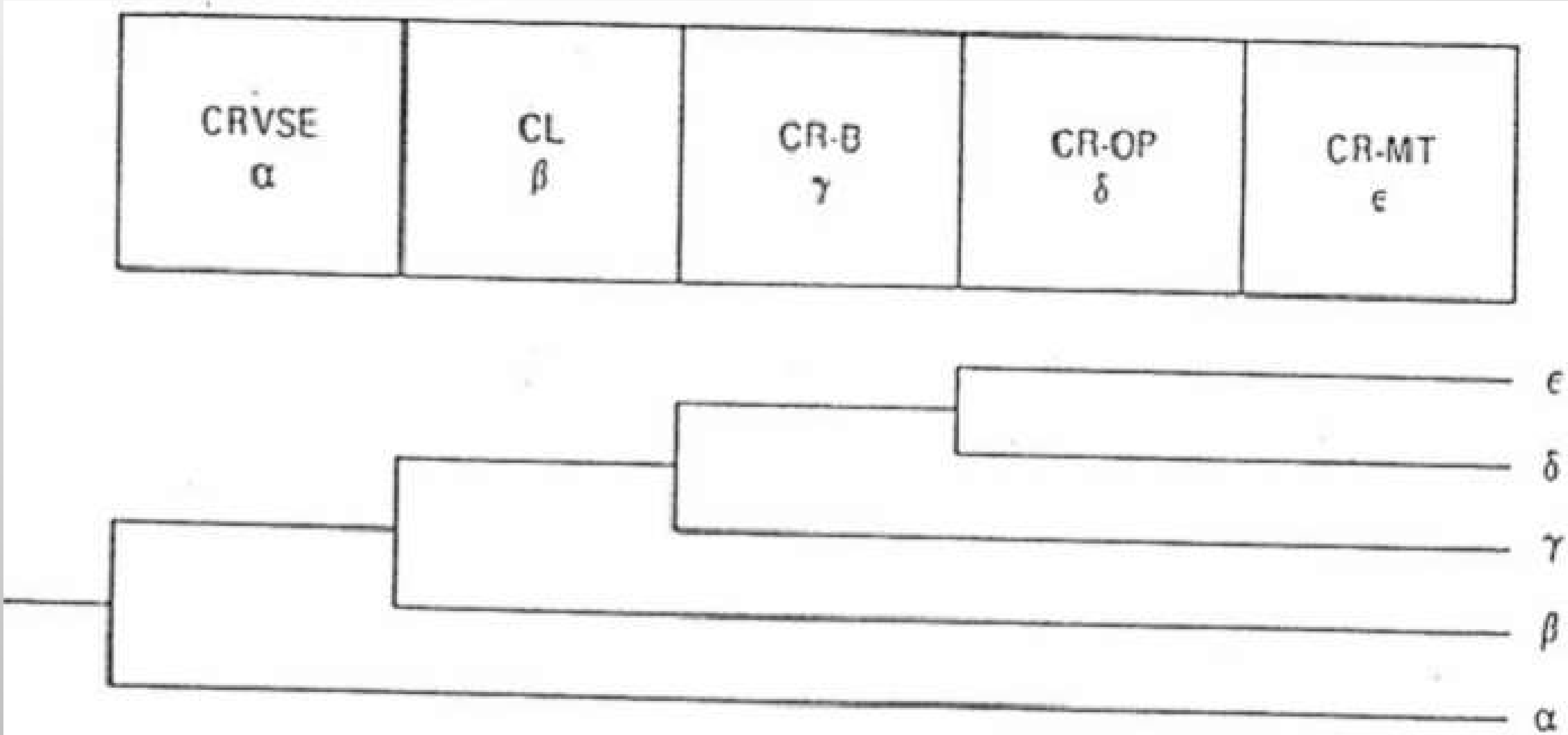
LEVEL 2 PSA MAIN PHASES

- **Accident Progression and Containment Response Analysis**
- **Quantification of the Source Term Due to Severe Accidents**

Accident Progression and Containment Analysis

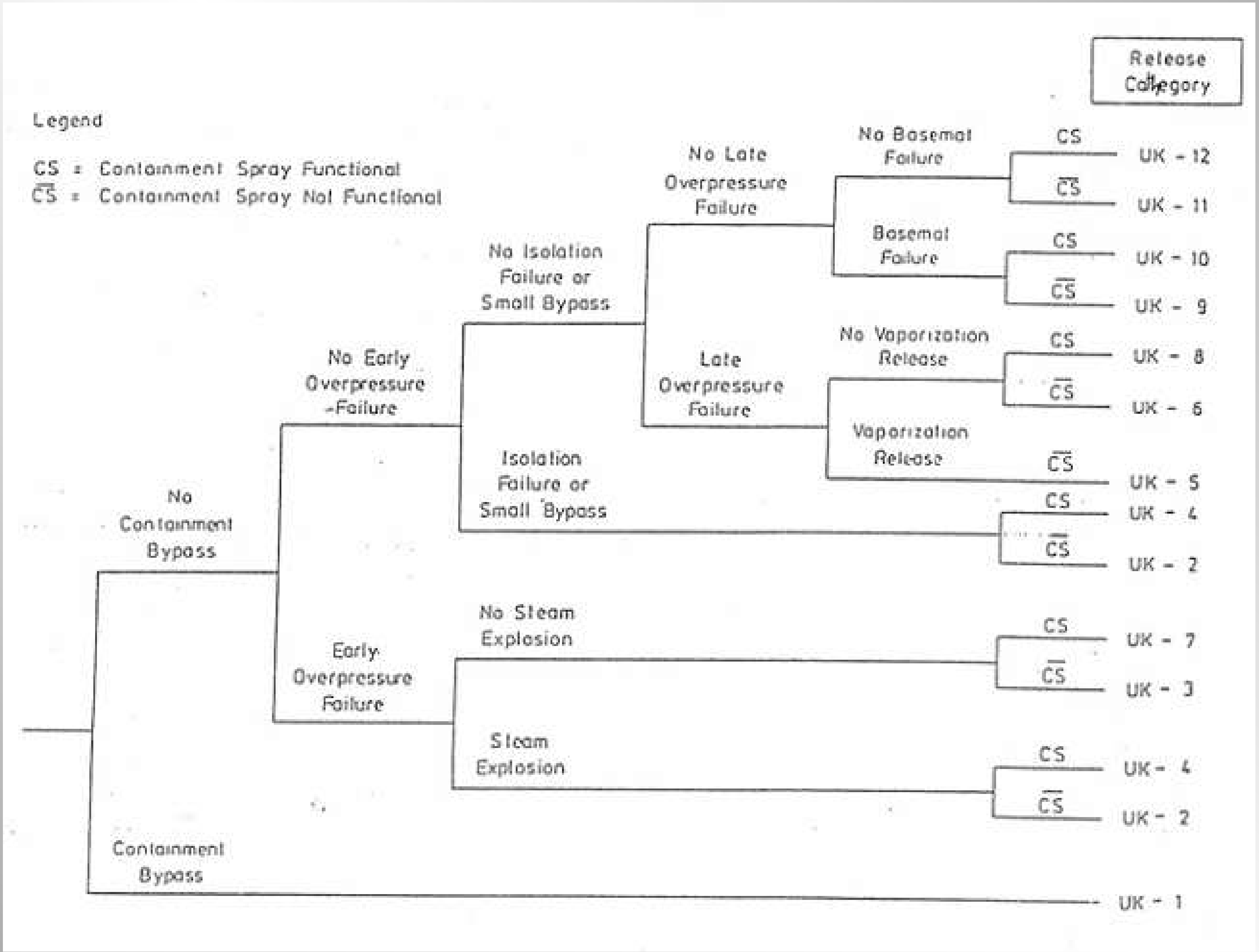
- **Analysis of SA progression and containment performance**
- **Development of containment ET**
- **Binning of ET end-states in release categories**
- **Treatment of uncertainties in accident progression**

CONTAINMENT EVENT TREE OF WASH 1400



CRVSE	Containment failure from in-vessel steam explosion
CL	Containment isolation failure
CR-B	Containment failure from hydrogen combustion
CR-OP	Containment failure from overpressurization
CR-MT	Containment failure through basemat penetration

SIZEWELL PWR RELEASE CATEGORIES



SIZEWELL B RELEASE CATEGORIES

Release Cat and Frequency	Release Characteristics					Release Fractions							
	Time Release Starts (hrs)	Duration (hrs)	Warning time (hrs)	Energy(1) MBTu/hr	Height (m)	Xe-Kr	Org I	I	Cs-Rb	Te	Ba-Sr	Ru	La
UK-1 2.4 (-9)	1	3	0	0.3	10	9(-1)	7(-3)	7(-1)	5(-1)	3(-1)	6(-2)	2(-2)	4(-3)
UK-2 4.0 (-10)	1	0.5	0	20	10	9(-1)	6(-3)	7(-1)	4(-1)	3.5(-1)	5(-2)	2(-1)	3(-3)
UK-3 2.4 (-9)	1	0.5	0	20	10	8(-1)	6(-3)	6(-1)	6(-1)	1(-1)	8(-2)	2(-2)	2(-3)
UK-4 5.9 (-10)	2	1	1	6	10	8(-1)	6(-3)	2(-1)	2(-1)	2.5(-1)	2(-2)	1.5(-2)	3(-3)
UK-5 8.0 (-9)	8	0.5	4	20	10	1 (0)	7(-3)	6(-2)	3(-1)	5(-1)	4(-2)	3(-2)	6(-3)
UK-6 4.2 (-9)	12	0.5	8	20	10	9(-1)	6(-3)	9(-3)	2(-1)	4(-2)	2(-2)	7(-3)	7(-4)
UK-7 1.2 (-9)	1	0.5	0	20	10	8(-1)	6(-3)	8(-3)	2(-3)	4(-4)	3(-4)	8(-5)	8(-6)
UK-8 2.0 (-10)	5	0.5	4	0.3	10	8(-1)	6(-3)	6(-3)	5(-6)	1(-6)	7(-7)	2(-7)	2(-8)
UK-9 5.2 (-9)	2	10	1	0	0	3(-1)	2(-3)	8(-4)	8(-4)	1(-3)	9(-5)	7(-5)	1(-5)
UK-10 4.2 (-9)	2	10	1	0	0	6(-3)	2(-5)	2(-5)	1(-5)	2(-5)	1(-6)	1(-6)	2(-7)
UK-11 6.2 (-7)	2	long	1	0	10	6(-2)	3(-5)	3(-5)	3(-5)	3(-5)	3(-6)	2(-6)	4(-7)
UK-12 5.1 (-7)	2	long	1	0	10	5(-2)	3(-5)	8(-6)	1(-6)	2(-7)	1(-7)	4(-8)	4(-9)

Note 1: 1 Btu/hr = 0.29 Joules/sec = 0.29 watts

Note that numbers in ( ) refer to powers of 10. Thus 3(-1) represents 3×10^{-1}, which is 30 per cent

INITIAL EVENTS WEIGHTS ON PSA RESULTS

INITIAL EVENTS	RELATIVE WEIGHT											
	UK-1	UK-2	UK-3	UK-4	UK-5	UK-6	UK-7	UK-8	UK-9	UK-10	UK-11	UK-12
LARGE L.O.C.A.	0.00	21.05	9.10	1.93	0.40	1.07	2.70	6.32	35.60	5.97	25.21	5.01
MEDIUM L.O.C.A	0.00	21.05	9.10	8.99	0.40	1.07	12.94	29.39	35.60	27.75	25.21	23.33
SMALL L.O.C.A	0.00	39.63	45.40	8.09	20.99	21.51	20.76	19.00	24.24	21.10	45.53	22.76
S.G.T.R.	0.00	10.30	7.17	62.54	17.40	17.19	5.95	0.77	0.07	0.06	0.30	3.53
S.S.B.I.C.	0.00	1.34	1.27	1.25	0.02	0.05	1.05	4.05	1.71	3.03	1.54	3.23
S.S.B.O.C.	0.00	0.30	2.92	3.12	0.00	0.01	4.72	10.07	0.71	9.54	0.35	8.09
LOSS OF M.F.W.	0.00	2.36	5.02	0.39	11.20	11.12	1.25	0.05	1.86	0.92	1.71	0.99
CLOSURE OF ONE M.S.I.V.	0.00	0.02	0.12	0.00	0.31	0.30	0.00	0.00	0.00	0.00	0.00	0.00
LOSS OF R.C.S.F.	0.00	0.04	0.20	0.00	0.50	0.49	0.00	0.00	0.00	0.00	0.00	0.00
CORE POWER EXCURSION	0.00	0.00	0.02	0.00	0.03	0.04	0.00	0.00	0.00	0.00	0.00	0.00
TURBINE TRIP	0.00	0.42	2.02	0.00	5.12	5.03	0.01	0.01	0.01	0.01	0.02	0.01
SPURIOUS SAFETY INJECTION	0.00	0.07	0.03	0.03	0.08	0.08	0.00	0.00	0.03	0.00	0.00	0.00
REACTOR TRIP	0.00	0.42	2.04	0.00	5.17	5.08	0.01	0.01	0.00	0.01	0.01	0.01
A.T.W.S.	0.00	0.00	0.00	12.86	0.00	0.00	41.65	20.60	0.03	30.67	0.00	32.98
LOOP / TT	0.00	3.00	14.55	0.03	37.42	36.10	0.09	0.05	0.04	0.06	0.12	0.06
INTERFACING SYSTEM L.O.C.A.	100.0	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.03	0.00	0.00	0.00

IMPORTANCE OF SAFETY SYSTEMS FOR RELEASE CATEGORY UK-2

SYSTEMS	(%)
ELECTRIC POWER (OFFSITE + ONSITE)	21,4
COMPONENT COOLING WATER SYSTEM	21,1
CONTAINMENT SPRAYS SYSTEM	5,7
CONTAINMENT FAN COOLERS	17,8
EMERGENCY CORE COOLING SYSTEM (LP-1) (*)	1,6
EMERGENCY CORE COOLING SYSTEM (LP-2) (*)	1,2
EMERGENCY CORE COOLING SYSTEM (HH-1A) (*)	1,7
EMERGENCY CORE COOLING SYSTEM (HH-1B) (*)	1,0
EMERGENCY CORE COOLING SYSTEM (HH-2) (*)	0.4
RECIRCULATION + RESIDUAL HEAT REMOVAL SYSTEM	14,7
RECIRCULATION	1,0
AUXILIARY FEEDWATER SYSTEM	10,8
OPERATOR ACTION (BLEED AND FEED)	1,6
MAIN STEAM ISOLATION VALVES	0.0

(*) different ways of ECCS systems operation

Treatment of uncertainties in accident progression

Uncertainties are due to:

- No completeness of considered accident scenarios
- Modeling inadequacy (incomplete knowledge of phenomena, model simplifications, etc.)
- Uncertainties in input parameters (lack of data, uncertainties from level 1 PSA results, etc.)

Treatment of uncertainties in accident progression

Approach to uncertainty analysis:

Evaluation of each uncertainty issue:

– Selection of issues by sensitivity analysis or expert judgment

– Evaluation of uncertainties, using discrete probability distributions or direct simulation methods (Monte Carlo methods)

Display and interpretation of results

HISTOGRAM OF UK-2 RELEASE CATEGORY

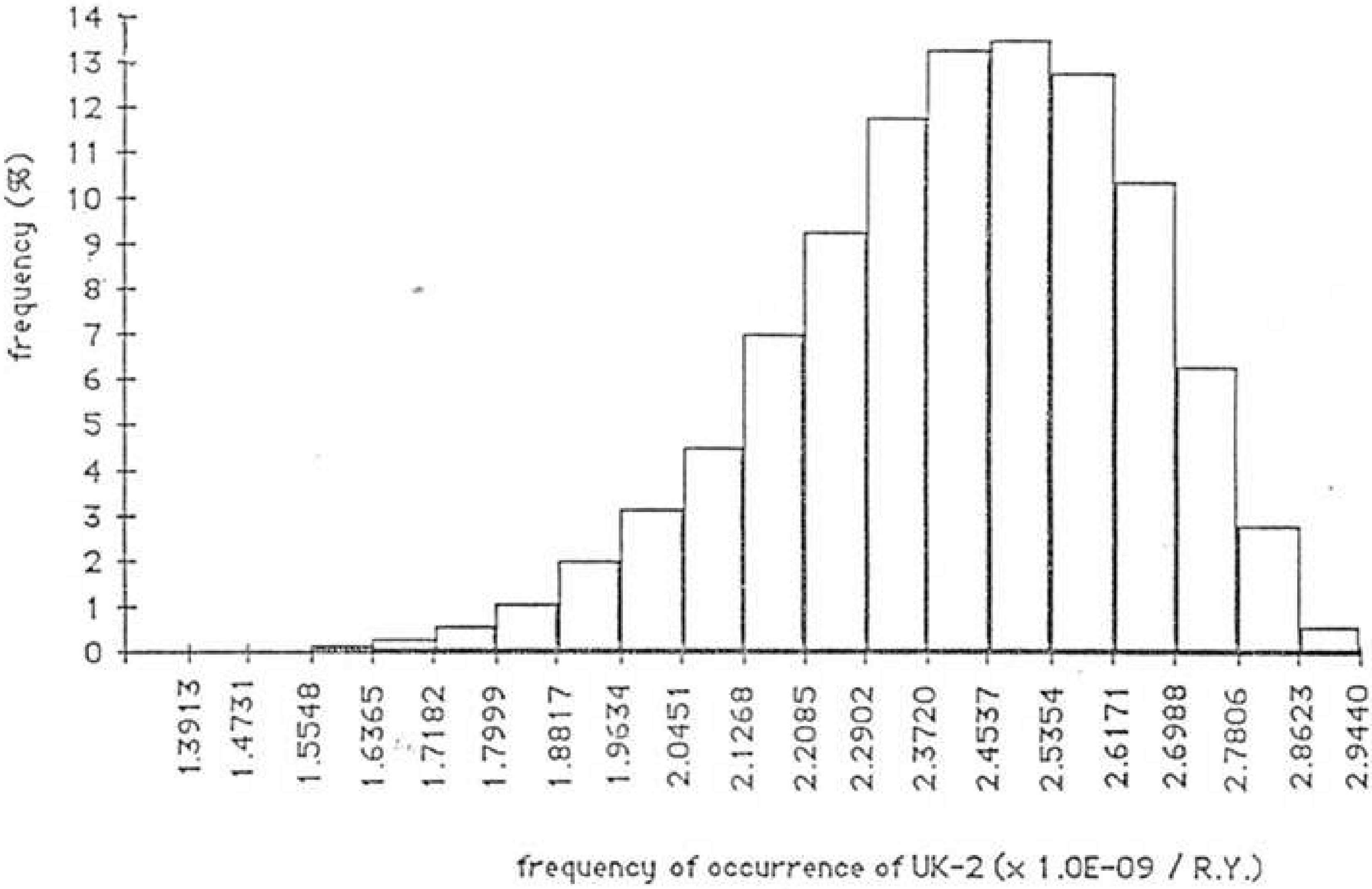


Fig. 17 - Histogram of UK-2 Release Cathegory Obtained by Montecarlo Simulation

HISTOGRAM OF UK-4 RELEASE CATEGORY

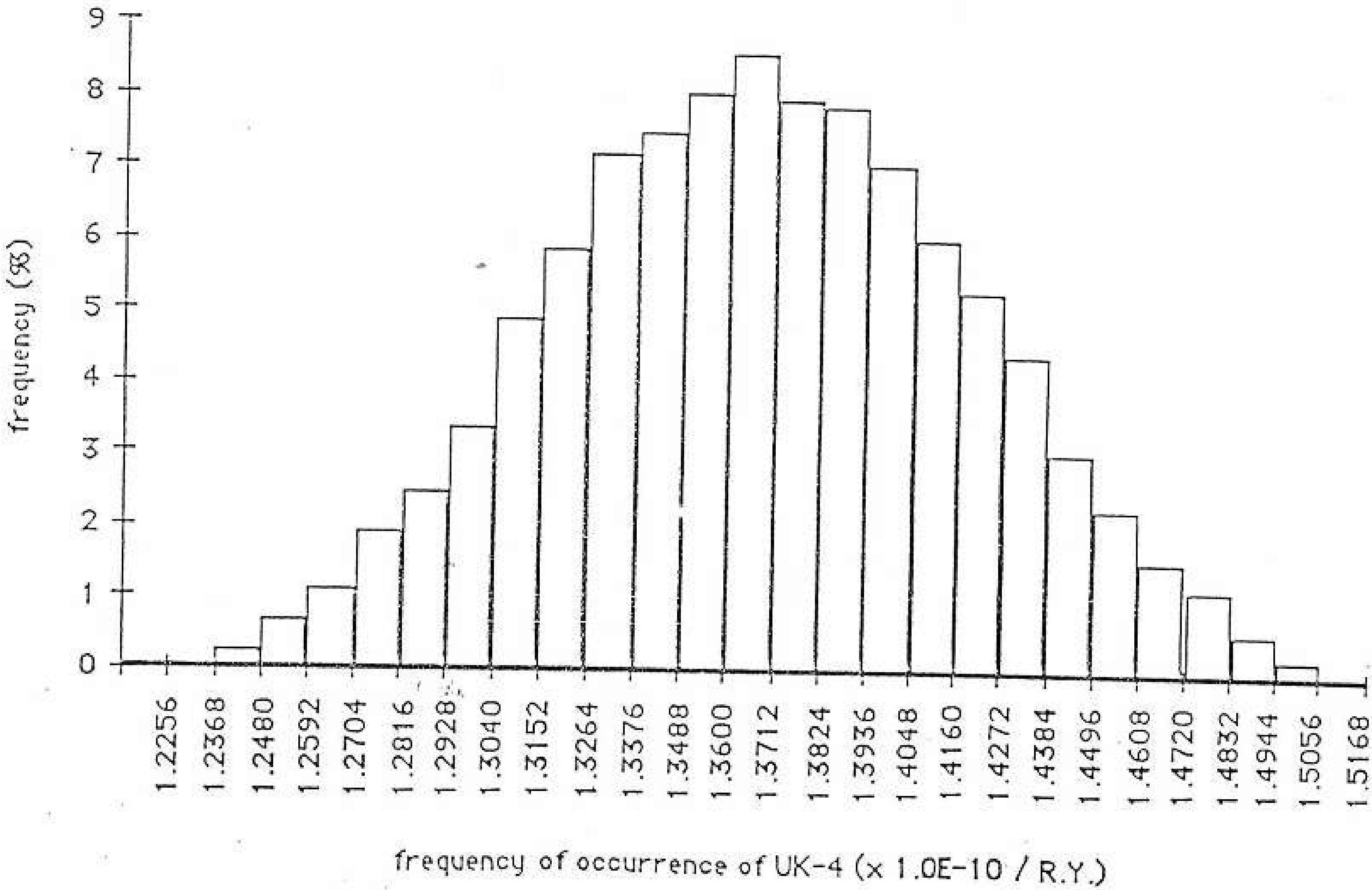


Fig. 18 - Histogram of UK-4 Release Cathegory Obtained by Montecarlo Simulation

Assessment of PSA Results

- Biases and uncertainties in results increase substantially from Level 1 to Level 3 assessments.
- By definition, relative conclusions are more reliable than absolute statements.
- The associated range of uncertainty shall be quoted in conjunction with any absolute statement (the range of uncertainties can be so wide as to require caution in the application of the results).

Uncertainties in PSA Results

- Presently available Level 1 PSA of proven reactor designs, the uncertainties in CDF cover a range of roughly one order of magnitude.
- In Level 2 analyses, the uncertainties are much larger, because of the difficulties in modelling containment failure mechanisms associated with severe accidents. Uncertainty range of Level 2 results can extend over several orders of magnitude.

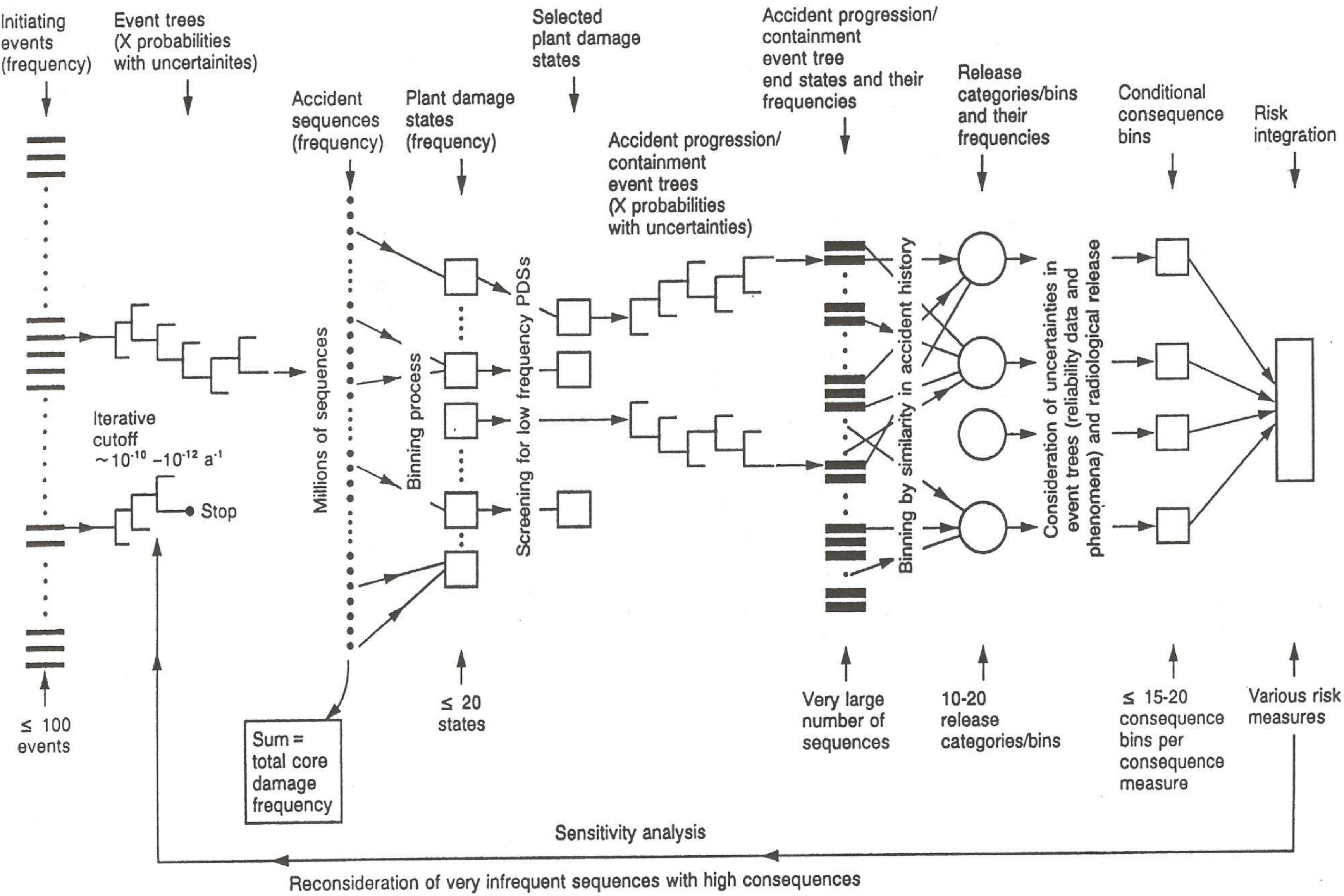


FIG. 1. Method for probabilistic safety assessment.