Unclassified **NEA/CSNI/R(97)11**

Organisation de Coopération et de Développement Economiques
Organisation for Economic Co-operation and Development

OLIS : 14-Nov-1997
Dist. : 17-Nov-1997

English text only

NEA/CSNI/R(97)11
Unclassified

NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

LEVEL 2 PSA METHODOLOGY AND SEVERE ACCIDENT MANAGEMENT

English text only

58677

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COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA) is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop, and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of the programme of work. It also reviews the state of knowledge on selected topics on nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the co-ordination of work in different Member countries including the establishment of co-operative research projects and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences and specialist meetings.

The greater part of the CSNI's current programme is concerned with the technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also cooperates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

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ABSTRACT

Current Level 2 PSA results and methodologies are reviewed and evaluated with respect to plant type specific and generic insights. Approaches and practices for using PSA results in the regulatory context and for supporting severe accident management programmes by input from level 2 PSAs are examined.

The work is based on information contained in:

- PSA procedure guides, PSA review guides and regulatory guides for the use of PSA results in risk informed decision making;
- Plant specific PSAs and PSA related literature exemplifying specific procedures, methods, analytical models, relevant input data and important results, use of computer codes and results of code calculations. The PSAs are evaluated with respect to results and insights.

In the conclusion section, the present state of risk informed decision making, in particular in the level 2 domain, is described and substantiated by relevant examples.

FOREWORD

For the NEA Committee on the Safety of Nuclear Installations (CSNI) an essential factor in achieving its mandate is the continuing exchange and analysis of technical information. To facilitate this exchange CSNI has established various working groups. To deal with technology and methods for analysing contributors to risk and assessing their importance, the Committee established Principal Working Group No. 5 (PWG5) - Risk Assessment, in 1982. The work programme of PWG5 was at first focused on PSA Level 1 methods, uses and assessments. In this area, data and methods were sufficiently consolidated to enable practical applications for improved reduction and control of risks. But in 1987, the Committee supported careful extension of “the consideration of PSA Level 2 issues where appropriate”.

Over the last 10 years the scope of PSA programmes increased progressively to where today, in some countries, a Level 2 PSA is considered the normal standard. Accordingly, with the advent of increasing use of PSAs, a proposal was made at the 1993 PWG5 Annual meeting for future work in the area of Level 2 PSA. The main objective of the proposed task was to perform a state-of-the-art review of the methods available for performing level 2 PSAs and severe accident/source term uncertainty analyses for use in the regulatory process and the evaluation/implementation of severe accident management strategies. This proposal was accepted by PWG5 and forwarded to the CSNI. The new task was endorsed by CSNI during its annual meeting in 1993.

The overall scope of the task included the review of current Level 2-PSA methodologies and practices and to investigate how Level 2-PSA can support severe accident management programmes, i.e. the development, implementation, training and optimisation of accident management strategies and measures. For the most part, the presented material reflects the state-of-the-art in 1996.

In offerings thanks to the task group members listed below, who provided valuable time and considerable knowledge towards the production of this report, the NEA Secretariat also wishes to provide acknowledge the specific service of several key persons and organisations. Dr. U. Schmocker who made the original proposal and whose organisation, HSK, provided substantial support throughout the task especially in formulation of Section 3.2. Dr. P. M. Herttrich who as task leader, provide clear insights on the objectives, skilfully chaired the many meetings and provided overall co-ordination towards completing the report. Dr. Wolfgang Werner, who as an expert consultant provided much of the in-depth technical analysis provided throughout the report as well as many man-hours in editing and compiling the final report. In addition Dr. Cazzoli, Dr. Cojazzi, Mr. Seebregts, Mrs. Otero, and Mr. Muramatsu provided meaningful input to various sections of the report.

Task Group Members contributing to the report were:

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1. TASK DESCRIPTION AND WORKING METHODOLOGY

1.1 Background

During its twelfth annual meeting in September 1993 PWG 5 discussed a proposal for future work in the area of level 2 PSA /SCU 93/:

The proposal was based on the observation, that the Probabilistic Safety Analysis (PSA) approach is becoming an integral part of safety decision making, especially in dealing with determination of plant specific vulnerabilities, plant backfits, operational and maintenance practices, development and evaluation of severe accident management (SAM) strategies.

Methods for Level 2 PSAs and severe accident modelling comprise traditional fault tree/event and containment event tree analyses of differing complexity, direct numerical simulation of physical models, including uncertainty analysis, and others. An important element of the analysis is the quantification of uncertainties in severe accident progression, and assignments of subjective probabilities at differing levels of decomposition for key severe accident phenomenological processes. Discussions are still ongoing as to the most efficient, transparent, and effective approaches for quantification of severe accident and source term uncertainties.

The main objective of the proposed task was to perform a state-of-the-art review of the methods available for performing level 2 PSAs and severe accident/source term uncertainty analyses for use in the regulatory process and the evaluation/implementation of severe accident management strategies. A document was to be prepared summarising PSA methods for use in the regulatory and licensing arena, including:

1. State-of-the art of PSA methods
2. Uncertainty issue quantification technique
3. Use of expert judgement
4. Identification of issues of key significance to Level 2 PSA and SAM for LWRs
5. Integrated and risk-based approach to regulatory decision making.

This proposal was accepted by PWG5 and forwarded to the CSNI as Task 16: Level 2 PSA Methodology and Severe Accident Management. The new task 16 was endorsed by CSNI during its annual meeting in 1993. Afterwards a task force has been established. In the following it will be described how the task force has approached the task and which items have been treated in the different chapters of this report.

1.2 Objective and Scope

The objective of the work was to review current Level 2-PSA methodologies and practices and to investigate how Level 2-PSA can support severe accident management programmes, i.e. the development, implementation, training and optimisation of accident management strategies and measures. For the most part, the presented material reflects the state in 1996.

1.3 Level 2 PSA methodologies

Two types of sources of information on current Level 2-methodologies have been used. The first type are procedure guides and review guides as listed in /NRC 82, NRC 89, KHA 92A, SEJ 91, MET 91/. In addition, the IAEA has recently finalised "Procedures for conducting probabilistic safety assessments of nuclear power plants" /IAE 93/ intended as a contemporary guide for conducting level 2 PSAs. The procedural steps and the individual tasks, as well as the methodological aspects discussed in this IAEA guide are used as a framework for the more detailed work within this task (table 1.3-1).

References of a second type for current level 2 methodology are plant specific PSAs that exemplify specific procedures, PSA-methods, analytical models, data and code calculations. These references have been evaluated with respect to results and insights and will be used on a case by case basis.

Scope and methods of a specific level 2 PSA are strongly influenced by its intended use. For the purpose of this task it was examined how level 2 PSAs can support accident management programmes and the regulatory use of level 2 PSA.

1.4 Accident management

Accident management is the pre-accident implementation of hardware and procedures and performance of a set of actions during the progression of an accident that are capable to return the plant to a controlled state and to mitigate any consequences of the accident. Accident management actions can be taken during different phases of accident progression depending on the severity of damage to plant systems, components and fuel. Typically, actions at the following phases are distinguished:

1. during the evolution of an event sequence before the design basis is exceeded, to prevent core damage and containment bypass sequences
2. during the progression of core degradation to prevent reactor vessel breach and containment failure
3. at or after vessel breach to prevent containment failure
4. during all phases to control releases with the objective to minimise off site consequences.

Actions during the first phase are termed “preventive accident management”, the other three actions “mitigative accident management” or “severe accident management” (SAM).

Accident management has first been introduced in nuclear power plants in the early eighties. The TMI-2 accident, severe accident analyses, and PSAs showed, that there is potential to control plant states even beyond design limits. Using PSA findings and deterministic code calculations, emergency operating procedures (EOP) have been extended from the design basis area to the severe accident domain.

For the development, verification, training and further improvement of AM/SAM, level 2 PSAs have provided major contributions.

1.5 OECD/NEA activities on accident management

Accident management has been a major subject of past CSNI and CNRA activities. The "Specialist Meeting on Severe Accident Management Programme Development" was the first meeting sponsored by CSNI on this subject /OEC 92A/. Three main areas were covered:

- approaches to severe accident management
- technical issues associated with severe accident management programmes for existing and future reactors
- severe accident management information needs and operator aids.

Based on the progress achieved by this meeting further activities have been performed by OECD.
Main OECD accident management references used for the purpose of this task are listed in table 1.5-1:

1.6 OECD/NEA activities on severe accident phenomena

Both accident management programmes and level 2 PSAs need a phenomenological basis. OECD-NEA has performed major activities in the area of severe accident phenomena. Table 1.6-1 contains a list of all relevant OECD reports. Reports reflecting the state of current knowledge and methodology on certain phenomena will be quoted in the context of the respective tasks of SAM-oriented level 2 PSAs.

1.7 Structure of the report

In each of the seven chapters the objective is to present the current state of the art. Main issues considered in the different chapters are:

In Chapter 2, state of application, results and insights from recent level 2 PSAs are presented and summarised.

Chapter 3 discusses key severe accident phenomena and modelling issues. The findings are used to identify severe accident issues that should be treated in level 2 PSAs in the context of accident management applications. To support discussion on modelling issues, severe accident computer codes are reviewed. Limitations in assessing the impact of various severe accident issues are discussed. In another section the use of severe accident computer codes in published PSAs is presented, as well as their potential impact on the results.

In Chapter 4, current approaches and practices in the area of AM/SAM are reviewed and evaluated with respect to investigations and evaluations that should be performed in level 2 PSAs. The overall requirements for AM/SAM oriented level 2 PSA are summarised.

One important issue is the treatment of emergency operating procedures and preventive modelled in a preceding level 1 PSA, because success or failure of such measures significantly influences plant damage states (PDS) that have to be selected as the initial states for the level 2 PSA. As the PDSs represent the interface between the level 1 and level 2 PSAs, they have to transfer as much information as possible as input to the SAM considerations.

Further issues are:

- the selection of appropriate AM/SAM-measures,
- the validation of AM/SAM-measures in view of phenomenological uncertainties
- the consideration of potential detrimental effects of SAM-measures
- the updating and optimisation of AM/SAM-measures using growing experience and ongoing research

Chapter 5 starts with a presentation of available level 2 PSA-methodologies. Those level 2-methods are described and reviewed that are most effective to support AM/SAM-programmes. This includes containment event tree development, especially the number and selection of nodal points for branching (timing and phases of accident progression, phenomena governing progression, human interventions), interim states and end states.

Aspects important to quantification, including the use of expert judgement and the proper treatment of uncertainties (optimal and adequate level for formulation of stochastic and state-of-knowledge uncertainty distributions) are discussed in Chapter 6.

In Chapter 7, examples are presented on approaches to use PSA results and insights in the context of risk informed decision making.

Table1.3-1 Major procedural steps and tasks of a level 2 PSA

Main steps	Tasks
Management and Organisation	1. Definition of the objectives of Level 2 PSA 2. Definition of the scope of the Level 2 PSA 3. Project management 4. Team selection and organisation 5. Establishment of a quality assurance program and interactive peer Review
Plant familiarisation and identification of design aspects important to severe accidents	6. Plant familiarisation 7. Identification of design aspects important to severe accidents
Interface to Level 1 PSA	8. PDSs for internal initiators at power 9. PDSs for an existing Level 1 PSA 10. Extension to other initiators 11. Extension to other power states
Accident progression and containment analysis	12. Containment performance 13. Severe accident progression analysis 14. Development and quantification of accident progression/containment event trees 15. Binning of event tree end-states into release categories/bins 16. Treatment of uncertainties in 17. Summary and interpretation of containment performance results
Severe accident source terms	18. Grouping of fission products 19. Release of fission products from fuel during the in-vessel phase 20. Retention within reactor coolant systems 21. Release during ex-vessel phase 22. Retention inside containment 23. Treatment of source term uncertainties 24. Presentation and interpretation of source term results
Documentation of the analyses: display and interpretation of results	25. Objectives and principles of documentation 26. Organisation of Level 2 PSA documentation

Table1.5-1 OECD references on "Severe Accident Management"

Proceedings of the Specialist Meeting on Severe Accident Management Programme Development, NEA/CSNI/R(91)16
Instrumentation for Accident Management in Containment, NEA/CSNI/R(92)4
Specialist Meeting on Severe Accident Management Programme Development - Summary and Conclusions - NEA/CSNI/R(92)6
Positive/Negative Aspects of Measures Designed to Protect the Containment NEA/CSNI/R(93)1
Hydrogen Management Techniques in Containment, NEA/CSNI/R(93)2
Specialist Meeting on Instrumentation to Manage Severe Accidents Summary and Recommendations, NEA/CSNI/R(93)3
International Standard Problem 29, Distribution of Hydrogen within the HDR Containment under Severe Accident Conditions, NEA/CSNI/R(93)4
Proceedings of the Specialist Meeting on Operator Aids for Severe Accidents Management and Training, NEA/CSNI/R(93)9
Proceedings of the 1. OECD (NEA) CSNI - Specialist Meeting on Instrumentation to Manage Severe Accidents, NEA/CSNI/R(92)11, GRS-93
Proceedings of the 1. OECD (NEA) CSNI - Specialist Meeting on Severe Accident Management - Prevention and Mitigation, Paris 1992

Table 1.6-1 OECD references on "Severe Accidents"

Proceedings of the Workshop on Aerosol Behaviour and Thermal-Hydraulics in the Containment, CSNI Report No. 176
Workshop on Aerosol Behaviour and Thermal-Hydraulics in the Containment - Technical Summary - NEA/CSNI/R(92)1
Source Term Uncertainties - Recent Developments in Understanding Fission Product Behaviour, NEA/CSNI/R(92)2
Flame Acceleration and Transition to Detonation in Hydrogen/Air/Diluent Mixtures, NEA/CSNI/R(92)3
Workshop on Iodine Chemistry in Reactor Safety - Summary and Conclusions, NEA/CSNI/R(92)5
Positive/Negative Aspects of Measures Designed to Protect the Containment, NEA/CSNI/R(93)1
Hydrogen Management Techniques in Containment, NEA/CSNI/R(93)2
Specialist Meeting on Core Debris/Concrete Interactions - Summary and Recommendations NEA/CSNI/R(93)5
Effects of Hydrogen Combustion of Fission Products and Aerosols, NEA/CSNI/R(93)6
Physical and Chemical Characteristics of Aerosols in the Containment, NEA/CSNI/R(93)7
Report by a NEA group of experts: Severe Accidents in Nuclear Power Plants, OECD/NEA/CSNI, Paris 1986
The Role of Nuclear Reactor Containment in Severe Accidents, OECD/NEA/CSNI, Paris 1989

2. RESULTS AND INSIGHTS FROM RECENT LEVEL 2 PSA

The referencing to publications in this section refers to the list in subsection 2.6.

2.1 Examined PSAs and considered aspects

Nineteen recent level 2 PSAs - eleven for PWRs with large dry containments and eight for BWRs with various containment designs - /1-32/- have been evaluated and compared with respect to issues important to the objectives of this report. In table 2.1-1 the main steps and aspects of the evaluation are listed and assigned to the main steps of a level 2 PSA (compare figure 1.1 /IAEA/).

This chapter provides an overview of the examined studies and issues, and summarises the main results and insights of the evaluation. Details can be found in the report /WER 96/.

Table 2.1-1 Main steps and principal aspects of the evaluation of Level 2 PSAs

Main steps	Principal aspects of the evaluation
Management and organisation	Objectives and scope of the PSA
Familiarisation with the plant	Plant characteristics that are decisive for the progression and the consequences of beyond-design-basis sequences
Interfaces with Level 1 PSA	Definition of plant damage states (including the possibility of restoring lost system functions and/or the consideration of preventive AM measures)
Accident progression and containment analysis	Methodology for developing containment event trees. Classification and assessment of containment failure mode.
Source terms	Selection and treatment of phenomena influencing the releases. Methodology for calculating source terms (representative releases, parametric calculations)
Documentation of the analysis: Presentation and interpretation of the results	Qualitative results Quantitative results Conclusions and insights

2.2 Objectives and scope of recent level-2 PSAs

A level 2 PSA analyses:

- event sequences progressing from core degradation to core destruction,
- the attending loads and failure modes of the reactor coolant boundary,
- events inside the containment resulting from breaches of the reactor coolant boundary and the attending loads of the containment structure,
- containment failure modes resulting from such loads, including containment bypass sequences,
- all aspects of fission product transport and retention inside the reactor coolant system and containment,
- fission product release to the environment due to degraded containment function or containment bypass.

Objective of the analyses is to generate:

- an overview of the spectrum of possible severe accidents,
- insights into the principal causes for such events,
- insights on important phenomena governing such events,
- a grouping and importance ranking of typical severe accident sequences, dependent on plant design characteristics.

Such results provide important input to the implementation of additional protective measures for coping with severe accident conditions beyond the plant's design basis. The analyses incorporate current experimental and analytical results of severe accident research. In some of the research areas there still exist significant uncertainties.

Mitigation of severe accident consequences is attempted by putting in place hardware and procedures for:

- protecting the containment function,
- avoiding difficult to control high consequence event sequences, by transforming them into sequences with lesser consequences
- providing additional fission product retention for sequences with degraded containment function.

Examples for objectives typically quoted in level 2-PSAs or respective procedure guides are compiled in Table 2.2-1.

Table 2.2-1 Principal objectives of level 2 PSAs

• To gain insights into the progression of severe accidents and containment performance
• To identify plant specific vulnerabilities of the containment, including containment bypass sequences, to severe accidents
• To identify major containment failure modes, including containment bypass sequences, and to estimate the corresponding releases of radionuclides
• To evaluate the impacts of various uncertainties, including assumptions relating to phenomena, systems and modelling
• To provide a basis for the development of plant specific accident management strategies
• To assess respective plant conditions determining the diagnosability, availability or restorability of safety functions (protective functions)
• To select and specify candidate SAM-measures
• To assess the efficiency of countermeasures and the feasibility of operator interventions
• To check the compatibility of SAM-measures with designed safety features
• To assess the robustness and sufficient completeness of SAM-strategies
• To identify events or phenomena important to risk that need further research
• To provide a basis for plant specific backfit analysis and evaluation of risk reduction options
• To provide a basis for the prioritisation of research activities for minimisation of risk significant uncertainties
• To provide a basis for the resolution of specific regulatory concerns
• To provide a basis for the evaluation of off-site emergency planning strategies
• To provide a basis for the demonstration of conformance with quantitative safety criteria

2.3 Plant characteristics influencing severe accident progression

In tables 2.3-1 and 2.3-2, the examined studies for PWRs, respectively BWRs, are compiled. Also included are plant characteristics with major influence on accident progression. Plant/containment design characteristics with influence on the capabilities of the plants to respond to severe accident challenges and provisions for severe accident management available at the plants are shown in tables 2.3-3 and 2.3-4.

Information and results on a Canadian CANDU reactor (Pickering A) is included in the PWR tables.

Below is a listing of phenomena that have been identified in the examined studies to have major influence on the evolution of severe accidents and on potential failures modes of the containment/confinement function.

- Generation of hydrogen during in-the vessel phase of core degradation.
- Arrest of core melt progression.
- Mitigation of the consequences of steam generator tube rupture (SGTR) events by scrubbing in a water filled steam generator.
- Mitigation by suppression pool scrubbing in BWRs.
- Temperature induced structural failure of the reactor coolant system (RCS).
- Containment loads resulting from RPV failure under high pressure.
- Erosion of containment basemat in PWR plants.
- Ex-vessel hydrogen production in PWR containments.
- Containment failure due to slow pressure build-up.
- Drywell-attack by molten corium in BWRs with Mark I containments.

Other phenomena that may influence the progression of the accident and size of the source term are discussed in Chapter 3.

The physical and chemical interactions as well as the influences of systems or human interventions governing the progression of severe accidents are very complex, involving many uncertainties. Some phenomena are stochastic in nature, some phenomena are not fully understood. Therefore, many alternative accident progressions have been considered in the different studies based on experimental and theoretical severe accident research, and on calculations with parametric and with mechanistic codes.

The codes are applied to model the interaction of phenomena, but they are not always able to analyse a wide range of accident scenarios with largely different boundary conditions. Typically, analysis results are available only for a small set of sequences. Therefore, in many recent Level 2 PSAs expert judgement is introduced in a systematic way to quantify dominant influences on the accident progression.

Tables 2.3-3 and 2.3-4 provide an overview of features or measures for severe accident management available at the examined plants. More detailed information is presented in Chapter 4.

Table 2.3-1. PWRs with Large Dry Containments. Plant/Containment Design Characteristics

Plant Characteristic/ relevant for	**Surry**	**Zion**	**Maine Yankee**	**Robinson**	**Beznau**	**Biblis-B**
Power, MWth	2441	3250	2630	2300	1130	3750
Containment type / Load capacity, tightness	Concrete	Concrete	Concrete	Concrete	Steel	Steel
Containment volume (m^3)	51000	81000	52600	59400	36400	70000
Power/Containment volume/ Containment loads, Time scale of accident Time budget for AM	0.048	0.040	0.05	0.039	0.031	0.054
Fuel mass (kg)	79000	98000	80000	78900	43500	
Fuel mass / Containment volume/ Containment loads from DCH	1.5	1.2	1.5	1.3	1.2	
Zirconium mass (kg)	16500	20050	24300	16335	12000	29750
H_2 -mass (kg), with 100% Zr-oxidation.	780	880	1062	718	530	1350
Average H_2-concentration (%), at 30^0C, dry, 100% Zr-oxidation/ Potential for H_2 burn Containment loads from H_2 burn	15	10.8	20.2	12.1	12.8	19
Estimated pressure (bar) due to H_2 burn, 100% Zr-oxidation	9.4	6.7	12.4	7.6	7.9	11.7

Plant Characteristic/ relevant for	**Sizewell-B**	**Ringhals 2**	**Borssele PSA-3 (PSA97)**	**Japan 1100 Mwe PWR**	**Pickering A (CANDU)**
Power, MWth	3411	2660	1365	3441	1760
Containment type / Load capacity, tightness	Double primary: prestressed concrete secondary: concrete	Concrete	Double, inner. Steel outer: concrete	Prestressed concrete	Concrete
Containment volume (m^3)	85500	58000	37130	73300	81300 (accident unit + pressure relief duct) + 438700 (vacuum building + other units) (Total: 520000)
Power/Containment volume/ Containment loads, Time scale of accident Time budget for AM	0.04	0,045	0.037	0.046	0.0034
Fuel mass (kg)	101.000	69000	42955	89500	101.300
Fuel mass / Containment volume/ Containment loads from DCH	1.18	1.2	1.16	1.2	0.19 (total volume)
Zirconium mass (kg)	19600	16400	9910	19500	9600 in fuel sheaths, 34500 in channels and rest
H_2 -mass (kg), with 100% Zr-oxidation.	866	770	435	855	648 (100% oxidation in fuel sheets, 15% other)
Average H_2-concentration (%), at 30^0C, dry, 100% Zr-oxidation/ Potential for H_2 burn Containment loads from H_2 burn	10	13.2	11.5	12.4	8.9% (with above oxidation)
Estimated pressure (bar) due to H_2 burn, 100% Zr-oxidation	6.3	8.2	7.1	6.7	not assessed

Table 2.3-2. BWRs. Plant/Containment Design Characteristics

Plant Characteristic / relevant for	Peach Bottom	Browns Ferry	Grand Gulf	Perry	Mühleberg
Power, MWth	3.293	3.293	3.833	3.579	1.097
Containment type	MK I	MK I	MK III	MK III	MK I
Estimated containment failure pressure, bar / Load capacity, tightness	10.7	10.7	6.5	6.5	9.4
Containment volume, m^3	8.230	8.100	40.300	40.800	5.000
Power/Containment volume/ Containment loads, time scale for accident, time budget for AM	0.4	0.4	0.097	0.09	0.21
Power/Suppression pool volume/ Likelihood of containment challenges	0.9	0.9	1.0	0.8	0.5
Fuel mass, kg	159.400	155.600	166.200	156.000	48.500
Fuel mass/containment volume/ Containment loads from DCH	19.4	19.2	4.2	3.9	9.5
Zirconium mass, kg	65.500	55.000	80.000	75.000?	24.000
H_2-mass, kg, 100% Zr-oxidation / Potential for H_2 burns	2.850	2.410	3.510	3.290	1.044

Plant Characteristic / relevant for	Swedish Generation I/II	Swedish Generation III/IV	Dodewaard	La Salle	Japan 1100 Mwe BWR
Power, MWth			183	3293	3293
Containment type			Humbold Bay (pre-MKI)	MK II	MK II
Estimated containment failure pressure, bar / Load capacity, tightness			7	13.4	9.4
Containment volume, m^3			754	10300 ???	10300
Power/Containment volume/ Containment loads, time scale for accident, time budget for AM			0.24	0.3 ???	0.3
Power/Suppression pool volume/ Likelihood of containment challenges			0.45	1.0 ???	1.0
Fuel mass, kg			10.550		
Fuel mass/containment volume/ Containment loads from DCH			14		11.2 (based on UO_2), 9.7 (based on U)
Zirconium mass, kg			4.338		57000
H_2-mass, kg, 100% Zr-oxidation / Potential for H_2 burns			190		2940

Table 2.3-3. PWRs with Large Dry Containments. Plant/Containment Design Characteristics, Provisions for Accident Mitigation

Plant Characteristic/ relevant for	Surry	Zion	Maine Yankee	Robinson	Beznau	Biblis-B
Estimated containment failure pressure (bar)	9.7	10.2	10.5	10.4	7.8	8
Containment spray / Potential for hydrogen burn, late containment failure by overpressurisation	yes	yes	yes	yes	yes	no
Hydrogen control / Potential for hydrogen burn					containment venting	Igniters/ recombiners
Additional water injection to containment / Ex-vessel cooling of core debris, late containment failure by overpressurisation and/or BMP					External from fire truck: • backup water source for containment spray • flooding of containment • External from river for cooling of fan coolers	
Depressurisation of RCS for prevention of HPME ("primary side bleed") / Likelihood of DCH	Transients with loss of all FW	Most events with loss of all FW	Transients with loss of all FW	Transients with loss of all FW	Transients SLOCA with loss of all FW	Most events with loss of all FW
Depressurisation of containment/ Late containment failure by overpressurisation					filtered containment venting	filtered containment venting
Use of PB/F in the event of SGTR / Release attending SGTR		yes			yes	yes
Filling of SG with water in the event of SGTR / Release attending SGTR					yes	under study

Table 2.3-3. (cont'd.) PWRs with Large Dry Containments. Plant/Containment Design Characteristics, Provisions for Accident Mitigation

Plant Characteristic/ relevant for	Sizewell-B	Ringhals 2	Borssele PSA-3 (PSA97)	Japan 1100 Mwe PWR	Pickering A (CANDU)
Estimated containment failure pressure (bar)	10.1	12	8		2.2
Containment spray / Potential for hydrogen burn, late containment failure by overpressurisation	yes	yes	yes	yes	yes, in vacuum building
Hydrogen control / Potential for hydrogen burn	Hydrogen mixing in the short term Recombiners in the long term	Recombiners	under study: • early venting • combination of recombiners/ igniters • post-accident inertisation/ recombiners	combination of recombiners and inertisation of atmosphere	Igniters
Additional water injection to containment / Ex-vessel cooling of core debris, late containment failure by overpressurisation and/or BMP	Fire water system for debris quenching and cooling	External from fire truck (CWIS) for: • backup water source for containment spray • flooding of containment		under preparation: water injection from RWST and fire water system	-
Depressurisation of RCS for prevention of HPME ("primary side bleed") / Likelihood of DCH	All events with high primary pressure	Most events with loss of all FW	Most (all) events with high primary pressure	Most events with loss of all FW	inherent due to failure of fuel channels, given core melt
Depressurisation of containment/ Late containment failure by overpressurisation		filtered containment venting	filtered containment venting		filtered containment venting
Use of PB/F in the event of SGTR / Release attending SGTR	yes	yes	yes	yes	yes
Filling of SG with water in the event of SGTR / Release attending SGTR	yes	yes	yes		-

Table 2.3-4. BWRs. Plant/Containment Design Characteristics, Provisions for Accident Management

Plant Characteristic / relevant for	Peach Bottom	Browns Ferry	Grand Gulf	Perry	Mühleberg
Primary containment venting / Containment overpressure failure	yes	no hardened vent path	yes	yes	yes, filtered
Flooding of lower drywell / Prevention of liner failure					yes, using fire water system
Additional water injection to containment / Containment protection					Fire water system, optionally with external water supply, for • back-up water source for containment spray • flooding of containment

Plant Characteristic / relevant for	Swedish Generation I/II	Swedish Generation III/IV	Dodewaard	La Salle	Japan 1100 Mwe BWR
Primary containment venting / Containment overpressure failure	yes, filtered	yes, filtered	yes, filtered is planned	yes	yes, hardened vent path planned
Flooding of lower drywell / Prevention of liner failure	yes	yes	planned, by wetwell-drywell connection		
Additional water injection to containment / Containment protection	Fire truck (CWIS) for back-up water source for • containment spray • flooding of containment Fire water pool for flooding of containment	Fire truck (CWIS) for back-up water source for • containment spray • flooding of containment Fire water pool for flooding of containment	Fire protection system as backup for containment spray		Fire water injection planned

2.4 Level 2 methodology and codes

The principal approach of the level 2 analyses in the examined studies are similar:

- definition of the initial conditions by binning of level 1 end states into plant damage states
- development of event trees: containment event trees (CET), accident progression event trees (APET)
- determination and evaluation of containment failure modes
- binning of containment states related to specific containment failure modes according to their release characteristics.

The APETs are developed in similar steps:

- establishing a set of questions about possible events
- design of the logic structure that forms the tree
- decision on events and phenomena to be included
- selection of quantities influencing branching probabilities
- analysis of dependencies between questions
- review of the consistency of paths especially with respect to the physical reality
- identification of risk-important, but uncertain issues, for expert judgement

In the examined plants there is considerable difference in the numbers of

- plant damage states,
- nodes used in the containment event tree analysis,
- containment end states,
- release categories

and the criteria used for the definition of these items, see tables 2.4-1 and 2.4-2.

In principle, such differences do not impair the quality of the results as long as the relevant factors that influence the evolution of the accident are treated in the detail necessary for the individual steps, and the information required in the subsequent steps is properly propagated.

The proper handling of the large number of plant and containment states requires special computational tools. For the NUREG 1150-studies the EVNTRE code has been developed. For other studies Level 1-PSA codes like RISKSPECTRUM have been adapted to Level 2 needs. New codes like the SPSA in Finland or SOLOMON in the UK are being developed.

The features these codes permit to perform the necessary uncertainty analyses.

Further details are presented in Chapter 5.

Table 2.4-1. Methods Used and level of detail of the logical models in the examined PSAs, PWRs with Large Dry Containment

Plant	Method	Number of PDS, > 10^{-7}/a	Number of Summary PDS	Number of CET/APET Nodes	Number of Source Term Bins
Surry, NUREG-1150	Large event tree	25	7	71	7
Zion, NUREG 1150	Large event tree	18	5	72	4
Maine Yankee, IPE	Small event tree	17	10	14	18
Robinson, IPE	Small event tree	21	-	12	8
Beznau, HSK/ERI	Small event tree	28	11	33	18
Sizewell-B	Small event tree	20	30	20	331/ (22 release categories)
Ringhals 2	Small event tree	23	23	12	16
Borssele PSA-3 (PSA-97)	Small event tree	25	111	51	13 ((16)
Japan 1100 Mwe PWR	Medium event tree	7	6	20	12
Pickering A	Small event tree	25 (irrespective of frequency	-	9	7

Table 2.4-2. Summary of methods and level of detail in the examined PSAs, BWRs

Plant	Method	Number of PDS, > 10^{-7}/a	Number of Summary PDS	Number of CET/APET Nodes	Number of Source Term Bins
Peach Bottom, NUREG-1150	Large event tree	9	4	145	10
Browns Ferry, IPE	Large event tree	20	8	125	10
Grand Gulf, NUREG-1150	Large event tree	12	4	125	8
Perry, IPE	Medium event tree	12	4	68	25
Mühleberg, HSK/ERI	Small event tree	6	6	18	15
Swedish Generation III/IV	Small event tree	7	7	10	5
Dodewaard	Medium event tree	44	6	70	10
La Salle	Large event tree	30 (including external events)	7 (including external events)	135	20
Japan 1100 Mwe BWR	Small event tree	4	8	21	46

2.5 Principal results, insights on containment failure modes and releases.

2.5.1 Pressurised water reactors

Main results and insights from level 2 PSAs for PWRs are presented in table 2.5.1-1 and illustrated by Figure 2.5-1. They can be summarised as follows:

- The largest releases generally result from containment bypass sequences, most notably steam generator tube rupture with unisolated steam generator. Therefore, the largest benefit in terms of reduction of offsite consequences is to be expected from severe accident management directed at mitigating the consequences from such accident sequences. Two strategies are reported in the examined PSA studies:
  - Application of primary side bleed/feed (PB/F) to steam generator tube rupture events. This is implemented at Biblis-B, Zion, Borssele, Beznau, Sizewell-B (but not credited in the PSA), the Swedish and many US PWRs. The application of PB/F to steam generator tube rupture events reduces
    - the occurrence frequency of such events
    - the release to the environment from the ruptured steam generator, because, by the split-up of mass flow between open pressuriser valves and the ruptured steam generator heating tube, the majority of the fission products released from the core is directed to the containment. Due to the long time to RPV failure in such accident sequences, significant depletion of fission products can take place inside the containment.
  - Filling up of the ruptured steam generator with water. The effectiveness of fission product scrubbing in the water column was investigated in the DRS-B study. The reported calculations show significant reductions of fission product releases. The strategy, using fire water (severe accident management), is now implemented and credited in the PSA at the Beznau and Borssele plants. At Sizewell-B it is implemented, but not credited in the PSA. At the Swedish plants a different strategy is being studied which aims at maintaining a high water level in the defective steam generator from the onset of the accident (except for accident sequences with extremely low frequencies of occurrence). On the negative side of this severe accident management strategy could be loss of steam generator integrity due to pressurised thermal shock. This possibility is discussed in the Robinson IPE.

  Further investigations into the subject appear to be necessary.

- Releases resulting from LOCAs at the high pressure system / low pressure system interface (V-sequence) have been made extremely unlikely at the plants examined in this comparison, due to improved redundancy/diversity of the high pressure system / low pressure system isolation, and improved surveillance, testing and maintenance strategies for the interface. For example, at Sizewell-B the frequency of an interfacing systems LOCA has been reduced by a number of measures that include increasing the extent of the pipework qualified to withstand full reactor pressure and temperature, and incorporating a third valve in the RHRS suction lines with diverse interlocks to the existing valves to prevent them being opened when the RCS pressure is above the normal operating pressure of the RHRS. At plants with very low frequencies of LRCF modes (order of 10^{-7} or lower), like Borssele PSA '97 and the Konvoi plants, isolation failures can be significant contributors to large releases because it is difficult to show that the frequency of V-sequences is below 10^{-7}/a.

- Early containment failure which potentially can cause large releases, is very unlikely at most plants with large dry, pre-stressed concrete containments. Due to their robust design, they can absorb the majority of the loads attending the early phase of severe accidents.
- Containment isolation was a concern in the original IPE for the Robinson plant. Plant modifications implemented since have removed this vulnerability.
- Due to their lower failure pressure, containments with steel shell construction are less resilient to such loads, than the large dry containments At the Beznau and Borssele plants, the lower failure pressure of the steel shell containment is compensated for by lower ratios of "(reactor power)/(containment volume)" and "(fuel mass)/(containment volume)" that lead to reduced severe accident loads.
- A parameter important to early containment failure is the amount of zirconium in the reactor core, which can vary considerably, depending on the fuel manufacturer. Among the investigated plants, there is substantially more zirconium in the core of the Maine Yankee and the Biblis-B reactor, than in the other PWRs. This leads to increased vulnerability to hydrogen combustion in the early (and late) phase of severe accident situations with operating containment sprays (at Maine Yankee). This issue could also become important at other plants if other fuel with thicker cladding will be reloaded.

  As defence against the threat from hydrogen combustion a combination of igniters and catalytic recombiners will be implemented at Biblis-B (and other German PWRs) and are being studied for the Borssele plant. At Beznau, hydrogen and oxygen can be removed from the containment atmosphere through the venting line. This strategy, as well as post accident inertisation is also being studied for the Borssele plant. For Sizewell-B, hydrogen control is achieved by mixing the hydrogen produced in the whole of the containment atmosphere in the short term and using hydrogen recombiners in the longer term. The hydrogen mixing is carried out by mixing fans assisted by the containment sprays and coolers. The hydrogen recombiners are only designed for the post LOCA duty.
- The releases from late containment failure are much lower than from the accident sequences described above, with the exception of Sizewell B, where late containment failure is the main contribution to large releases. The occurrence of late containment failure can be made less likely if filtered containment venting is available (currently at German and Swedish PWRs, Beznau Borssele). A further reduction of releases from late containment failure modes is possible by the addition from internal or external sources of large quantities of water for debris quenching, in combination with the availability of high capacity filtered venting. Such accident mitigation strategy is implemented at Beznau and at the Swedish plants. At Sizewell B, water addition from the fire water system is possible, but filtered containment venting is not available The PSA was used to show that it would not be cost effective to incorporate a filtered venting system

  There have been concerns about increasing the potential for steam explosions by the water injection strategy. However, with conditional probabilities of containment failure due to steam explosions in the range 10^{-4} to 10^{-3} or lower (see the discussion in section 3.3), the positive effect seems to outweigh the drawback.

Table 2.5.1-1 PWRs with Large Dry Containments. Frequencies and conditional probabilities of significant and large Cs releases. Dominant phenomena and their relative contribution

Plant	Frequency/a of		Exceedance frequency/a for		Conditional probability of exceeding	
	Total CDF	ECF +Bypass + ISF	1% release	10% release	1% release, given core damage	10% release, given ECF + Bypass + ISF
Surry	4.0 E-5,	5.1 E-6	6 E-6	2 E-6, SGTR >90%	0.15	0.39
Zion	6.5 E-5,	1.5 E-6	5,5 E-6	1 E-6, SGTR ~30%, DCH: ~70%	0.08	0.66
Maine Yankee	7.4 E-5,	7.4 E-6	4.4 E-6	1.4 E-6, SGTR: ~20%, H_2 burn: ~80%	0.06	0.19
Robinson	2.4 E-4,	8.6 E-6	2 E-5	2 E-6, SGTR ~50%, DCH: ~50%	0.1	0.23
Beznau	4.4 E-6,	5.3 E-7	1.2 E-7	3 E-8, SGTR ~45%, DCH ~55%	0.03	0.05
Biblis-B. Releases were quantified only for SGTR with low RCS pressure. Frequency of high pressure SGTR sequences: 1 10^{-8}	2.9 E-6,		< E-8, PB/F with scrubbing in SG >E-8 otherwise (only SGTR)	<<E-8, with scrubbing in SG > E-8 otherwise (only SGTR)		
Sizewell-B, conservative	2.2 E-5	2 E-6	8 E-6	5 E-6 late overpressurisation: 80%	0.36	0.25, given LRCF mode 0.99, given late overpressurisation 0.22, given core damage
Ringhals 2	2.0 E-5,	1.8 E-6	2E-7, ECF ca. 50%, isolation failure ca. 50%	5 E-8, ECF > 90%	0.01	0,03
Borssele PSA-3	3.6 E-5	8 E-7	8 E-7	3 E-7, V-seq.: 70%	0.02	0.37
Borssele PSA-97	1.7 E-6	1.1 E-7	1.5 E-7	1 E-7, V-seq.: 70%, ISF: 15%	0.08	0.6
Japan 1100 Mwe PWR	1.9 E-6,	7 E-7	7.4 E-7	6.9 E-7	0.39 (without credit to SAM)	0.36, given core damage (without credit to SAM)
Pickering A	1.3 E-4		< 1 E-7	< 1 E-8	< 8 E-4	?

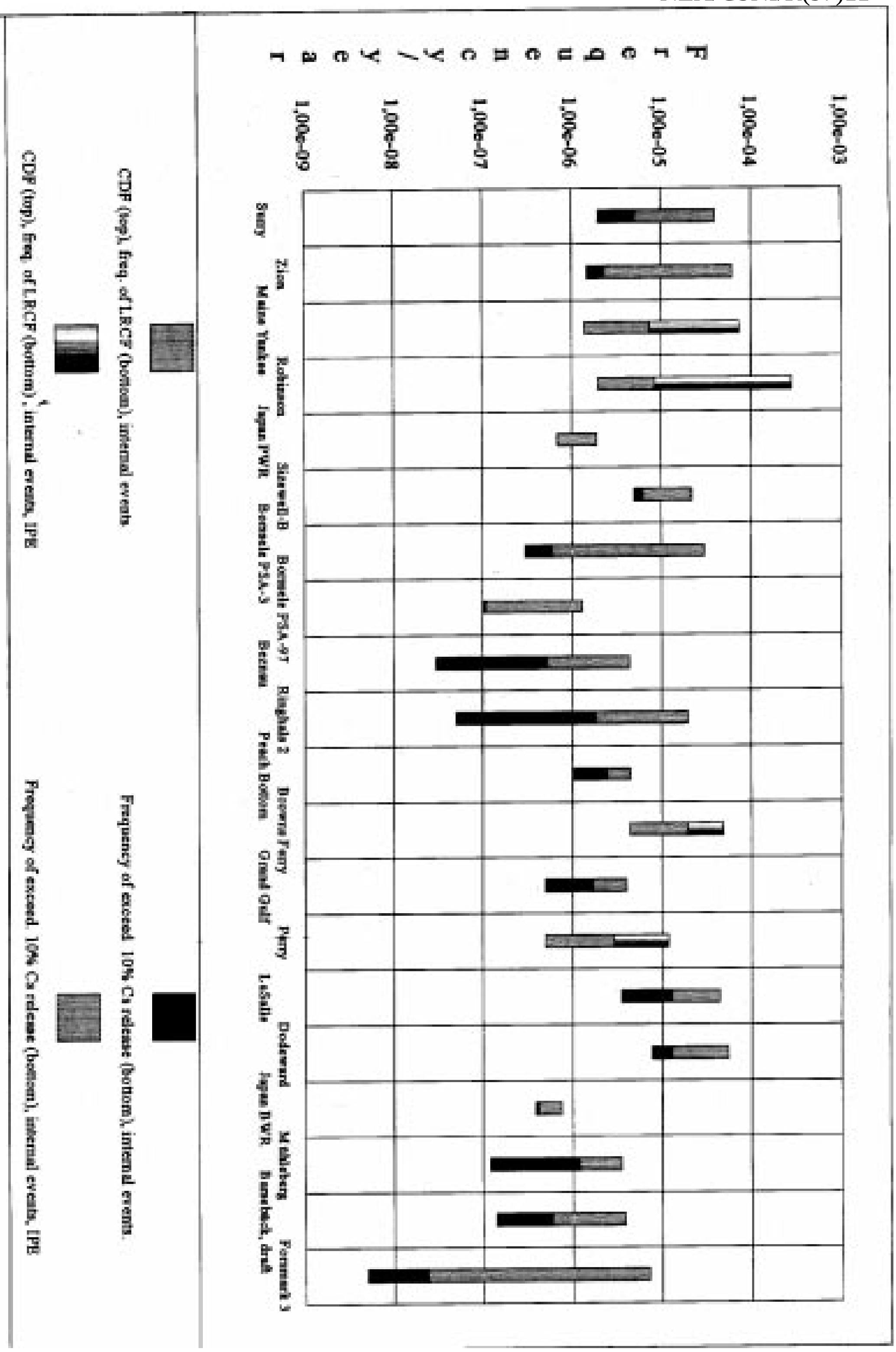

Figure 2.5-1. Frequencies of CD, LRCF, exceedance of 10 % Cs release, internal events

2.5.2 Candu Reactors

The PSA for Pickering A shows the following: in accident scenarios with total loss of core cooling, CANDU fuel channels fail early. This leads to rapid depressurisation of the reactor coolant system, initially to the calandria and subsequently - via the calandria rupture disk - to the containment. As a result, containment bypass events are of relatively low off-site consequence. Any further core degradation occurs at near containment pressure, reducing the potential for energetic release of core material and fission products. This limits the potential for challenges to containment integrity.

2.5.3 Boiling water reactors

Main results and insights from level 2 PSAs are presented in Table 2.5-2 and illustrated by Figure 2.5-2. They can be summarised as follows:

- In BWRs with MK I containment, there is the potential for early containment failure due to melt through of the drywell liner under attack of molten corium. In most studies, this early containment failure mode was the main source for offsite consequences. The uncertainties associated with that issue were very large at the time of conduction of the earlier studies included in this comparison. Recently, research results have become available suggesting that the probability of liner melt through could be significantly reduced if sufficient amount of water was available on the drywell floor. At the Mühleberg plant, with its large in-pedestal sump volume, this scenario is de-facto eliminated.
- Strategies for making available large quantities of water
  - as back-up water source for containment (drywell) spray
  - to keep a damaged core inside the RPV by outside cooling,
  - or - should this fail -
  - to protect the containment structures against attack by molten corium
  - are implemented at all Swedish BWRs, with water either from internal sources (fire water system) or from external sources (external containment water injection from fire trucks)
- Containment venting for avoiding late overpressure failure of the containment is available at all plants included in this comparison. At the Swiss and Swedish BWRs, high capacity filtered venting devices are implemented. In addition to severe accident mitigation, containment venting is also used for alternate heat removal in situations with failed suppression pool cooling. At most Swedish BWRs an additional unfiltered high capacity venting system is also available for this purpose.
- At Mühleberg, Forsmark 3 and at Barsebäck the combination of the possibility to flood the containment from internal or external sources and of overpressure protection by filtered containment venting lead to exceptionally low conditional probabilities for early containment failure and for significant and large releases.
- At Mühleberg, the large in-pedestal sump volume and the existence of a hardened containment with a vent path through an outer torus also contribute to the low releases.

- All Mark I containments are inerted, therefore, hydrogen combustion is of no concern at these plants.
- The load capacity of MK III containments is substantially lower than for MK I containments. In part, this drawback is compensated by the large volume of the containment that encloses drywell and wetwell. Nevertheless, the conditional probabilities of early containment failure due to hydrogen combustion and production of steam are significant. Because of their large volume, the Mark III containments are not inerted. As defence against hydrogen combustion, igniters are provided. Recently, igniters have been backfitted at some plants to no longer depend on DC power.

Table 2.5.3-1 BWRs. Frequencies and conditional probabilities of significant and large releases, given core damage. Dominant phenomena and their relative contribution

Plant/PSA	Frequency/a of		Exceedance frequency/a for		Conditional probability of exceeding	
	Totality of containment failure modes	ECF + bypass + ISF	1% release	10 % release	1% release, given core damage	10% release, given ECF + Bypass + ISF
Peach Bottom, NUREG-1150	4.3 E-6	2.4 E-6	2 E-6	1.3 E-6, liner failure	0.46	0.54
Browns Ferry, IPE	4.8E-5	2.2 E-5	1.2 E-5	5 E-6, liner failure	0.25	0.22
Mühleberg, HSK/ERI	3.5 E-6	9 E-7	3.3 E-7	1.2 E-7, early overpressure failure	0.1	0.13
La Salle	4.4 E-5	1.5 E-5	1.4 E-5	3.6 E-6, overpressure, CCI	0.32	0.24
Grand Gulf, NUREG-1150	4.1 E-6	8.6 E-7	1.5 E-6	5 E-7, hydrogen burn	0.36	0.58
Perry, IPE	1.2 E-5	1.9 E-6	4 E-6	5 E-7, hydrogen burn	0.33	0.26
Barsebäck 1/2 (Draft)	3.9 E-6	3.9 E-7	5.4 E-7	1.4 E-7, steam line isolation failure, CCI, impact of vessel head failure	0.13	0.36
Forsmark 3	7.2 E-6	2.4 E-8	2.7 E-8	5 E-9 containment bypass	0.0038	0.2
Japan 1100 Mwe BWR	7.6 E-7	3 E-8	6.5 E-7	4.2 E-7	0.86 (without credit to SAM)	0.56, given core damage (without credit to SAM)

2.5.4 Remarks

Inspection of Figure 2.5-1 hints at a tendency in the in the examined IPE studies to calculate lower conditional probabilities of large release containment failure (LRCF) modes, and lower conditional probabilities of large releases, given a LRCF mode.

All conclusions presented here apply to the plants included in this comparison. Any inference made to other plants requires caution, even if the designs are similar, because plant specific features could significantly affect the response of a plant to severe accident loads.

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3. KEY SEVERE ACCIDENT ISSUES

3.1 Key Severe Accident Phenomena

The referencing to publications in this section refers to the list in subsection 3.1.4.

A level 2 PSA requires the analysis of complex physical and chemical processes for which only limited experimental data are available. The phenomena to be considered in the course of the accident after the onset of core degradation can be grouped into two categories:

1. Phenomena associated with the thermal-hydraulics of the accident progression and the associated containment response. These phenomena range from hydrogen generation and core-material relocation during the in-vessel phase to containment failure due to loads attending the core destruction process. The associated analysis is called: “Accident progression and containment performance analysis”.
2. Phenomena associated with the chemical processes affecting the radionuclides during the accident and the transport of the radioactive material from the fuel through the containment to the environment. The associated analysis is called : “Source term analysis”.

In the accident progression and containment performance analysis the following main issues can be distinguished:

- Plant damage states, onset of core uncovery[1]
- In-vessel core degradation
  - Heat-up phase including severe accident thermal hydraulics
  - Oxidation/hydrogen generation
  - Chemical interactions (e.g., eutectics) amongst core materials
  - Cladding failure
  - Relocation and blockage formation
  - Core collapse
  - Late phase of core melt
  - Natural circulation leading to RCS-failure prior to RPV-failure
  - Influence of refill/quench on phenomena like hydrogen generation, pressure inside RCS and thermal shock.

[1] Often this part of the analysis is considered to be part of the level 1 PSA. Also the plant damage state analysis is often indicated as level 1 level 2 interface.

- Vessel attack, RPV-failure mode and core debris release to the containment.
  - Energetic in-vessel steam explosions
  - High -pressure melt ejection
  - RPV lower head gross failure by global creep rupture
  - RPV lower head local failure by:
  - jet impingement,
  - plugging and failure of lower head penetrations,
  - ejection of a lower head penetration
  - global creep rupture
  - Late slump, subsequent to the first main corium discharge
- Energetic ex-vessel phenomena immediately or shortly after vessel breach
  - Direct Containment Heating (DCH)
  - Ex-vessel steam explosions
  - Internal missiles due to energetic RPV failure.
  - Hydrogen burns, including deflagration and detonation
- Ex-vessel phenomena in the long term phase.
  - Corium-concrete interactions
  - Direct melt attack on other structures in the containment
  - Late overpressurisation
- Containment failure modes, including.
  - bypass (SGTR, ,V-sequence, ISF),
  - impact by an energetic missile (α-mode failure),
  - overpressurisation (due to hydrogen combustion, generation of steam and non-condensable gases),
  - basemat meltthrough.

In the source term analysis the following issues are important:

- Inventories of radionuclides and structural materials in the core
- In-vessel radionuclide release and transport
  - Iodine and caesium chemistry
  - Chemistry of other isotopes
  - Retention and deposition of fission products inside RCS

- Ex-vessel radionuclide release and transport
  - Aerosol behaviour inside the containment
  - Deposition of aerosols
  - Revolatisation of aerosols
  - Effect of energetic phenomena on in-containment fission product behaviour
- Radionuclide release from the containment (Source Term)

It is obvious that both the accident progression and containment performance analysis and source term analysis are strongly interrelated. For example, the timing and the containment failure mode are important factors for the release from the containment to the environment, or the deposition of radionuclides in steam generator tubes or the pressuriser surge line may create an extra local heat spot which might fail the RCS at that particular location.

3.1.1 In-vessel phenomena

3.1.1.1 Plant damage states, start of core uncovery

The interface between the level 1 systems (core damage) analysis and the level 2 containment (release) analysis consists of a set of defined plant damage states (PDS). These PDS are defined by a set of functional characteristics for plant conditions and system operation which are important to accident progression, containment failure, and source terms. Each PDS contains level 1 sequences with sufficient similarity of plant conditions and system functional characteristics that the containment accident progression for all sequences in that PDS can be considered identical. Each PDS defines a unique set of conditions regarding the state of the plant and containment building systems and the physical state of the core, reactor vessel, primary coolant system and containment boundary at (approximately) the time of core damage. All level 1/level 2 interfaces characteristics, for example, reactor coolant system and containment thermal-hydraulic conditions at the onset of core damage, for example, high or low RCS pressure or different timing: slow or fast progressing sequences, containment mitigation system's availability, support system's availability, possibilities and assumptions for human actions need to be incorporated. The defined PDSs must make possible the evaluation of potential preventive and mitigative measures.

A special category of PDS characteristics is given by containment bypass situations, for example, interfacing systems LOCA (IS-LOCA), steam generator tube rupture (SGTR), LOCA plus isolation failure, containment failed by the initiating event itself, etc.

3.1.1.2 In-Vessel Core Degradation

3.1.1.2.1 Heat-up phase

Following an initiating event, coolant loss from the primary system occurs. As the coolant level drops below the top of the core, the fission product decay heat generated in the fuel rods and the degraded cooling cause the core to heat up. In a PWR, during the heatup stage, natural circulation in the uncovered part of the core and in the upper- plenum region may transfer a significant fraction of the heat from the core to the upper-plenum structure and to the primary system boundary[2]. During the later stages of core

[2] In a BWR the zircalloy channel boxes, which surround each fuel rod bundle, prevent crossflow to occur. This prevents any upper-plenum natural circulation loop to penetrate the core.

heatup, radiation becomes important and ultimately becomes a dominant heat transfer mechanism. Both natural circulation and radiation heat transfer, reduce the rate of core heatup. Natural circulation and the resultant heat transfer between the uncovered region of the intact core (before collapse) and the upper plenum and even the steam generator can have a significant effect upon in-vessel severe accident behaviour. Experiments have shown that steam and hydrogen convection velocities in the intact core geometry are about an order of magnitude greater than the steam boiloff velocities. In-vessel natural circulation produces more uniform core temperatures and also transfers more of the core heat to the upper-plenum structure and walls, the hot leg nozzles, and even, by countercurrent flow in the hot leg, to the steam generator tubes.

3.1.1.2.2 Oxidation/hydrogen generation

As the temperature increases above 1500 K, oxidation of the zircalloy cladding by steam becomes an important heat source. The reaction produces heat comparable with that of the fission product’s decay heat, and it generates hydrogen. As the temperature increases, the rate of oxidation increases rapidly. As the temperature exceeds about 1855 K (phase transformation of ZrO_2), rapid autocatalytic oxidation occurs, except in those local places, where a significant part of the steam flow is replaced by hydrogen (steam starvation). The formed ZrO_2 surface layer that builds up on the surface of the zircalloy cladding limits the oxidation rate. A 'burn front' may develop in the upper regions of the core at the steam-starvation boundary and may move downward as autocatalytic oxidation progresses. Except during high-pressure sequences, ballooning of the cladding may occur as the heatup proceeds. This significantly reduces the clad-fuel heat transfer and can increase the rate of cladding oxidation and hydrogen generation. When the 'balloon' cracks open. As the steam can now attack the inner side of the cladding, the surface for steam-cladding interactions is now doubled. Experiments have shown that cladding ballooning does not block the steam flow, but it might significantly divert steam flow in the open-lattice PWR core and might reduce natural circulation through the core. Fuel temperature may increase by tens of degrees K per second during the autocatalytic oxidation of the fuel rod cladding, and much of the hydrogen generation in the accident occurs during this early phase.

During the heatup process, the first failures in the core typically occur in the control rods. For PWR silver-indium-cadmium control rods, failure occurs near the 1723 K melting point of the stainless steel control rod cladding. The cadmium rapidly vaporises at rod failure and condenses into an aerosol when cooled outside the core (melting point of Ag-In-Cd alloy is 1073 K). The molten silver and indium relocate downward with no interaction with the stainless steel control rod guide tubes, eventually to freeze in the colder regions of the core. In case it falls into the water of the lower plenum, additional steam is produced, which may temporarily overcome steam starvation.

3.1.1.2.3 Chemical interactions (e.g., eutectics) amongst core material

The molten stainless steel interacts strongly with zircalloy and inconel (rod spacer grids) to form eutectics at about 1500 K. The Al_2O_3 in the zircalloy clad and poison rods forms eutectics with zircalloy at 1750 K and with both ZrO_2 and UO_2 at about 2200 K. For BWRs eutectics occur between B_4C and stainless steel at temperatures as low as 1523 K, liquefying the stainless steel cladding of the control blades, with subsequent relocation and possible blockage formation by the liquefied material.

3.1.1.2.4 Cladding failure

The fuel rods normally fail when molten unoxidised metallic zircalloy fails the ZrO_2 surface layer produced by oxidation of the cladding. The molten metallic zircalloy then relocates downward along the individual rods in a 'candling' process. This process reduces the supply of metallic zircalloy for oxidation from the high-temperature region of the core where oxidation can occur, effectively limiting the rapid temperature rise and the rapid hydrogen generation from autocatalytic oxidation of the initially intact fuel rods. This relocation of the molten unoxidised metallic zircalloy is the first of three significant and distinct material relocation processes that occur during in-vessel core melt progression.

3.1.1.2.5 Relocation and blockage formation

Near its 2100 K melting point, molten metallic zirconium can dissolve up to 10% of its mass of solid UO_2, up to 20 mass % near the 2700 K liquid monotectic point and over 80% above this temperature. This 'liquefied' fuel relocates downward and freezes on colder portions of the fuel rods and rod spacer grids. As water boils of and core melt progression proceeds, this solidified material may remelt and relocate downward again in a repetitive process. This process was responsible in the TMI-2 accident for the formation of the tough 'hard pan' across the mid-region of the core.

After the initial autocatalytic oxidation transient and relocation of the molten metallic zircalloy and dissolved UO_2, free standing columns of declad, stacked, cracked ceramic (UO_2, ZrO_2) fuel pellets in essentially the original rod geometry remain.

3.1.1.2.6 Core collapse

The collapse of the ceramic-pellet columns is the second major material relocation process involved in core melt progression. This collapse forms a rubble bed on top of the layer of frozen relocated zircalloy and liquefied fuel and substantially changes the thermal characteristics of the debris, including its flow resistance. The flow circulation flow from the upper plenum to the damaged core is virtually interrupted by this collapse.

3.1.1.2.7 Late phase

As steam boiloff continues, the debris region, which consists of frozen relocated zircalloy and liquefied fuel in the fuel rod stubs at the bottom and mostly ceramic particulate rubble above, is heated by fission product decay and probably by some continued oxidation of the relocated zircalloy. Because of the surface heat removal, melting starts near the centre of the debris region, and increasing loads are imposed upon the lower crust and the core support structure.

The third major material relocation comes with failure of the lower support crust, or the core support plate, with slumping of the corium melt into the lower plenum and quenching of the surface of the melt mass by the lower plenum water. The slumping might occur instantaneously or more gradually by pouring molten fuel in the lower plenum of the RPV. During the quenching process, large quantities of steam are generated causing a pressure spike. Oxidation of the molten unoxidised zircalloy can generate considerable additional hydrogen. A steam explosion might occur when the corium mass slumps into the lower-plenum water in lower-pressure melt sequences (The higher the pressure, the lower the probability of a steam explosion). Following quenching of the melt surface, which occurs relatively rapidly because of the low thermal conductivity of the ceramic corium, the melt boils dry the lower plenum, re-heats, attacks the vessel lower head and its penetrations, and finally may breach the lower head. The rate of ejection of the melt and solid debris into the reactor cavity is dependent upon the mode of vessel failure (high or low pressure, via instrument-line nozzles in the lower head, sudden total failure of the lower circumferential weld, etc.).

3.1.1.2.8 Natural circulation leading to RCS-failure prior RPV-failure.

Natural circulation in PWRs during a severe accident may be an important mechanism for failure of the pressure boundary of the reactor coolant system. There are three potential natural flows in the RCS: in-vessel circulation, hot leg countercurrent flow including flow to the steam generator tubes, and flow through the coolant loops. The heat removed from the core via convective flow will heat up structures in the RCS and may result in failure of the pressure boundary. Such failures could occur prior to the breach of the RPV lower head by the core debris. If failure of the RCS pressure boundary is sufficiently early, the RCS could be depressurised to a level at which high pressure melt ejection would not occur.

3.1.1.2.9 Refill/Quench

The presence of water and/or the injection of water have an important effect on the in-vessel core melt progression. Right above the quench front, saturated steam generated by the heat input to the quench water rapidly cools down the embrittled, oxidised surfaces of the cladding. Thermal stress may lead to the cracking or fragmentation of embrittled cladding surfaces and to the formation of new, exposed metallic surfaces. This leads to an extended zircalloy oxidation by steam with localised melt formation, subsequent relocation, renewed rapid temperature rise and sharply increased hydrogen generation. It may also destroy, by thermal shock (quench induced scattering) parts of the core and extend the debris bed formation.

3.1.1.3 Vessel attack, RPV failure mode and core debris release to the containment.

The design of the lower reactor vessel region strongly influences core debris interactions and the mode of the lower head failure.

3.1.1.3.1 Energetic in-vessel steam explosions

Core debris relocation to the lower vessel may involve a coherent mixing of a large fraction of the core inventory with the residual water in the lower plenum, and creating the potential for large in-vessel steam explosions. For this to occur the melt has first to be fragmented. A layer of steam forms around the particles, reducing heat transfer. Already a weak shock can trigger the disruption of this state. During a few milliseconds a very high heat transfer rate from the fuel to the surrounding water occurs, causing an explosive phase transition from water to steam. As the growing shock wave moves through the system it strips away the steam layer and further fragments the melt. The greatly increased heat transfer will amplify the shock wave further. An energetic steam explosion can deliver significant impulsive shock loads, possibly failing the vessel lower head, and can significantly redistribute the core debris. It is even possible that a missile generated by an energetic in-vessel steam explosion will fail the containment (α-mode). However, a rate-limited relocation of the debris is more likely. In that case, steam explosion energetics (if any) are likely to be relatively minor. In the absence of a steam explosion, a slower relocation may increase the extent of oxidation and of steam generation. The result would be pressurisation of the primary system. In references 14 & 19, in-vessel steam explosions and the potential for consequential containment failure (α-mode failure) is described in detail. In NUREG-1150 the mean α-mode failure probabilities conditional on the occurrence of core meltdown was estimated between 10^{-4} and 10^{-3} for PWRs and below 10^{-3} for BWRs. A reassessment performed in the recently published reference 39. suggests that these figures are pessimistic estimates.

3.1.1.3.2 High pressure melt ejection

Both PWRs and BWRs may experience core damage sequences with the primary system being pressurised. When the bottom head of the reactor vessel is breached in such accident sequences, the core melt will be forcibly ejected. The ejected materials are likely to be dispersed out of the reactor cavity into surrounding containment volumes as small particles, quickly transferring thermal energy to the containment atmosphere. In addition, metallic components of the sprayed core debris, mostly zirconium and steel, can react with oxygen and steam in the atmosphere releasing a large quantity of chemical energy that further heats and pressurises the containment. The term 'direct containment heating' (DCH) is used as a summary description of the involved physical and chemical processes.

3.1.1.3.3 Jet Impingement

Ablation due to jet impingement is a potential cause of vessel failure. The erosion of steel structures by a high temperature jet is characterised by a rapid ablation rate near the point of impingement. The ablation rate would be considerably reduced by the formation of a crust layer of urania. Due to the presence of a large number of penetrations in some reactor designs and the potential for jet break-up in the water pool, it is unlikely that a molten jet will directly attack the lower vessel head. However, penetration tubes may fail if hit by the jet. Ablation due to direct jet impingement is a potential cause of early reactor vessel failure.

3.1.1.3.4 Plugging and failure of lower head penetrations

A large number of penetrations exists in the lower head of all BWRs and some PWRs. Erosion of the penetration tubes could allow molten material to flow down the tubes and refreeze to form a crust along the tube wall. If the temperature of the core debris is high enough, melting or creep rupture of the tube walls could occur.

3.1.1.3.5 Ejection of a lower head penetration

Core melt attack on a penetration tube and the sustained heating from accumulated debris may cause tube penetration weld failure. If under high system pressure tube ejection may result. However, the binding stress caused by differences in the thermal expansion coefficients of carbon steel (lower head) and inconel or stainless steel (penetration nozzle) may continue to hold the tube in the hole thus preventing ejection. The larger the temperature difference the more unlikely will tube ejection be. Once the penetration tube is ejected, the high temperature melt will ablate the hole to a much larger diameter.

3.1.1.3.6 Global Creep Rupture

In a PWR with no penetration tubes in the lower head, a direct contact between the core debris and the lower head wall will cause substantial heating of the lower head. The heating, in conjunction with the stress induced by elevated system pressure and/or the weight of the core debris, may lead to lower head failure by global creep rupture. Depending on the debris configuration and coolability, the average vessel wall temperature rise is likely to be relatively slow. The time to vessel failure depends on the system pressure, vessel wall thickness, decay heat of the core debris, and the contact between core debris and vessel wall. A depressurised reactor vessel should reduce the potential for creep rupture-induced vessel failure.

In the following table, the important phases of in-vessel accident progression, the related key issues, phenomena and physical processes and important design characteristics are compiled.

Table 3.1.1.3-1

In-vessel Accident Progression Phases & important parameters	Related key issues, phenomena and physical parameters	Relevant design characteristics
Core heat-up and degradation	level 1 accident sequence (PDS) - timing - pressure of RCS - temp of RCS - amount of water in RCS, - rate of heat removal; via break flow or primary relief valves	- RPV, RCS and core design - Feed and Bleed capability still available, although insufficient. - Reactor Power/RCS volume ratio. - Potential for ex-vessel cooling of the core via flooding of the cavity
In-vessel thermal-hydraulics	- Idem as above - Recirculation flows - Blockage formation - Steam starvation	- RPV and core design (e.g., geometry)
Hydrogen production	- Blockage formation - Cladding ballooning - Refill/quench	- Amount of Zr and stainless steel in core
RCS thermal-hydraulics	- Recirculation flows; hot leg, surge line and SG heat up. - Water level in RPV	- RCS design - Reactor Power/RCS Volume
Core loss of Geometry	- Core heat-up rate - In-vessel and RCS thermal-hydraulics - Potential for energetics - Refill	- Core design (e.g., composition of materials, or geometry)
In-vessel fuel-coolant interactions (energetic and non-energetic)	- Refill - Recriticality - Potential for energetics (e.g., RCS pressure and amount of water in bottom head)	- Lower plenum design
RCS failure mechanism/ vessel attack	- Stratification of melt - Molten pool heat transfer - Potential for global failure of bottom head by creep rupture - Potential for energetics - Recirculation flows (e.g., failure of SG tubes or pressurizer surge line) - Local failures of bottom head	- RPV and RCS design (especially, bottom head design, whether or not nozzles in bottom head)

3.1.1.4 In-vessel radionuclide release and transport.

3.1.1.4.1 Iodine and Caesium chemistry

During normal operation fission products migrate from the fuel pellets to the free spaces of the pins (primarily noble gases, iodides and Cs). When the cladding fails due to high temperature in an accident situation, the fission products are released from the gap. This so-called 'gap-release' is important when core cooling is recovered before melt. In severe accidents, this release is small compared with the subsequent 'melt release'. As the fuel heats up to its melting point the noble gases and the more volatile fission products will be released.

During the heatup, degradation, and meltdown (relocation and slump) phases more volatile fission products will be released from the fuel. The most important factor for this release is the maximum temperature reached in the fuel and the time the fuel remains at that temperature. A second factor is the composition and rate of steam/hydrogen mixing in the fuel matrix. The ratio of steam (an oxidant) and hydrogen (a reductant) governs the effective oxidation potential, which in turn can alter the chemical forms of the released species. The steam partial pressure will also impact the volatility of some material.

The important volatile fission products are iodine, caesium and tellurium. Since there is typically 10 times more Cs than iodine, usually all of the iodine is released as CsI and the rest of the Cs as CsOH. CsI is thermodynamically stable up to at least 2000 ^{o}C in the system UO_2 - zircalloy - steam. The remainder of the Cs of the inventory would be released from the fuel mainly in elemental form and could then react with steam to form very stable CsOH gaseous molecules. Above 2000 ^{o}C, CsI will react with steam to form CsOH + HI. There is also strong experimental evidence that CsI reacts with boric acid. Boric acid is present in the coolant and emergency cooling water, or is produced by the decomposition of B_4C control rods. If significant reduction occurs in the coolant system, HI and caesium borates will be produced. Both CsOH and HI are more volatile than CsI. HI is also chemically reactive, it would interact with RCS surfaces and aerosols. In reference 31. an extended discussion is given on iodine chemistry under severe accident conditions.

The caesium source term might be attenuated in the RCS by any reaction of both CsI and CsOH with boric acid, because these give less volatile caesium borates (e.g., CsI + HBO_2 -> $CsBO_2$ + HI). CsOH also interacts with steel, diffusing into the inner chromium oxide layer, providing further potential attenuation.

3.1.1.4.2 Other isotope chemistry

Te exists in the fuel as Cs_2Te. These molecules react with steam/hydrogen to form, dependent on the hydrogen concentration, TeO_2, TeO, elemental Te and H_2Te. Elemental Te is more volatile than Cs_2Te. Tellurium released as the element can react with Zr and Sn from the zircalloy, producing zirconium and tellurides, and being retained in the core debris up to ca 2000 ^{o}C. Removal of Zr either through oxidation, or through reactions with UO_2 fuel, can increase the release of Te. In contrast with ZrO_2, Zr reacts with Te quite easily. If Te is released in elemental form, it would probably react rapidly with metal aerosols and surfaces to form less volatile metal tellurides. Vaporisation of structural and control rod material also can contribute to fission product release, as this process can make up the bulk of the aerosols which subsequently carry the fission products.

Barium and strontium oxides in the fuel matrix could react with unoxidised zircalloy cladding to release volatile elemental Ba and Sr to the RCS relatively early. These would form hydroxides on contact with steam, which may exhibit similar behaviour as CsOH, e.g., reaction with boric acid and steels.

In the event of HPME an additional release may occur: oxidation of fission products by oxygen or steam in the containment atmosphere can lead to the so called oxidation release. Ruthenium is very susceptible to this release, because although most of its chemical forms are quite refractory, the oxide RuO_4 is much more volatile.

3.1.1.4.3 Retention and deposition of fission products inside the RCS

Following release from fuel, fission products are carried along with steam and hydrogen, both as vapours and as aerosols, or dissolved in any water retained in the circuit. Fission product vapours can condense on colder surfaces, as well as on other aerosol particles during their passage through the reactor coolant system to the containment. Fission product aerosols can agglomerate with other radioactive and non-radioactive (for example, inert structural aerosols) to form larger particles which can in turn settle on structural surfaces.

Chemical interactions between fission product vapours/aerosols and metallic surfaces might lead to heatup of structural surfaces (due to decay heat content of deposits) beyond the temperature required for re-vaporisation of previously deposited, chemically unbound volatile fission products. Also mechanisms like re-entrainment may cause re-volatilisation of previously deposited fission products.

In the primary system, the flow of liquid films on walls may be a major mechanism for the mass transport of deposited liquid droplets, and their subsequent re-injection into the containment atmosphere at the end of a broken pipe. Similarly, the transport of a large fraction of the solid particles deposited in the primary system may occur in the form of creep flow of the particles on the piping surfaces or saltation of the particles close to the surface.

In the following table, the important phases of in-vessel fission product release and transport, the related key issues, phenomena and physical processes and important design characteristics are compiled.

Table 3.1.1.4-2

In-vessel release and transport of fission product issues & important parameters	Related key issues, phenomena and key physical parameters	Relevant design characteristics
In-vessel release of fission products	- temperature of degraded core/debris/melt - chemical reactions with other fission products and core materials - time after scram - burn-up of fuel - Refill/quench	- maximum burn-up of fuel - enrichment of fuel - amount of fuel - Reactor power - Amount and composition of absorber material in control rods
RCS fission product transport	- Recirculation flows - Break flow or flow via primary relief valves - Potential for containment bypass (IS-LOCA, SGTR, etc.)	- RCS design - Size and location of break - Potential to restore isolation - Potential to keep water in secondary side of failed SGs. - Potential to depressurise RCS

	- Suspension, resuspension, agglomeration, plate-out, re-volatisation, etc. of fission products. - Fuel-coolant interactions - Refill/quench - Debris bed dryout/rewet - RCS failure prior RPV failure via induced SGTR or surge line failure. - Amount of water /steam	
Release of fission products at vessel breach into containment	- Timing of release - Mode/mechanism of RPV failure - Potential for energetics (HPME/DCH) - Potential for containment bypass (IS-LOCA, SGTR, etc.)	- RCS design - RPV design - Potential for ex-vessel scrubbing (e.g., suppression pool)

3.1.2 Ex-Vessel Phenomena

3.1.2.1 Energetic ex-vessel phenomena immediately or shortly after vessel breach.

3.1.2.1.1 High Pressure Melt Ejection (HPME)

The ejection of melt and/or solid debris under high RCS pressure at the time of RPV is called high pressure melt ejection (HPME).

3.1.2.1.2 Direct Containment Heating

Analyses of the containment heat balance indicates that even a large, dry containment of a PWR plant can be pressurised beyond its ultimate strength if a significant fraction of the core materials participates in DCH. The peak containment pressure is normally reached within seconds after the melt ejection. A large amount of aerosols, including refractory fission products, could be generated in a high-pressure melt ejection. If the containment should fail from the DCH loading, a massive release of radioactive materials could result. Dispersed core debris can induce further hazards: If hydrogen is present in the containment atmosphere, dispersed hot debris particles could serve as catalyst to promote recombination of hydrogen with free oxygen even when the H_2 concentration is below the conventional flammable limit. Hydrogen recombination will release additional energy to raise the pressure and the temperature in the containment.

Experiments predict that the metallic components in the melt will be completely oxidised by steam in the reactor-cavity region during high-pressure melt ejection. Such reactions would generate a large quantity of hydrogen that can readily mix in the containment atmosphere regardless of debris transport. Burning of this hydrogen could challenge the containment integrity.

Other hazards associated with DCH are the effects of high temperature on containment structure and equipment, and possible missile generation. .

A recent state-of-the-art report is provided in reference 40.

Because of the risks of DCH, a possible and often applied strategy for avoiding containment failure due to DCH is to deliberately depressurise the primary circuit prior to vessel failure. On the down side of this strategy is that lowering the pressure in the reactor coolant system may increase the likelihood of in-vessel steam explosions. However, as the conditional probability for containment failure due to steam explosion, given core melt, is in the range 10^{-3} to 10^{-2} for PWRs , and below 10^{-3} for BWRs, the benefits appear to outweigh the drawbacks (compare section 3.1.1.3.1).

3.1.2.1.3 Ex-vessel steam explosions

If the reactor cavity is filled with water at the time of high pressure melt ejection, a steam explosion may occur that could contribute to further debris fragmentation and dispersion, at the same time generating dynamic loading of the containment. Some of the factors contributing to the in-vessel steam explosions are also of importance for the ex-vessel steam explosions, to mention: the amount of water available to participate, the composition of the melt, including the amount of unoxidised metals that may react during the explosion, cavity or pedestal region geometry, as far as it may lead to confinement of shock waves through a water pool, pouring rate and contact mode, i.e., water on corium, corium on water, or jet ejection in water, and fraction of the core participating. On the other hand, some of the initial conditions are different from those for in-vessel steam explosions. Firstly, ex-vessel steam explosions will always be at low pressure, no higher than the containment failure pressure. Steam explosions tend to be more likely at low pressure. Second, the geometry is different, involving varying degrees of confinement. Third, there are three contact modes to consider. The corium may pour from the vessel into a water pool or water may be added to flood the molten corium, or the corium may be ejected from the vessel as a high pressure jet into a water pool. Experiments indicate that steam explosions are likely under these conditions, but with magnitudes that are very likely too small to fail a reactor containment (see section 3.3.4.2). Also here the fragmentation of the corium is an important factor for the coolability of the debris bed. A fine fragmentation as a result of a steam explosion or high pressure ejection may lead to a non coolable debris bed which will interact with the concrete. However, another and even more likely possibility is that the debris will be dispersed in the containment by the steam explosion, reducing the possibility of attack by corium of the concrete.

3.1.2.1.4 Missiles due to energetic RPV failure

For some reactor types another high pressure core melt scenario may be important. This is the so called 'reactor vessel launch' (rocket) scenario (reference 12.). In this case the reactor pressure vessel fails catastrophically during a high pressure core melt scenario prior to vessel melt-through, because of a sudden and complete failure of the lower circumferential welding. The melt heats up the inner surface of the RPV. After > 2000 seconds, high temperatures may the lead to plastic deformation of the material. High pressure may then lead to a sudden rupture of the lower welding. In case of a small cavity, the sudden release of steam and ejected corium will generate high upward forces on the RPV. In case of pressures > 3 MPa (30 bar), the anchoring of the RPV on the pedestal as well as the anchoring by the connected piping of the primary loop may fail. At pressures > 8 - 10 MPa the launched upper part of the RPV may fail the containment.

Also in-vessel steam explosions can generate an upward moving slug of water and molten fuel, which lifts the upper head of the vessel. The reactor vessel head then acts as a missile that perforates any structures above the vessel and, ultimately, could penetrate the containment building. According to reference 39., this scenario is highly unlikely.

3.1.2.1.5 Hydrogen burns, including deflagration and detonation

During core melt accidents, zircalloy as well as other in core metallic materials react at high temperatures with water or steam. Consequently, large amounts of hydrogen are produced, e.g., 880 kg of hydrogen is produced if the total Zr mass of a 1000 MW_e PWR is oxidised (= ca. 20000 kg), or ca 2800 kg hydrogen is produced in case total oxidation of the zircalloy in a 1000 MW_e BWR-core (65000 kg Zr). In the in-vessel stage of the accident ca. 20 - 80 percent of the zircalloy may be oxidised, and be released to the containment when the primary circuit fails. This may occur gradually if the primary loop has failed prior to the complete meltdown of the core, or suddenly at vessel rupture. The rest of the zircalloy will be oxidised during the corium-concrete interactions (concrete contains ca. 6.5 % water). Also in case the melting process is arrested in the RPV due to reflooding of the core, large amounts of hydrogen will be produced.

Regarding ignition of hydrogen three different rates of combustion are to be distinguished: local burning by diffusion flames, deflagration and detonation. Deflagration is a form of combustion in which the flame moves at subsonic speed relative to the unburned gas. Unburned gas is heated to reaction temperature by thermal conduction and mass diffusion from the hot burned gas. Local burning as well as deflagration may cause static or quasi-static pressure loads on the containment due to the extra heating of the containment atmosphere. Hydrogen detonations involve the reaction of hydrogen through supersonic propagation of a burning zone or combustion wave. The pressure loads developed are essentially dynamic loads. Detonations may threaten the integrity of the containment and of important safety-related equipment due to the dynamic pressure loads.

The mode of combustion primarily depends on the concentration of hydrogen, steam and other gases like CO or CO_2. Also the initial temperature and pressure of the gas are important parameters. In some cases transition from deflagration to detonation can occur. Obstacles in a confined area or flow turbulence can cause acceleration of the flame front. Also in long tunnel-like structures deflagration-detonation transitions may occur. In references 27., 28. & 29. a more detailed description is given of the hydrogen issue; especially of the deflagration to detonation transition.

In the following table, key issues, phenomena, physical processes and important design characteristics relevant for the early ex-vessel energetic accident progression phase are compiled.

Table 3.1.2.1-1

Early Ex-vessel Energetic Accident Progression Phases & Important Parameters	Related key issues, phenomena and physical parameters	Relevant design characteristics
Vessel lift-off	- Potential for global bottom head failure - Potential for energetic ex-vessel fuel-coolant interactions - Pressure of RCS at RPV failure	- Cavity geometry - RPV bottom head design - RCS depressurisation capability

Table 3.1.2.1-1 (cont'd)

Debris ejection from vessel/ high pressure melt ejection (HPME)	- Failure mode of RPV - Pressure of RCS at RPV failure - Debris trapping & transport - Inducing ex-vessel steam explosions	- Depressurisation capability - Cavity design - RPV bottom head design
Direct containment heating	- Potential for HPME - Zr oxidation/hydrogen generation - Hydrogen combustion/ recombination	- Depressurisation capability - Cavity design under-vessel pathways to containment. - RPV failure size and location - Free volume of containment - Design pressure of containment
Ex-vessel fuel-coolant interactions	- Potential for steam explosions - Steam spikes - Ex-vessel debris bed coolability (fragmentation of debris) - Capability to fail structures inside containment (e.g., vessel support)	- Mass of debris - Geometry of cavity - Potential for flooding the cavity prior RPV failure and amount of water

3.1.2.2 Ex-vessel phenomena in the long term phase.

3.1.2.2.1 Corium concrete interactions

After core melt/debris has fallen on the concrete at the bottom of the cavity in a uncoolable configuration, it will begin to transfer its heat to the concrete. As the concrete heats up it will begin to disintegrate and decompose chemically.

At high temperatures (approximately 1300 - 1500 °C), concrete decomposes; the erosion products typically include water, CO_2, and refractory oxides such as Sio_{ux} and CAE. The liquefied oxidic components of the concrete mix with the uranium oxide fuel and the metallic oxides of the debris. Typically, gases released at the debris-concrete interface bubble through the debris pool. Some of the gaseous components, e.g., water vapour and carbon dioxide, may react chemically with the debris while others escape from the pool surface and enter the containment atmosphere directly. Some of the gases (H_2 and CO) are combustible and can contribute to containment loading. As the bubbles break up at the surface, aerosols are formed due to vapour condensation and film rupture. These aerosols contain non-radioactive components as well as radioactive fission products that contribute to the radiological source term. The volatility of the fission products carried away by the produced gases depends on the chemical conditions in the melt. These conditions may be different in different places of the melt: the melt may be stratified into an oxidic layer, with oxidising conditions, and a metallic layer, in which conditions are reducing. The oxidic layer may consist largely of uranium oxide, and would therefore be denser than the metallic layer and lie underneath it. But as concrete decomposition proceeds, this density of the oxidic layer may decrease as the concentrations of calcium and silicon oxides increases, and the arrangement of the layers may reverse.

This reversal will also change the rate of exothermic oxidation reactions between metallic compounds and the water vapour originating from the decomposed concrete.

Alternatively, the passage of the gas through the melt may act to mix the oxidic and metallic components, producing more uniform chemical conditions through the melt. In addition to fission product volatilisation, the corium-concrete interaction may produce considerable quantities of non-radioactive aerosols, which may influence fission product transport at later times.

After the corium-concrete interaction has begun, water may find its way into the cavity, or it may be introduced there deliberately in an attempt to cool the debris and stop the interaction. Whether or not the debris is coolable depends strongly on the heat transfer through the pool and into the water layer, and therefore depends on the depth of the pool/debris. Also, a partly insulating crust may be formed on top of the debris, preventing cooling until break up at later times.

3.1.2.2.2 Direct melt attack on other structures inside the containment

Apart from the direct attack of the basemat, which may result in basemat melt-through, there are also other direct attacks on structures and engineered safety features inside the containment that may result in even larger release pathways into the environment (e.g., drywell liner or reactor pedestal). Also the convective and radiative heat from the pool surface contribute to containment overpressurisation, as well as the produced gases by the decomposition of the concrete have their contribution.

3.1.2.2.3 Late over-pressurisation

The range of pressure loads resulting from severe accident conditions can roughly be categorised in two qualitatively distinct areas. These are:

- Gradual pressure rises. Gradual pressurisation of the containment building would result from the protracted generation of steam and non-condensable gases through the interaction of molten core material with the concrete floor beneath the reactor vessel. This pressurisation could last from several hours to several days, depending upon accident-specific factors such as the availability of water in the containment and the operability of engineered safety features. An additional mechanism for gradual pressurisation in BWR pressure-suppression containments is the generation of steam from the suppression pool in the event that pool heat removal capability is degraded.
- Rapid pressure rises. The high-pressure melt ejection from the vessel, the deflagration of combustible gases, and the rapid generation of steam through the interaction of molten fuel with water in the containment are phenomena that could lead to significant pressure rises in the containment within a few seconds. Such pressure rises may be viewed as rapid in a thermodynamic context; however, from a structural perspective, they are, apart from global hydrogen detonations, effectively static or quasi-static. It is essential, nevertheless, to distinguish between gradual and rapid pressure rise, since the rate of pressure increase may have significant influence on the timing and mode of containment failure.

In the following table, key issues, phenomena, physical processes and important design characteristics relevant for the long term ex-vessel accident progression phase are compiled.

Table 3.1.2.2-1

Long term Ex-vessel Accident Progression Phases & Important Parameters	Related key issues, phenomena and physical parameters	Relevant design characteristics
Core-concrete interactions	- Production of non-condensable gases - Hydrogen production - Basemat melt-through - Potential interactions with RPV supports and thereby failing the containment - Stratification of melt	- Cavity geometry - Chemical composition of concrete - Containment design
Ex-vessel debris quenching	- Idem as above - Basemat melt-through - Steam pressurisation - Debris spreading and potential interactions with containment shell	- Cavity geometry - Potential for flooding the cavity - Containment design
Containment thermal hydraulics	- Containment pressure - Hydrogen mixing/stratification - Late Hydrogen combustion - Effect of ESNs (e.g., potential for suppression pool cooling, containment spray, fan coolers on containment pressure and temperature) - Composition of containment atmosphere - Heat conduction through containment walls - Circulation flows	- Containment design - Containment geometry - Reactor Power/ containment free volume ratio. - Fuel and Zr mass/ containment free volume ratio - Pressure suppression/relief capability (suppression pools, containment sprays, fan coolers) - Capability for containment venting/ availability of venting procedures in EOPs. - Potential for pressure suppression bypass
Hydrogen combustion	- Mixing/stratification - Detonation - Deflagration/detonation transition - Deflagration - Pressurisation loads	- Containment geometry and design - Potential for Hydrogen control - Zr mass/ containment free volume ratio.

3.1.2.3 Containment loads and containment failure modes.

The timing and the way in which a containment fails is important to the consequences of an accident. Early failure or bypass of the containment could result in a large release of fission products to the environment. Late containment failure occurs more than a few hours after the start of core damage. Thus, particulate and aerosol removal mechanisms can greatly reduce the concentration of fission products in the containment atmosphere and the magnitude of the release. The magnitude of the fission product release is also determined by the size and location of the break and the pressure in the containment. All containments are potentially susceptible to some design specific form of early failure, as well as late failure.

Containment design criteria are based on a set of deterministically selected load scenarios. Pressure and temperature challenges are usually based on the design basis LOCA. External events such as earthquakes, floods, high winds or aircraft crash are in some cases considered as well.

Assessments of beyond design accidents show that in some cases significant containment loading can occur, reaching or even exceeding the design loads. None of the design basis accident scenarios involve rapidly increasing containment loads. Therefore, loads like fluid jet impingement, direct containment heating, rapid deflagration or detonation of hydrogen pockets attending severe core degradation accidents, may pose significant threats to containment integrity.

The failure mode of the containment is a crucial factor for the off-site consequences. Potential failure modes of the containment that have been identified in PSA studies can to be grouped under the following headings:

3.1.2.3.1 Containment Bypass

- Interfacing-systems LOCA
- Failure to isolate containment
- Steam generator tube rupture
- Containment function failed before or at the onset of core degradation, e.g., failure by an external initiating event

3.1.2.3.2 Early Containment failures

- Overpressurisation and temperature loads
  - due to non-condensable gases and steam
  - due to hydrogen burn
  - due to direct containment heating
- Missiles or pressure loads
  - due to steam explosions
- Attack by core melt of containment structures
  - due to direct contact between core debris and containment (e.g., drywell liner meltthrough scenario in Mark I containment)

- Vessel thrust force
  - due to blowdown at high pressure

3.1.2.3.3 Late Containment failures

- Overpressurisation and temperature loads
  - due to non-condensable gases and steam
  - due to hydrogen burn
- Meltthrough
  - due to basemat penetration by core debris
- Vessel structural support failure
  - due to core debris erosion

3.1.2.4 Ex-vessel radionuclide release and transport

As the gases produced in the interaction between the molten core debris and the concrete bubble up through the melt, they carry fission product vapours and aerosols with them. The volatility of the fission products carried away by the produced gases depends on the chemical conditions in the melt. These conditions may be different in different places of the melt: the melt may be stratified into an oxidic layer, with oxidising conditions, and a metallic layer, in which conditions are reducing. The oxidic layer may consist largely of uranium oxide, and would therefore be denser than the metallic layer and lie underneath it. But as concrete decomposition proceeds, this density of the oxidic layer may decrease as the concentrations of calcium and silicon oxides increases, and the arrangement of the layers may reverse. This reversal will also change the rate of exothermic oxidation reactions between metallic compounds and the water vapour originating from the decomposed concrete.

Alternatively, the passage of the gas through the melt may act to mix the oxidic and metallic components, producing more uniform chemical conditions through the melt. In addition to fission product volatilisation, the corium-concrete interaction may produce considerable quantities of non-radioactive aerosols, which may influence fission product transport at later times. If there is a pool of water overlying the melt, bubble scrubbing may act to reduce the release of fission products and aerosols to the containment.

When the molten core debris contacts the concrete basemat, the high temperature causes thermal decomposition of the concrete. Large volumes of carbon dioxide and steam are released. As these gases sparge through the overlying layer of molten material they are reduced by the metallic constituents to hydrogen and CO. Some fission products (such as Al_2O_3) may be reduced to more volatile sub-oxides (Lao) or metals at this point. This would enhance their release. As the gases passes through the melt they pick up materials that are vaporised at the elevated temperatures (above 2270 ^{o}K). The gases leave the surface of the melt where the mix with vapour and are transported into the containment as aerosols.

3.1.2.4.1 Aerosol Behaviour inside containment

Volatile fission products, such as Cs, I, and Te, which are volatile at the high temperatures of a degraded core either condense to form liquid or solid particles or combine chemically to form lower volatile species which then condense. The non-volatile fission products, together with non-radioactive material, can be released during the hot corium-concrete interaction. These then become solid aerosols or absorb water vapour to become liquid aerosols. The behaviour of aerosols governs the rate of deposition of fission products in the containment (references 1., 2. & 6.).

An aerosol is a collection of particles suspended in a gas. The particles may be liquid, solid, or a mixture. Their diameters can range from as small as 1 μm to as large as 1 mm. Liquid aerosol particles can be taken to be spheres owing to the action of surface tension and their small size. Solid aerosol particles may have a variety of shapes, some departing very significantly from spherical. However, these odd shaped particles have a reduced mobility and thereby a decreased *decomposition* rate, compared with a spherical particle of equivalent volume. On the other hand, the effect on coagulation is more complex; the area for contact may be increased, thus partially offsetting the reduction in mobility.

Aerosols are dynamic systems: Particles are convected by the gas in which they are suspended. Particles move relative to the suspending gas if they are acted upon by external forces or if they possess sufficient inertia that they are unable to follow changes in the gas flow. Particles diffuse relative to the suspending gas if the gas concentration is spatially non-uniform. New particles may be formed by nucleation from supersaturated vapours or by mechanical disintegration of larger masses. Existing particles may increase in size owing to condensation or decrease in size owing to evaporation. Particles may increase in size and decrease in number by coagulation. Finally particles may be lost from the gas by deposition onto surfaces. The kinetics of these processes, and their relative importance for a given set of conditions, depend on the properties of the suspending gas, the geometry of the system, and the nature of the gas motion.

3.1.2.4.2 Deposition of aerosols

The following mechanisms (references 1. & 2.) are recognised as being potentially important for removal of aerosols from gases in severe accident scenarios:

- Sedimentation caused by gravity onto horizontal surfaces.
- Thermophoresis resulting from temperature differences between the gas and the surfaces.
- Deposition by Stefan flow (convective transport) associated with the condensation of vapours on surfaces and by diffusiophoretic.
- Inertial deposition from turbulent flow near surfaces.
- Deposition by impacting owing to abrupt changes in gas velocity near surfaces which the particles cannot follow (inertial impacting).
- Diffusional deposition owing to particle concentration differences near surfaces.

3.1.2.4.3 Resuspension of Aerosols

Radioactive material deposited on surfaces or in water pools can, under certain circumstances, get airborne again. This resuspension process may occur if the gas flow across the surface increases, the surface temperature increases, the flashing or boiling occurs in the pool, or the material in the pool is converted to volatile form. The terminology for these phenomena have been discussed by a PWG4 group and the definitions proposed by them are as follows (reference 38.). The term resuspension is also used for representing these four phenomena.

(1) Resuspension

Deposited aerosol or condensed materials on the stracture surface is suspended as fine fragments due to drag force by fluid flow. (It may be noted that the resuspension can be initiated by the steam spikes from the water injection on relocated melt or the flow due to hydrogen burns.)

(2) Revaporisation

Chemical compound in the deposited aerosol or condensed materials on the structure surface is vaporised when the vapour pressure at the surface is larger than its partial pressure in the gas due to, for example, FP decay heating or the heat by chemical reactions. (It may be noted that revapourisation can be initiated by a change of chemical conditions such as occurs when air from the containment enters the reactor coolant system.)

(3) Reentrainment

Chemical compound once dissolved in the liquid or deposited on walls is entrained as droplets by the boiling of liquid or steam flashing due to, for example, depressurisation.

(4) Revolatilisation

Dissolved materials in the pool water is evaporated due to the conversion to volatile form by chemical reactions in particular under the radiation field.

3.1.2.4.4 The effects of energetic phenomena on in-containment fission product behaviour

The consequences of hydrogen combustion or steam explosions can be divided into three categories: chemical effects, physical effects and aerosol generation. Hydrogen combustion is expected to have the most significant chemical impact on the fission product iodine. Both hydrogen combustion and steam explosions will have an impact on the aerosols in the containment. The energy deposition in the gas may promote turbulent agglomeration and also lead to changes in aerosol sizes. Steam explosions are not likely to lead to substantial aerosol generation but may in fact lead to phenomena which attenuate existing aerosol concentrations. The phenomenon of pressurised melt ejection could lead to significant aerosol generation.

Steam explosions and hydrogen flames are two high-temperature processes that may lead to chemical changes in the airborne radionuclide inventory in the containment. In steam explosions, the hot core debris will be propelled into the containment atmosphere where high particle temperature, large surface area and plentiful gas reactants will promote reaction. The major consequences of this will be oxidation reactions and a rapid approach to chemical thermodynamic equilibrium.

Hydrogen combustion, in addition to generating very high temperatures at the moving flame front (1000 - 2200 ^{o}C), also generates large transient concentrations of reactive radicals. These radicals may react with radionuclides in either airborne molecular or particulate form; e.g., CsI + OH -> CsOH + I, where the I atoms subsequently form I_2 or HI. However the airborne lifetime of the I_2 or HI will be quite short. I_2 will be subject to plate-out on the aerosol particles, which are present after the H_2 burn. Also, HI is expected to react rapidly with aerosol Ag to form AgI. The iodine on new aerosol particles will be subject to the same aerosol depletion mechanisms as the original aerosol-borne CsI and only a fraction of the I_2 or HI will persist for a longer time as airborne iodine.

A steam explosion or hydrogen combustion will produce a pressure pulse propagated by a shock wave. The high velocity of the particles at the shock front may have an impact on aerosol behaviour via a number of mechanisms. In addition, high temperatures may cause particle vaporisation, which will also affect aerosol physics.

Airborne particles will be subject to the competing processes of agglomeration and de-agglomeration at the shock front. Enhanced agglomeration will be promoted by the fast movement of particles at the shock front which will increase their collision frequency. De-agglomeration will occur if enough energy is transmitted by collisions with fast particles to break the bonds between primary particles and the body of the aggregate. Particles may also be deposited or resuspended from surfaces as a result of a shock.

Aerosol generation via a hydrogen burn can occur via combustion of containment materials, which generates soot. This will have the beneficial effect of increasing the amount of airborne aerosol material, which can help to remove fission product aerosols via agglomeration and deposition.

Fission product aerosols may be generated via two 'explosive' mechanisms. One source is a steam explosion, which propels finely divided particles and water into the containment atmosphere due to the expanding steam. Both fine fragmented material and water droplets may be a source of aerosol formation.

In the following table, key issues, phenomena, physical processes and important design characteristics relevant for the ex-vessel release and transport of fission products are compiled.

Table 3.1.2.4-1

Ex-vessel release and transport of fission product issues & important parameters	Related key issues, phenomena and key physical parameters	Relevant design characteristics
Ex-vessel release of fission products	- Corium-Concrete interactions - Oxidation in containment atmosphere - Coolability of melt	- Concrete aggregation - Potential for adding water on melt in cavity - Cavity geometry
Transport of fission products in containment	- Aerosol behaviour - Effects of hydrogen combustion, e.g., reaction of CsI with radicals suspension, resuspension, deposition, revapourisation, agglomeration, de-agglomeration, etc., of aerosols.	- Containment geometry and design - Active and passive ESFs - Composition of containment atmosphere (e.g., inertisation)
Pool scrubbing	- Effects like: inertial impacting and deposition, diffusional deposition, sedimentation, convective transport, etc., during bubble formation and bubble rise - Temperature of pool (saturation)	- Suppression pool geometry and design - Potential for suppression pool cooling - Design of spargers in vent lines (drywell to wetwell)
Effects of engineered safety features on fission products	- Scrubbing efficiency of ESFs - Filtering efficiency of ESFs - Deposition efficiency of ESFs - Potential for reducing the probability of certain containment failure modes, and thereby the probability of certain Source Terms. - Intentional release of fission products during unfiltered venting	- Design of ESFs - Availability of Emergency Operating Procedures and Severe Accident Management Guidelines

3.1.2.5 Phenomenological RPV and containment failure mode issues related to accident management.

Given severe core damage there are still various possibilities to prevent and/or mitigate large source terms by preventing substantial damage to the containment or by enhancing fission product retention inside containment. The installed A.M. Measures (both hardware and software) have to be accounted for in a level 2 PSA. Some, some of the more commonly planned, implemented, or studied measures will be discussed below.

3.1.2.5.1 In-vessel coolability of core debris by ex-vessel flooding.

By keeping the molten core inside the RPV, ex-vessel phenomena that might threaten the containment can be avoided: e.g., DCH and ex-vessel steam explosions. If in-vessel reflooding of the core debris is impossible, cooling of the core debris by external flooding of the RPV might be a successful accident management strategy. Several phenomena affect the success and feasibility of external reactor vessel cooling and melt retention inside the RPV, to mention (reference 5.):

- natural convection of an in-vessel, molten corium pool,
- corium pool crust formation and stability,
- radiative heat exchange and heat transfer to cooler surfaces inside the vessel,
- thermal hydraulics outside the vessel; heat transfer regimes,
- structural integrity, and
- long term phenomena, like: temperature of upper vessel structures, the effects of a late coolant injection into the vessel.

However, there is a potential downside of cooling the in-vessel core-debris by ex-vessel flooding, see the discussion on ex-vessel steam explosions in section 3.1.2.5.2. In references 20., 21., 25., 26. & 32.; a method is described to model the uncertainties which are involved in the assessment of the benefits against the adverse effects.

3.1.2.5.2 Ex-vessel coolability of core debris in a flooded reactor cavity.

The key factors affecting the coolability of a debris bed are the decay heat, its configuration, and its particle sizes. The higher the power generated in a bed, the more difficult the bed is to cool. If flooded from above, deeper debris beds tend to be less coolable than shallow debris beds of the same volume. Beds of smaller particles are less porous, the surface area for heat transfer is larger, and therefore, the vapour generation rates are increased relative to water ingress rates. Many particle sizes are possible during a severe accident, ranging from fractions of millimetres up to centimetre size and larger. There is no one exact particle size that defines a threshold for coolability. However, particle sizes of a few millimetre and smaller, which could result from steam explosions, are most likely not coolable. A deep bed with sufficiently small or stratified particle sizes, and/or a small coolant volume could produce dryout in the bed even after it was initially quenched.

3.1.2.5.3 Core catcher.

In order to avoid molten core-concrete interactions (and steam explosions) several studies have been performed to assess the effect of so called core catchers. In a study performed by ENEL-DCO (Italy), a stack of staggered graphite beams is proposed (reference 30.) with the aim of obtaining a three-dimensional corium redistribution after melt-through of the RPV. The thin layers of the corium, on one side, and the large heat capacity and thermal conductivity of the stack of graphite beams, on the other, permit to achieve an initial quick solidification and cooldown of the melt. The final cooldown is achieved by gradually flooding the cavity. The flooding of solidified and relatively cold corium prevents steam explosions. The thickness of the corium layer, the thermal conductivity of the corium (dependent on the metallic fraction in the corium) are important factors in this Accident Management Strategy.

3.1.2.5.4 Hydrogen mitigation.

The effect of diluent gases, like steam or CO_2 reduce the likelihood of detonations. This effect can be used as a possible accident management strategy to prevent hydrogen detonations. Filling the containment with another diluent and inert gas (both pre- and post-accident inertisation may be considered) can prevent detonations. A disadvantage of pre-accident inertisation is the inaccessibility of the containment during normal operations for maintenance activities, refuelling preparations, etc. Other possibilities for preventing hydrogen detonation and deflagration, or to mitigate their effects, is to ignite it before dangerous concentrations are reached. See section 3.1.2.1.5. for a discussion on the phenomena.

3.1.2.5.5 Filtered containment venting system.

Filtered containment venting systems are intended to prevent containment failure due to overpressure. Because of the uncertainties in the timing and mode of containment failures, for a number of plants the decision has been made to install a vent as the ultimate protection system. This trades off an increased probability of a smaller release of fission products (one cannot filter out noble gases) against a decreased probability of a larger uncontrolled release.

3.1.2.5.6 Depressurisation of primary coolant system to prevent high pressure melt ejection.

Because of the risks of DCH, considerable thought has been given to adopting deliberate RCS depressurisation prior to vessel failure as an action to mitigate high pressure accident sequences.

3.1.2.6 Source term issues related to Accident Management

Most nuclear power plants have one or more engineered safety features the purpose of which is to limit the pressure build-up inside the containment by condensing the generated steam, and cooling non condensable gases. For example, water sprayed from the upper part of the containment dome might be very effective under certain circumstances for reducing the pressure, as well as for aerosol depletion. However, the containment spray system is an active system, which, in case of electric driven spray-pumps, will fail in a station blackout scenario. On the other hand, the passive engineered safety systems, like suppression pools, ice bed condensers, containment filters may be useful in some accident scenarios where the active safety features are in a failed state.

The magnitude of severe accident source terms for those BWR sequences which do not by-pass the suppression pool is dominated by the effectiveness of the pool for removing the fission products and aerosols. Similarly, the retention capability of a water column in a damaged steam generator is extremely important for the risk of PWR plants. A significant number of important PWR sequences have pathways that include the pressurizer relief tank which can also behave as a suppression pool. A fraction of the aerosol entering the water pool would be trapped inside the pool, with the remainder entering the atmosphere above the pool. The material which penetrates the pool is a source to other containment compartments, and a potential airborne source term to the outside atmosphere.

A large number of the previously mentioned deposition mechanisms play a role in the scrubbing of aerosols carried to the suppression pool by steam-gas mixtures. An important early process in pool scrubbing is the break-up of the gas into fine bubbles. Much of the scrubbing that occurs does so during the fine bubble formation process or during bubble rise through the pool. Inertial impacting, inertial deposition, diffusional deposition, sedimentation, and convective transport due to steam condensation play an important role during bubble formation and bubble rise. The inertial mechanism results from bubble circulation (vortex motion), which is assumed to occur as the bubbles rise through stagnant water. On the other hand, the bursting of the bubbles as the penetrate the surface plus the possible desorption is a mechanism for release of the aerosols to the containment atmosphere and the environment.

Due to the significant condensation on the walls of the containment structure under saturated pool conditions, a suppression pool compartment will also be effective for decontamination in later stages of the accident.

3.1.3 Significance of uncertainties

Uncertainties are associated with the modelling of all of the above discussed phenomena. The uncertainties may substantially influence the selection of appropriate SAM measures.

The treatment of these uncertainties varies among the PSAs. Most widely used are three techniques:

- Inclusion in the containment event tree structure by defining a set of probabilistically weighted outcomes,
- Incorporation of discrete subjective probability distributions over models of an uncertain issues.

Uncertainties modelled in such ways can be propagated through the analyses steps by Monte Carlo simulation, permitting to quantitatively assess their impact on end result quantities.

- A different approach is the incorporation in the PSA by making specific assumptions; to understand the impact of assumptions made on the calculated results, sensitivity studies have to be performed.

Although the state of knowledge on the different phenomena is increasing, there is no general consensus on the uncertainties that remain to be addressed, see, for example, OECD Specialist Meeting on Severe Accident Management Implementation, Niantic, Connecticut, USA., June 12-14, 1995.

An obvious observation is that the impact of some uncertainties can be reduced or even eliminated by suitable SAM measures. For example, keeping the RPV intact reduces the uncertainties associated with ex-vessel cooling and containment issues; therefore, restoration of core cooling either by in-vessel injection or by ex-vessel flooding is considered by some experts as effective protection against the impacts from temperature induced lower head failure.

The significance of the phenomena and the impact of their uncertainties are different for different plant designs. To judge the merits of further reducing uncertainties with impact on SAM, the influence on currently recommended hardware provisions and operator actions should be assessed. The aim should be robust SAM guidelines, regardless of the nature and extent of the uncertainties. The phenomena identified as significantly impacting SAM strategies are compiled in tables 3.1.3-1 to 3.1.3-6. Methodologies for quantitatively factoring in the uncertainties in the end result are presented in Chapter 6 of this report.

Table 3.1.3-1 Characterisation of uncertainties in the analysis of containment loads from in-vessel phenomena in PWRs

Phenomena	Arrest of core melt progression	Temperature induced hot leg/surge line/SGT rupture	In vessel hydrogen generation	In-vessel steam explosion	Bottom head failure
Magnitude of uncertainties and their impact on PSA results	Small for restoration of injection by operator action (significant at US plants) Large for • rate of accident progression • passive depressurisation mechanisms that could lead to restoration of injection Involves expert judgement Significant impact on the share of high pressure sequences	Large Involves expert judgement Significant impact on the share of high pressure sequences	Large Involves expert judgement Significant impact on the amount of hydrogen generated Impact of the uncertainties on early containment failure probabilities is small for large dry concrete containments because containment loads due to hydrogen combustion stay well below critical containment loads. .	Large Based on expert judgement The impact of the uncertainties on conditional probabilities of containment failure due to steam explosion, given core melt, is small due to the low value of the respective conditional probabilities, relative to other events.	Large Based on expert judgement The loads from bottom head failure with the system at high pressure are significant contributors to early containment failure, but in absolute terms, the conditional probabilities of early containment failure due to bottom head failure are small. However, at plants without mitigation of releases from SGTR events, the latter events dominate LRCF modes, thus reducing the relative importance of bottom head failure events. At plants with mitigation of releases from SGTR events, the loads due to bottom head failure at elevated pressure are the dominant cause for LRCF modes.

Table 3.1-3-2. Characterisation of uncertainties in the analysis of containment loads from ex-vessel phenomena, PWRs

Phenomena	Loads at vessel breach	Ex-vessel steam explosion	Ex-vessel generation of non-condensable gases	Combustion of hydrogen and carbon monoxide	Molten corium /containment structure interaction	Containment structural response to pressurisation
Magnitude of uncertainties and their impact on PSA results	• Large for individual DCH phenomena • Small for conditional probabilities of containment failure due to DCH, given core damage (DCH loads stay well below containment capacity The loads from DCH are significant contributors to early containment failure, but in absolute terms, the conditional probabilities of early containment failure due DCH are small however, at plants without mitigation of releases from SGTR events, the latter events dominate LRCF modes, thus reducing the relative importance of DCH. At plants with mitigation of f releases from SGTR events, the loads due to DCH are the dominant cause for LRCF	Large Based on expert judgement The impact of the uncertainties on conditional probabilities of containment failure due to steam explosion, given core melt, is small due to the low value of the respective conditional probabilities, relative to other events.	• Small for gases generated by reaction of unoxidised zirconium with water • Large for gases generated by core debris/concrete interaction (CCI). MAAP assumes suppression of CCI if the cavity is filled with water. This assumptions is not made in the other codes • Large for time history of gas generation.	Large Involves expert judgement (see in-vessel generation of hydrogen) Impact of the uncertainties on early containment failure probabilities is small for large dry concrete containments because containment loads due to hydrogen combustion stay well below critical containment loads. Impact of uncertainties on late containment failure probabilities is small: for plants without filtered venting because containment failure is almost guaranteed. For plants with filtered containment venting, containment structural integrity is preserved with high conditional probability.	Large MAAP assumes suppression of CCI if the cavity is filled with water. This prevents basemat erosion. This assumption is not made in the other codes, therefore basemat penetration is highly likely.	Small to medium

Table 3.1.3-3 Characterisation of uncertainties in the analysis of source term issues, PWRs.

Phenomena	In-vessel fission product release, transport and retention	Scrubbing in water filled steam generator or in water pool	Fission product release, transport and retention inside containment	Environmental release
Magnitude of uncertainties and their impact on PSA results	• Medium for noble gases and volatile fission products • Large for refractory aerosols	Medium to large Impact of the uncertainties on releases is large. More clarification is needed.	Large Phenomena, in particular time history are not well understood. Impact of the uncertainties on releases attending LRCF modes is small to moderate, because depletion processes are not effective. The impact on late releases is small because the time at which such releases occur is long relative to the time constants of depletion processes.	Medium For comparable plant designs, MAAP calculations tend to predict lower releases than NUREG-1150 analyses.

Table 3.1.3-4 Characterisation of uncertainties in the analysis of containment loads from in-vessel phenomena, BWRs.

Phenomena	Arrest of core melt progression	In vessel hydrogen generation	In-vessel steam explosion	Bottom head failure
Magnitude of uncertainties and their impact on PSA results	Small for restoration of injection by operator action (significant at US plants) Large for rate of accident progression	Large Involves expert judgement Significant impact on the amount of hydrogen generated Impact of the uncertainties on early containment failure probabilities is small because containments are inerted, or igniters are available. .	Large Based on expert judgement The impact of the uncertainties on conditional probabilities of containment failure due to steam explosion, given core melt, is small due to the low value of the respective conditional probabilities, relative to other events.	Large Based on expert judgement The impact of the uncertainties on conditional probabilities of containment failure is significant.

Table 3.1.3-5 Characterisation of uncertainties in the analysis of containment loads resulting from ex-vessel phenomena, BWRs

Phenomena	Loads at vessel breach	Ex-vessel steam explosion	Ex-vessel generation of non-condensable gases	Combustion of hydrogen and carbon monoxide	Molten corium/containment interaction	Containment structural response
Magnitude of uncertainties and their impact on PSA results	Large Significant impact on conditional containment failure probabilities.	Large Based on expert judgement The impact of the uncertainties on conditional probabilities of containment failure due to steam explosion, given core melt, is small due to the low value of the respective conditional probabilities, relative to other events.	Small for gases generated by reaction of unoxidised zirconium with water Large for gases generated by core debris/concrete interaction (CCI). MAAP assumes suppression of CCI if the wetwell or pedestal area is filled with water. This assumption is not made in the other codes Large for time history of gas generation.	Small for plants with inerted containment Large for plants with Mark III containment (not inerted) because of uncertainties about the effectiveness of igniters.	Large In general, the impact of the uncertainties on conditional probabilities of containment failure is large	Small to medium

Table 3.1.3-6. Characterisation of uncertainties in the analysis of source term issues, BWRs.

Phenomena	In-vessel fission product release and retention	Scrubbing in suppression pool	Ex-vessel fission product release, transport and depletion inside containment	Environmental release
Magnitude of uncertainties and their impact on PSA results	• Medium for noble gases and volatile fission products • Large for refractory aerosols	Medium to large Impact of the uncertainties on releases is large. More clarification is needed.	Large Phenomena, in particular time history are not well understood. Impact of the uncertainties on releases attending LRCF modes is small to moderate, because depletion processes are not effective. The impact on late releases is small because the time at which such releases occur is long relative to the time constants of depletion processes.	Medium For comparable plant designs, MAAP calculations tend to predict lower releases than NUREG-1150 analyses.

3.1.4 References

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3.2 Review of severe accident computer codes

The referencing to publications in this section refers to the list in subsection 3.2.5.

3.2.1 Introduction

The physical and chemical processes governing the progression of severe accidents are very complex, often involving many simultaneous phenomenological interactions, for which detailed experimental information is not available. Therefore, mathematical modelling, and computer simulation of these phenomenological processes are influenced by various uncertainties. Experimental and analytic studies (including benchmark exercises) are being conducted at the international level in order to improve the confidence in the predictive capabilities of the models embedded in the various severe accident computer codes.

The purpose of this section is to list the codes which have been developed following these studies, and which are most commonly used in Level 2 PSAs, giving for each code a very brief description of the design objectives, models, features, and known limitations (experimental bench marking status is discussed in Section 3.2.4). Note that, a review of the available severe accident computer codes with special emphasis on in-vessel melt progression models has already been published in reference 1., thus, there is some degree of overlap between the present compilation and that of reference 1. However, the status of the codes under development has considerably changed since the earlier report was published, thus this section is both an update and a complement to the information provided in reference 1. In addition, some general principles on the use and application of accident analysis models/codes in PSAs are outlined, based on the recent IAEA level 2 PSA procedures guide (reference 2.)

3.2.2 General Principles of Computer Codes

3.2.2.1 Types of Codes

The codes which model the phenomenology of severe accidents are divided into three types according to their modelling details and design objectives. The separate phenomena codes attempt to model the phenomena with detailed models, consistent with the state-of-the-art, and the availability of experimental data (references 3. & 4.). These codes are used typically in severe accident research, evaluation of severe accident management alternatives, and as benchmarks for the simpler, more parametric and integrated computer codes.

Among the separate phenomena codes are also those describing the thermal-hydraulic and core degradation mechanisms following severe reactor accidents; the fission product release, aerosol retention and transport models; the containment building structural response models to evaluate the internal loads resulting from LOCAs, steam explosions, hydrogen burn or internal missiles. Generally these codes consist of multi-purpose models for analysis of complex physical, chemical, and mechanical conditions, following severe reactor accidents. Detailed discussions of separate phenomena severe accident and radionuclide release and transport models will be provided later.

The containment structural response analysis codes are able to deal with non-linearities, both geometrical (i.e., large displacement) and material (i.e. plasticity). The code architecture allows easy implementation of various material models and failure criteria into it. The codes can usually model both, local (e.g. crack propagation, penetration) and global responses of the loaded structure. There are many codes available to analyse containment response, e.g., DYNA3D, ABAQUS, NASTRAN, HONDO, NEPTUNE, WHAMS.

In contrast, the integrated PSA codes, intended for routine application in PSAs, are designed to be (relatively) fast running, so that they can calculate many sequences (and a number of times for a single sequence if uncertainty analyses are required). In order to achieve these shorter run times, the modelling is sometimes simpler than in the separate phenomena codes. As an example, consider the approach to modelling of aerosol transport, following core melt accidents. In the separate phenomena codes a numerical solution is found for the integral-differential equation for aerosol agglomeration and deposition, giving the aerosol size distribution at each time step. In contrast, the Electric Power Research Institute (EPRI) sponsored MAAP code (reference 5.), developed for PSA application, uses a correlational approach for aerosol behaviour. This model uses correlations of exact solutions of the polydisperse integro-differential aerosol equation. If for a sequence one is concerned that the correlation may be outside its range of applicability, one could reinforce the information with further calculations using a more detailed aerosol transport code (reference 2.)

It should be noted that often the separate phenomena codes can go into greater mechanistic detail than the integrated PSA codes, but in fact just as often, the more detailed models of the separate phenomena codes are just as often subsumed into the integrated PSA codes.

There are also the so-called simple parametric codes, and computational tools, intended for specific PSA applications, such as source term estimation or hydrogen combustion loads under a variety of boundary conditions, in which the assessment of uncertainties on accident progression pathways requires extensive repetitive calculations. These are based on simple parametric models which interpolate between fixed points for which calculations with a more detailed code have been performed to determine the values of the parameters. The use of such codes is reasonable for generating uncertainty ranges, but it is important to take into account that the parameters used in these codes as well as the results produced by them have to be calibrated by more detailed calculations and/or experimental data (reference 2.).

In order to cover the whole range of phenomena needed to determine the challenges to the containment integrity, and the radiological releases, one typically does not use a single monolithic code. A suite of separate codes, each dealing with a particular phase or aspect of severe accident behaviour, are coupled with some interfacing facility for the transfer of information between the codes. A more integrated and modular approach tends to be adopted in the newer generation of severe accident codes (reference 2.).

In general, these could not and should not be used as "black-boxes". A deep understanding of both the code used and the problem analysed is the necessary prerequisite for application of these complex tools.

3.2.2.2 Code Verification and Validation

Verification and validation of computer codes are crucial mechanisms in a process to enhance confidence in their application. It is useful to distinguish between the different stages of testing a code using the words "verification" and "validation". In the most common usage, verification means testing by performing the calculations for which it is intended. A code which solves a differential equation might be tried out on a known analytic solution of the equation to test that it is indeed giving solutions to an acceptable level of accuracy. However complex the phenomena may be, the laws of conservation of mass, energy, and momentum must apply. Checking that the code predictions obey the conservation laws would be another simple verification test. Validation, on the other hand, is a process that a code must undergo to see if it provides a sufficiently accurate representation of the reality of the severe accident phenomena modelled (reference 2.).

Achieving a state with severe accident codes which could reasonably be called validation is very difficult. The extreme conditions which occur in a severe accident and the scale of the physical geometry are difficult to realise in laboratory experiments. The process of validation, in general, comprises a validation matrix involving many simulations, ranging from perhaps comparison with separate effects experiments in examining the more fundamental aspects of the phenomena, to larger scale integral experiments. Typically, the experiments, designed on the basis of scaling arguments, are conducted in smaller scale facilities using some representative simulant materials (reference 2.).

One should always be alerted to some code validations which have been achieved by varying user-supplied parameters until a reasonable fit to experimental data is achieved. At best this is an indirect experimental measurement of the parameter values, and not an independent validation of the code. A true validation should involve the accurate prediction of many more data points than there are adjustable parameters within the code (reference 2.).

Experimental validation of computer codes requires an evaluation of the governing spatial and temporal scales, a determination of code applicability to the phenomena of interest, and finally benchmarking studies.

3.2.2.3 Minimum Requirements for Code Users

By definition, PSA codes should be designed so that a Level 2 PSA analyst, with a good degree of familiarisation of overall accident phenomena, can use these codes reliably. It is also essential that the analyst must have a good working knowledge of the reactor systems. In order for the code calculations to be meaningfully incorporated in the framework of a Level 2 PSA, it is essential that the analyst has a reasonable knowledge of the following:

- The phenomena addressed, applicability of the models, and their limitations;
- The meaning and significance of the input variables;
- The meaning and significance of the output variables.

The point to be emphasised here is that, given the complexity of these issues, the codes must not be simply treated as "black boxes."

Many codes ask the user to specify the time steps for the differential equation solvers within the code. A choice of too small a time step will make the run time unacceptably long, while a choice of too large a time step will make the solution inaccurate. Numeric instabilities can also occur with either. The analyst should check the sensitivity of the predictions to the choice of time step and look for convergence of the results as a function of decreasing time step. If, for a given application, convergence cannot be achieved without going to impractically long run times, the code may be inappropriate for this application (reference 2.).

3.2.3 Severe Accident Codes

Table 3.2-1 lists the various severe accident computer codes in use for Level 2 PSAs, which deal with most or all of the phenomena pertinent to LWR severe accidents. The parametric special purpose codes and computational tools are not included. A detailed description of these codes is provided in Appendix A.

Table 3.2-1 Severe accident computer codes

Country	Computer Codes	**In-Vessel Phenomena**				
USA		thermal-hydraulics	core melt progression	release from fuel	transport in RCS	vessel failure
	MELCOR [3.6]	+	+	+	+	+
	MAAP [3.5]	+	+	+	+	+
	SCDAP-RELAP5 [3.7]	+		+	+	+
	VICTORIA [3.8]	+				+
	COMMIX-1C [3.15]					+
	IFCI [3.9]					FCI
	TEXAS [3.10]					FCI
	PM-ALPHA/EPROSE [3.11]					FCI

Country	Computer Codes	**Ex-Vessel Phenomena**					
USA		high pressure melt injection	core-concrete interaction	fission product release from debris	fission product transport in containment	hydrogen combustion	containment response/ loads
	MELCOR [3.6]	+	+	+	+	+	+
	MAAP [3.5]	+	+	+	+	+	+
	CONTAIN [3.12]		+	+	+	+	+
	CORCON/MOD3 [3.13]			+		+	+
	HMS-BURN [3.14]					+	+
	IFCI [3.9]						FCI
	TEXAS [3.10]						FCI
	PM-ALPHA/EPROSE [3.11]						FCI

Table 3.2-1 (Cont.) Severe accident computer codes

Country	Computer Codes	In-Vessel Phenomena				
France		thermal-hydraulics	core melt progression	release from fuel	transport in RCS	vessel failure
	ESCADRE [3.53 –3.58]	+	+	+	+	+
	MELCOR [3.6]	+	+	+	+	+
	CATHARE [3.59]	+				
	ICARE [3.60-3.61]	+	+	+		
	MC3D [3.64-3.65]					FCI

Country	Computer Codes	Ex-Vessel Phenomena					
France		high pressure melt injection	core-concrete interaction	fission product release from debris	fission product transport in containment	hydrogen combustion	containment response/ loads
	ESCADRE [3.53-3.58]	+	+		+		
	MELCOR [3.6]	+	+	+	+		+
	PLEXUS [3.62]						+
	CASTEM [3.63]						+
	MC3D [3.64-3.65]						FCI

Country	Computer Codes	In-Vessel Phenomena				
Japan		thermal-hydraulics	core melt progression	release from fuel	transport in RCS	vessel failure
	THALES [3.66]	+	+			+
	ART [3.67-3.68]			+	+	
	THALES-2 [3.69]	+	+	+	+	+
	MACRES [3.70]	+			+	
	REMOVAL [3.71]					
	MAPLE [3.72]					
	JASMINE [3.73]					+

Country	Computer Codes	Ex-Vessel Phenomena					
Japan		high pressure melt injection	core-concrete interaction	fission product release from debris	fission product transport in containment	hydrogen combustion	containment response/ loads
	THALES [3.66]		+			+	+
	ART [3.67-3.68]	+		+	+		
	THALES-2 [3.69]		+	+	+	+	+
	MACRES [3.70]				+		
	REMOVAL [3.71]				+		
	MAPLE [3.72]	+					
	JASMINE [3.73]						FCI

Table 3.2-1 (Cont.) Severe accident computer codes

Country	Computer Codes	In-Vessel Phenomena				
UK		thermal-hydraulics	core melt progression	release from fuel	transport in RCS	vessel failure
	MAAP [3.5]	+	+	+	+	+
	SCDAP-RELAP	+		+	+	+
	VICTORIA [3.8]	+				+

Country	Computer Codes	Ex-Vessel Phenomena					
UK		high pressure melt injection	core-concrete interaction	fission product release from debris	fission product transport in containment	hydrogen combustion	containment response/ loads
	MAAP [3.5]	+	+	+	+	+	+
	CONTAIN [3.12]		+	+	+	+	+
	CORCON			+		+	+
	CORDE			+	+		

Country	Computer Codes	In-Vessel Phenomena				
Germany		thermal-hydraulics	core melt progression	release from fuel	transport in RCS	vessel failure
	ATHLET-CD [3.94-3.95]	+	+	+	+	+
	IVA-4 [3.96]	+				+, FCI

Country	Computer Codes	Ex-Vessel Phenomena					
Germany		high pressure melt injection	core-concrete interaction	fission product release from debris	fission product transport in containment	hydrogen combustion	containment response/ loads
	WECHSL [3.97]			+			
	RALOC [3.98]					+	+
	FIPLOC [3.99]				+		+
	SAGE PROC [3.100]			+			
	IVA-4 [3.96]	+				+	FCI

Table 3.2-1 (Cont.) Severe accident computer codes

Country	Computer Codes	**In-Vessel Phenomena**				
EC		thermal-hydraulics	core melt progression	release from fuel	transport in RCS	vessel failure
	ESTER 1.0 [3.101-3.104]	+	+	+	+	

Country	Computer Codes	**Ex-Vessel Phenomena**					
EC		high pressure melt injection	core-concrete interaction	fission product release from debris	fission product transport in containment	hydrogen combustion	containment response/ loads
	ESTER 1.0 [3.101-3.104]		+	+		+	

3.2.4 Assessment Status of Codes

A systematic code assessment effort requires identification of dominant phenomena for which appropriate physical models exist in the code, then the available and planned separate effects and integral experiments should be matched against the code model for assessment. Any experimental benchmarking effort must consider the issues of scaling and code applicability.

3.2.4.1 Assessment Status of MAAP

The assessment matrix listed in Table 3.2-2 shows the experimental benchmarking status of the MAAP computer code (reference 16.). It is seen that the various code versions (entries in the matrix refer to MAAP version number) have been compared to several separate effects and integral experiments. These include: CORA and PHEBUS (core damage); LOFT FP-2 (integral severe accident test); ABCOVE (aerosol behaviour); CSE (containment spray); COPO (molten pool heat transfer); FARO (debris quenching); Surtsey IET (DCH); SWISS, SURC-4, ACE, KfK BETA (core-concrete interaction); NUPEC mixing tests; Marviken, FAI, and GE vessel blowdown tests; and HDR containment experiment, among many others. The recent version of the code, MAAP 4 (reference 17.), has also been benchmarked against the TMI-2 accident. This comparison study shows that MAAP4 provides a reasonable simulation of the TMI-2 accident in terms of the system response prior to core uncovery, during core degradation, following core reflood, and the lower head behaviour after 224 minutes. These are all severe accident processes that are essential for application of computer codes for decisions related to design, operations, emergency operating procedures, and accident management (reference 53.).

The comparison of MAAP 4 calculations with the HDR T31.5 experimental data showed that the pressure was predicted accurately, but the local prediction of the temperature was not as good since the code uses lumped parameter models (reference 18.). The MAAP 4 code predictions of the NUPEC mixing tests showed good agreement with the experimental data, and specifically, the gas concentration comparisons were encouraging (reference 19.). In general, the PHEBUS FPT0 test results were in good agreement with the MAAP 4 predictions, but the hydrogen production was over predicted (reference 20.). MAAP has also been benchmarked for FLHT-2 and PBF-SFD 1-1 experiments. There is also a Browns Ferry fire benchmark (reference 53.).

3.2.4.2 Assessment Status of MELCOR

As part of the MELCOR Code Assessment Program (MCAP), currently underway under NRC sponsorship, the experimental basis for MELCOR is being increased through an international co-operative effort. An overview of the MELCOR assessment program is documented in reference 21. Table 3.2-3 lists the experimental assessment matrix for the various versions of the MELCOR computer code based on the information in reference 21.

A code assessment program concentrating on the in-vessel phenomena and on full plant calculations has been underway at Brookhaven National Laboratory. The code was benchmarked against experimental results from the PBF-SFD and NRU-FLHT core damage tests. In addition, full plant calculations were performed for the Peach Bottom, Zion, Oconee, and Calvert Cliffs plants.

The program at Sandia concentrated on thermal/hydraulics and fission product release and transport for both in-vessel and ex-vessel phenomena. This included the analysis of the FLECHT SEASET natural circulation tests, the OECD LOFT integral severe accident experiment LP-FP-2, fission product release and deposition tests LACE LA4 containment geometry aerosol and deposition test, the ACRR ST-1/ST-2 in-pile source term tests, the GE vessel blowdown and level swell tests, the PHEBUS B9+ and CORA-13 core damage tests, the DF-4 BWR damaged fuel experiment, the HDR T31.5 containment blowdown and mixing experiment, and the IET direct containment heating experiments.

Other tests analysed using the MELCOR code by various organisations include BMC-F2 containment thermal/hydraulic test, NUPEC hydrogen mixing tests M-4-3 and M-7-1, DEMONA F2 containment experiment, FALCON fission product transport and deposition experiment, and FIST BWR thermal hydraulic tests 6SB2C and T1QUV.

MELCOR simulation of TMI-2 accident (reference 21.) showed that the Phase 1 (0 to 100 min) and Phase 2 (100 to 174 min) was well predicted, as compared with the available data. However, Phase 3 (174 to 200 min) and Phase 4 (200-300 min) comparison of code predictions with the available data showed limitations in MELCOR modelling of reflood and core relocation behaviour.

The results of primary system thermal/hydraulic code assessment indicated deficiencies in the prediction of two-phase natural circulation, modelling sensitivity to input values for the flow paths connecting vertically stacked control volumes. The MELCOR modelling of the blowdown appeared adequate. The MELCOR prediction of in-vessel core damage behaviour was found to be satisfactory, and comparison with the SCDAP/RELAP5 results showed similar behaviour. Even though the overall behaviour of the containment response can be modelled reasonably well with MELCOR, the detailed response in complex geometries is not predicted well (this is expected for a control-volume code such as MELCOR). The results of the fission product source term comparison with experimental data showed reasonably good agreement.

Table 3.2-2 Summary of Benchmarks for MAAP3B and MAAP4 [3.16]

		Type of Comparison						Documentation			
		Separate Effects Experiments		Integral Experiments		Industry Experience (TMI-2)	Detailed Analysis				
Physical Process	**Experiment / Code**	**Small Scale**	**Large Scale**	**Out-of-Reactor**	**In-Reactor**			**Open Literature**	**IDCOR Reports**	**MAAP User's Manual**	**EPRI Reports**
Core Heatup[3]	TMI-2					3B/4				4	
	CORA		4	4						4	
	PHEBUS[4]				4						
Clad Oxidation[5]	Numerous Experiments	1							1		
	LOFT FP-2				3						3/[3.51]
	TMI-2					2/3/3B/4				4	2/[3.44]
	BWR Heatup Code						½		½	½	
	PWR Heatup Code						½		½	½	
	CORA									4	
	PHEBUS[6]				4						
Fission Product Release[7]	ORNL experiments	2							2	2	
	SASCHA experiments[8]	2							2	2	
	LOFT FP-2				3						3/[3.51]
	TMI-2					3/3B/4				4	
Aerosol Transport and Deposition[9]	ABCOVE Tests			3/3B/4				[3.30]	85.2	3B/4	
	CSE Tests			3/3B/4					85.2	3B/4	
	Gillespie and Langstroth	3							85.2		
	Discrete (Sectionalised) Code						4				

3 New model in MAAP4
4 Results proprietary to Electricite de France (EdF)
5 Model essentially the same since MAAP1)
6 Results proprietary to Electricite de France (EdF)
7 Model is unchanged since MAAP2
8 These experiments (ORNL and SASCHA) are the bases for the correlation
9 Model essentially the same since MAAP3

Table 3.2-2 (cont'd.) Summary of Benchmarks for MAAP3B and MAAP4 [3.16]

		Type of Comparison						Documentation			
		Separate Effects Experiments		Integral Experiments		Industry Experience (TMI-2)	Detailed Analysis				
Physical Process	**Experiment / Code**	**Small Scale**	**Large Scale**	**Out-of-Reactor**	**In-Reactor**			**Open Literature**	**IDCOR Reports**	**MAAP User's Manual**	**EPRI Reports**
Hydrogen Combustion											
Complete[10]	Thermodynamic Anal.						3B/4			3B/4	
Incomplete[11]	Whitehell Tests		3B/4					[3.49]		3B/4	
	EPRI Tests		3B/4					[3.49]		3B/4	
	SNL VGES Tests		3B/4					[3.49]		3B/4	
	EPRI Nevada Tests		3B/4							3B/4	
In-Vessel Cooling[12]	TMI-2					4		[3.38]		4	
RPV External Cooling[13]	CECo/FAI Tests	4		4				[3.35]			
	Finite Element Tests										[3.50]
RPV Failure Models[14]	EPRI Lower Head Penetration Response Tests		4	4						4	[3.34]
	TMI-2						4			4	
Molten Debris Heat Transfer[15]	COPO Tests		4	4				[3.52]		4	
	UCLA Tests	4		4							]3.46]
Debris Fragmentation in the RPV Lower Plenum[16]	KfK Tests	4						[3.39]			
	FARO		4	4				[3.39]			
	TMI-2					4				4	

10 New Model for MAAP3B and MAAP4
11 New Model for MAAP3B and MAAP4
12 New Model in MAAP4
13 New Model in MAAP4
14 New Model in MAAP4
15 New Model in MAAP4
16 Jet entraintment Model new in MAAP4

Table 3.2-2 (cont'd.) Summary of Benchmarks for MAAP3B and MAAP4 [3.16]

		Type of Comparison						**Documentation**			
		Separate Effects Experiments		**Integral Experiments**		**Industry Experience (TMI-2)**	**Detailed Analysis**				
Physical Process	**Experiment / Code**	**Small Scale**	**Large Scale**	**Out-of-Reactor**	**In-Reactor**			**Open Literature**	**IDCOR Reports**	**MAAP User's Manual**	**EPRI Reports**
Debris Dispersal[17]	Sandia SURTSEY 1/10th Scale Experiments			4				[3.37]		[3.37]	
Debris Coolability[18]	Numerous experiments discussed in the IDCOR reports							[3.35]	14.1 and 15.2		
Core-Concrete Attack[19]	1st SNL Steel-Conc Exp.			1				[3.36]	15.3	3/3B	
	WECHSL Analysis						1	[3.36]	15.3	2	
	SNL Swiss Experiments	3/3B		3/3B					86.2	3/3B	
	SNL CC Test	3/3B							86.2	3/3B	
	SNL SURC-4 Test	4							86.2	4	
	KfK BETA Tests (V51, V52 and V61)	4								4	
	ACE Tests (L2, L5, L6 and L7)	4								4	
Wall Ablation[20]	Closed Form Solution						1				
Fan Cooler[21]	Westinghouse Exp.	1								4	
	TMI					4					
Revaporisation[22]	ANL Results	3							85.2	3B/4	

17 New optional Model/DCH1/ added in MAAP4

18 Model essentially the same since MAAP1

19 1D Model in MAAP1-MAAP3B,Model made 2D in MAAP4

20 Model essentially the same since MAAP1

21 Model essentially the same since MAAP1

22 Model essentially the same since MAAP3

Table 3.2-2 Summary of Benchmarks for MAAP3B and MAAP4 [3.16]

		Type of Comparison						**Documentation**			
		Separate Effects Experiments		**Integral Experiments**		**Industry Experience (TMI-2)**	**Detailed Analysis**				
Physical Process	**Experiment / Code**	**Small Scale**	**Large Scale**	**Out-of-Reactor**	**In-Reactor**			**Open Literature**	**IDCOR Reports**	**MAAP User's Manual**	**EPRI Reports**
Primary System T/H	Davis-Besse LOFA					3B					
	Browns Ferry					3B		[3.28]			
	Mist Exps.			3B							[3.29]
	Semiscale Experiments										[3.45]
	Fist Experiments										[3.32]
	BWR Code Results										[3.32]
	PWR Code Results										[3.45]
	TMI-2					4				4	
	Peach Bottom TT Tests					4					
	Oyster Creek LOF					3B					
	PHEBUS[23]		4	4							
	OSU AP600 Tests[24]										
Primary System Natural Recirculation[25]	Westinghouse Exp.			3					85.2	3B/4	
Fan Cooler[26]	Westinghouse Exp.	1								4	
	TMI					4					

23 Results proprietary to Electricite de France (EdF)

24 Results proprietary to West.

25 Model essentially the same since MAAP3

26 Model essentially the same since MAAP1

Table 3.2-2 Summary of Benchmarks for MAAP3B and MAAP4 [3.16]

		Type of Comparison						Documentation			
		Separate Effects Experiments		Integral Experiments		Industry Experience (TMI-2)	Detailed Analysis				
Physical Process	**Experiment / Code**	**Small Scale**	**Large Scale**	**Out-of-Reactor**	**In-Reactor**			**Open Literature**	**IDCOR Reports**	**MAAP User's Manual**	**EPRI Reports**
Containment Natural Circulation[27]	HEDL CSTF Tests		3B/4								[3.32]
	FAI Brine-Water Mixing Exps.	3B/4								[3.31]	
	HDR		4	4						4	
	NUPEC Tests[28]		4	4							
	AP600 Cont. Tests*		4	4							[3.46]
Containment Strain[29]	Canadian Experiments		3B/4							3B/4	
	SNL Experiments		3B/4							3B/4	
	SNL Analysis						3B/4			3B/4	
Pressuriser Model[30]	Marviken Blowdown Test		4	4						4	
	FAI 2-Phase Blowdown Exp.		4	4						4	
	GE Vessel Blowdown		4	4							

Where models have remained essentially unchanged since MAAP1, MAAP2B or MAAP3 the previous benchmarks are also listed

27 New Model in MAAP4 which used 3B aux. bldg. model
28 Results proprietary to NUPEC
29 Model is unchanged since MAAP3
30 New Model in MAAP4 (FLOEXP)

Table 3.2-3 Summary of Benchmarks for MELCOR

Physical Process	Experiment/Code	MELCOR Version				
		1.7.1	1.8.0	1.8.1	1.8.2	1.8.3
Primary Thermal/ Hydraulic	PMK Bleed and Feed			+		
	FLECHT-SEASET NC Tests			+		
	FIST 6SB2C T1QUV				+	
	MIST 3109AA 3404AA				+	
	GE Level Swell			+	+	+
	HFIR "Spring Constant"			+		
	OECD LOFT FP-2			+		
Core Heat Transfer	OECD LOFT FP-2			+		
Oxidation H_2 Production	OECD LOFT FP-2			+		
	PHEBUS B9+ (ISP-2B)		+			
	CORA 13 (ISP-31)			+		
	ACRR DF-4				+	
Core Melt Progression	OECD LOFT FP-2			+		
	PHEBUS B9+ (ISP-2B)		+			
	CORA 13 (ISP-31)			+		
	ACRR DF-4				+	
	ACRR MP-1 and MP-2				+	
	FLHT-2, FLHT-4			+		
	FLHT-5				+	
	PBF SFD 1-1	+				
	PBF SFD 1-4		+			
Fission Product Release	OECD LOFT FP-2			+		
	FLHT-2, FLHT-4			+		
	FLHT-5				+	
	VI-3, VI-5, VI-6				+	
	ACRR ST-1, ST-2			+		
Aerosol Transport and Deposition	OECD LOFT FP-2			+		
	Marviken ATT-2b, ATT-4			+		
Integral Analysis	OECD LOFT FP-2			+		
	TMI-2		+			
Containment ESF	PNL Ice Condenser 11-6, 16-11			+		
Containment T/H	HDR V44		+			
	HDR T31.5 (ISP-23)	+	+			
	HDR E11.2 (ISP-29)					
	NUPEC M-4-4, M-7-1 (ISP-35)				+	
	BMC F2		+			
	DEMONA F2		+			
Aerosol Behaviour	ABCOVE AB5, AB6, AB7				+	
	LACE LA-			+		
DCH	Surtsey IET, ANL CWT. IET				+	
Hydrogen Combustion	Surtsey IET, ANL CWT. IET				+	

3.2.4.3 Assessment Status of VICTORIA

Extensive code assessment work has been completed for VICTORIA, this includes comparison to ACRR-ST1, Oak Ridge National Laboratory (ORNL) HI-3 and VI-3, FALCON, Marviken V, EPRI-LACE, and PHEBUS-FP experiments.

3.2.4.4 Assessment Status of SCDAP/RELAP5

Experimental validation of the SCDAP/RELAP5 code has been performed primarily by the code developers at Idaho National Engineering Laboratory (INEL) (reference 7.). Examples include, simulation of PBF/SFD series of tests, NRU/FLHT full length tests, PHEBUS B9+ experiment, OECD LOFT LP-FP-2 experiment, ACRR experiment DF-4, PBF/SFD, and CORA tests (references 3. & 7.). These so-called developmental validation activities have always found weaknesses in the available models, including (1) deficiencies in the initial relocation of the liquefied fuel rod material, (2) lack of models for flow diversion due to changes in geometry, (3) problems associated with modelling of multi-dimensional flow patterns in the upper plenum region, and (4) deficiencies in the oxidation model, once the initial bundle geometry is lost. Other SCDAP/RELAP5 damage progression model deficiencies include (reference 3.) (a) influence of ballooning on flow and heatup, (b) oxidation of the inner surface of a fuel rod, (c) oxidation of relocated material that forms a metallic blockage, (d) hydrogen generation during reflood, (e) relocation of ceramic fuel rod material, and (f) the interaction between the bundle material and the complex flow of rivulet and droplets. Additional validation activities are also underway in several other countries, including United Kingdom, France, Sweden, Germany, and Switzerland.

Following identification of the modelling inadequacies in SCDAP/RELAP5, new or improved models are being proposed for incorporation into the code. Examples include, debris heatup and melting, molten pool formation and growth, molten pool crust failure, candling, cladding deformation, and core fragment mixture heat conductivity.

3.2.4.5 Assessment Status of CONTAIN

The independent peer review of CONTAIN code has been completed and the review findings have been documented (reference 4.).

Examples of CONTAIN experimental benchmark studies include: comparison of code results with the test data from the Hanford Engineering and Development Laboratories (HEDL) ABCOVE tests, benchmarking against the LACE test data, the V44, T31.5 and T31.6 test at HDR, PNL CSE tests, hydrogen burn tests at the Nevada Test Site (NTS), and the NSPP tests at the Oak Ridge National Laboratories. Although, to a leading order, the code predictions were found to be comparable to the test data, significant limitations in the code capabilities for the prediction of mixing and stratification were identified.

The comparison of the CONTAIN calculations with the NUPEC mixing experiments (reference 22.) showed that the gas concentration and temperature were well predicted, but the final gas pressure was always over predicted. The modelling of the CONTAIN spray system revealed that modelling modifications are required in the input deck. An assessment of the CONTAIN DCH models have been performed using comparison with the experimental data (reference 23.). The code calculations showed sensitivity to the model parameters, but the containment pressurisation was found to be reasonably predicted.

3.2.4.6 Assessment Status of HMS

Previous code versions of HMS have been used to simulate experimental results of Sandia FLAME and VGES tests, Nevada hydrogen tests in the NTS, Tests in the HCOG facility, PHDR large scale hydrogen mixing tests, and PHDR large scale fire experiments.

The current code version has been assessed against the Sandia Flame Facility data where injection of hydrogen into the test vessel was simulated using the HMS code. Two-dimensional and three-dimensional simulations showed the buoyant plume development and subsequent stratification in the test vessel. The code has also been used to simulate the HDR T31.5 experiment. The experiment was intended to simulate a large break LOCA. The calculated results were shown to compare reasonably well with the test results.

3.2.4.7 Assessment Status of IFCI

A number of parametric calculations using the experimental conditions in FARO quenching tests and IET-8 experiments have been performed (reference 24.). The predicted pressures and temperatures in the FARO quenching test were within 10% of the experimental values. The simulations included parametric variations of the user input constants (e.g., convergence criteria, effect of nodalisation, etc.) and the effect of such variations is documented in reference 24. In the IET-8A simulations, no direct comparison with the experimental data was performed since the simulations were merely done to demonstrate the capability of IFCI in producing energetic steam explosions. The fine fragmentation model in IFCI for the explosion propagation is purely parametric, and the sensitivity of code calculated results to the user-input value for the fragmented particle size is provided in reference 25. The results showed that the maximum predicted pressure increased by a factor 3 by decreasing the particle size by about the same factor.

3.2.4.8 Assessment Status of TEXAS

A comparison of TEXAS simulations with results of several KROTOS experiments is provided in reference 26. The tests included KROTOS-21, KROTOS-26, and KROTOS-28. Even though the KROTOS-21 and KROTOS-26 simulations showed good agreement with the experimental data both in terms of the magnitudes of the dynamic pressures and the duration of the pressure pulse, the maximum pressures in the KROTOS-28 simulation were much smaller than the experimental data.

3.2.4.9 Assessment Status of PM-ALPHA/ESPROSE

The PM-ALPHA code predictions have been compared to the MAGICO experimental data (reference 11.), and in general good agreement between the code calculations and the MAGICO test results were obtained. Some comparison between the results of the ESPROSE calculations and the KROTOS-21, KROTOS-26, and KROTOS-28 tests were also reported in reference 11. The latest version of ESPROSE code is called ESPROSE.m, and a comparison with the KROTOS-28 experimental data are provided in reference 22. The code predictions show results consistent with the experimental data in terms of explosion propagation; however, differences in the magnitudes of the pressures at different locations are noted.

3.2.4.10 Assessment Status of ESCADRE

The different codes composing ESCADRE have been assessed on French and foreign experiments. The validation program is still on going and a specific effort will be devoted to the validation on the global PHEBUS-FPT experiment (CEA/Cadarache).

The release of fission product from the core (EMISS) has been validated on the HEVA, EMAIC and VERCORS experiments, performed in CEA/Grenoble. The validation of the behaviour of fission product in the primary circuit (SOPHAEROS) has been conducted using the TUBA and TRANSAT experiments (CEA/Grenoble) as well as the LACE (USA) experiment as regards aerosols and using the DEVAP experiment (CEA/Grenoble) as regards the FP vapour behaviour. Calculations on PBF-SFD (USA) and PHEBUS-FPT programs are planned to validate the core degradation models of ESCADRE (VULCAIN). Comparisons with the mechanistic code ICARE have also been performed and have shown a good agreement between results in the early phase of the core degradation.

The corium-concrete interaction model (WECHSL) has been widely validated on the BETA (Germany), SURC and ACE/MCCI (USA) experiments. As concerns the containment thermal-hydraulics and the hydrogen behaviour, calculations of PITEAS (CEA/Cadarache), BMC and HDR (Germany), LACE (USA) and PHEBUS-FPT have been performed. This made it possible to validate the mono-compartment model (JERICHO) as well as the German multi-compartment model (RALOC). The behaviour of the fission product behaviour models in containment (AEROSOLS-B2) have been assessed on the PITEAS, DEMONA (Germany) and LACE experiments.

The physical phenomena where an effort will be enhanced in the future will be the advanced stages of the core degradation , the vessel rupture and the direct containment heating.

Calculations of different accidental sequences on a French 900 MW PWR have been performed and comparisons of the results with MELCOR and MAAP results are on going.

3.2.4.11 Assessment Status of THALES/ART and THALES-2

Since THALES-2 is based on the THALES/ART code system, assessment of the old versions of THALES and ART should be valid for THALES-2.

Thermal hydraulics models of THALES in RCS were assessed by comparison with the RETRAN code which was considered to be sufficiently validated (reference 74.) and analysis of the TMI-2 data (reference 75.). Analyses of PBF-SFD scoping and 1-1 tests were made to assess models for heat transfer and oxidation (reference 74.). Two parallel calculations by THALES/ART and by hand calculations provided some verifications for models in THALES/ART for thermal-hydraulics in RCS(mass and energy balance) and FP release from fuel and transport in the containment (reference 76.).

Two series of code comparisons with modelling review and benchmark calculations were made: the first one (reference 77.) compared THALES/ART(both PWR and BWR versions) with STCP and MAAP and the second one (reference 78.) compared THALES-2 BWR version with STCP and MELCOR.

3.2.4.12 Assessment Status of ART Mod2

The models of radionuclide aerosol behaviour in ART Mod 2 have been validated by experiments such as NSPP (reference 79.), FALCON (reference 80.), STORM, WIND (reference 81.) and WAVE (reference 82.) experiments. Concerning the iodine chemistry models, PHEBUS/FP and ACE Phase B (reference 80.) data has been used for validation. The analytical capability of the code was also confirmed by comparing with the MELCOR and VICTORIA (reference 83.) codes.

3.2.4.13 Assessment Status of REMOVAL code

The assessment and validation work of the REMOVAL code has been performed mainly for aerosol behaviour analysis part. LACE LA1, LA2 and LA4 experiments were analysed with the code as the pre-test and post-test calculations (references 84. to 89.). Recently the code was employed for the intentional standard problem using VANAM-M3 test (ISP-37) to assess the model of hygroscopic aerosol behaviour in the multi-compartment containment. The code validation and improvement work is still on-going by the analyses of the aerosol re-entrainment experiments of ALPHA program and ACE phase B test.

3.2.4.14 Assessment status of JASMINE

Analysis with the JASMINE code (premixing module) was compared with previous experiments (references 90. to 91.). An isothermal solid particle-water mixing experiment and a hot particle-water mixing experiment were referred for the comparison.

The analytical results of the penetration velocity and shape of the isothermal particle cloud agreed well with the experiment. Also the envelope of the premixing region for the hot particle cloud with boiling was roughly agreed with the experiment. Validation of the premixing module against liquid-liquid mixing has not been published and validation of the propagation module has not been performed so far.

3.2.4.15 Assessment Status of MACRES

The MACRES code has been applied to a pre-test analysis of the PHEBUS FP test (FPT-0) and a post-test analysis of the FALCON FAL-ISP-1 test (ISP-34). The differences between the FALCON and PHEBUS circuits such as thermal-hydraulic flow conditions (laminar, turbulent and flow velocity) were shown to have a strong effect on the aerosol deposition. The inertial impacting was important only in the PHEBUS circuit, whereas thermophoresis is a dominant mechanism in both experiments. Although the gas flow rate in FALCON is low, the bend impaction was underestimated. This information was fed back to code improvement.

3.2.4.16 Assessment Status of MAPLE

Experimental validation of MAPLE has not been done. However, an application to a BWR gave some verification of the coding. An analyses of the pressure load for a BWR Mark-II containment (reference 93.) was made and gave important information such that the heat of oxidation from zircalloy in the debris may significantly increase the peak pressure in the DCH process and the heat transfer in the suppression pool would lower the pressure rise. An interesting result was that there is a peak in the pressure history in the containment due to consumption of steam by oxidation. This means that a dynamic heat and mass transfer model is necessary to properly predict the peak pressure (i.e. instantaneous heat and mass transfer may not be conservative).

3.2.4.17 Assessment Status of ATHLET-CD

At present the code is validated using data from CORA and PHEBUS-SFD tests. Validation on LOFT AM and PHEBUS-FP is underway.

3.2.4.18 Assessment Status of RALOC

For the validation of the different models a large amount of experiments have been analysed, as pre- and post-calculations. The code has successfully been used for 4 International Standard Problems and some other benchmark exercises.

3.2.4.19 Assessment Status of WECHSL

The WECHSL code in its present version was validated by the BETA, ACE, and SURC experiments.

3.2.4.20 Assessment Status of SageProc

The database of Gibbs energies for applications in MCCI simulations was qualified by successful applications to predict releases in MCCI release tests (ACE) and by phase diagram analyses with special emphasis of reproduction of experimentally determined liquidus and solidus temperature lines (ANL tests).

3.2.4.21 Assessment status of ESTER

ESTER couples together modules (codes) coming from EU organisations and the USNRC. Non-regression tests (reference 105.) have assured that the integrated modules have the same assessment status as the original codes, which include core degradation codes ICARE2 mod1 and KESS 1.7 with their native release models, circuit thermal-hydraulics and FP transport module VICTORIA92, containment thermal-hydraulics modules RALOC 2.2 and JERICHO, containment aerosol codes AEROSOLS and RALOC 2.2, pool scrubbing code BUSCA 1.1, core-concrete interaction code WECHSL/CALTHER and iodine chemistry codes IODE and INSPECT 1.0.

ESTER also contains links between the modules and a common database which create combined modules such as JERICHO-AEROSOLS-IODE or ICARE-VICTORIA. The latter have been applied for assessment purposes to the first PHEBUS FP experiment (reference 106.), (which covers most severe accident phenomena), the containment ISP 37 based on a VANAM experiment (reference 107.), and to aerosol deposition experiments of the STORM series (reference 108.). The individual modules are more convenient to use in their Esterase form (easier data transfer from one module to the next, common graphics and other facilities) but have the same level of physical modelling and hence the same assessment status as their parent codes. The combined modules, particularly ICARE-VICTORIA, allow the simultaneous detailed calculation of core degradation, circuit thermal-hydraulics (single-phase at present) and fission product release and transport in the RCS. This capability facilitates the investigation of coupled phenomena such as FP plate-out and revapourisation in the core and circuit, effects on transport of varying temperature and chemical composition of the carrier gas, and resuspension phenomena

3.2.4.22 Summary of Assessments

Tables 3.2-4 and 3.2-5 provide a summary of the status of experimental and/or benchmarking assessment of the separate phenomena and integrated PSA codes, respectively.

Table 3.2-4 Status of experimental and other benchmarking studies of separate phenomena computer codes

Models	SCDAP/ RELAP5	CONTAIN	HMS	IFCI	TEXAS	PM-ALPHA/ ESPROSE
Thermal/Hydraulics	+ (a)	+	NA	NA	NA	NA
Natural Circulation	+	+	NA	NA	NA	NA
Blowdown Model	+	NA	NA	NA	NA	NA
Cladding Oxidation	+	NA	NA	NA	NA	NA
Melting/Relocation	+	NA	NA	NA	NA	NA
Debris Quenching	--	NA	NA	+	+	+
Vessel Failure	+	NA	NA	NA	NA	NA
In-Vessel Fission Product Release	+	NA	NA	NA	NA	NA
HPME/DCH	NA	+	NA	NA	NA	NA
Light Gas Transport and Mixing	NA	+	+	NA	NA	NA
Hydrogen Combustion	NA	+	+	NA	NA	NA
Core Concrete Interactions	NA	+	NA	NA	NA	NA
Ex-Vessel Fission Product Release	NA	+	NA	NA	NA	NA
Retention in Pools	NA	+	NA	NA	NA	NA
Ex-Vessel Debris Cooling	NA	+	NA	+	+	+
Containment Fission Product Transport	NA	+	NA	NA	NA	NA
TMI-2 Accident	+	--	--	NA	NA	NA
PWR Applications	+	+	+	+	+	+
BWR Applications	+	Limited	+	+	+	+

(a) "+" indicates experimental benchmark studies have been performed and are reported in the code documentation and/or other supporting publications

Table 3.2-4 (Cont.) Status of experimental and other benchmarking studies of separate phenomena computer codes

Models	WECHSL	ATHLET-CD	RALOC	FIPLOC	SAGEPROC	IVA4
Thermal/Hydraulics	NA	+[a]	NA	NA	NA	+
Natural Circulation	NA	+	NA	NA	NA	+
Blowdown Model	NA	+	NA	NA	NA	+
Cladding Oxidation	NA	+	NA	NA	limited	NA
Melting/Relocation	NA	+	NA	NA	NA	+
Debris Quenching	NA	planned	NA	NA	NA	+
Vessel Failure	NA	NA	NA	NA	NA	+
In-Vessel Fission Product Release	NA	+	NA	NA	+	NA
In-Vessel transport	NA	+	NA	NA	NA	+
HPME/DCH	NA	NA	NA	NA	NA	+
Light Gas Transport and Mixing	NA	NA	+	+	NA	+
Hydrogen Combustion	NA	NA	+	NA	limited	+
Core Concrete Interactions		NA	NA	NA	+	NA
Ex-Vessel Fission Product Release	NA	NA	NA	+	+	NA
Retention in Pools	NA	NA	NA	+	NA	NA
Ex-Vessel Debris Cooling	limited	NA	NA	limited	NA	+
Containment Fission Product Transport	NA	NA	NA	+	NA	NA
TMI-2 Accident	NA	+	NA	NA	NA	NA
PWR Applications	+	+	+	+	+	+
BWR Applications	+	+	+	planned	planned	+

[a] "+" indicates experimental benchmark studies have been performed and are reported in the code documentation and/or other supporting publications

Table 3.2-5-1 Status of experimental and other benchmarking studies of integrated PSA and accident management computer codes

Models	ESCADRE
Thermal/Hydraulics	+
Natural Circulation	+
Blowdown Model	+
Cladding Oxidation	+
Melting/Relocation	planned
Debris Quenching	
Vessel Failure	
In-Vessel Fission Product Release	+
In-Vessel Transport	+
HPME/DCH	planned
Light Gas Transport and Mixing	+
Hydrogen Combustion	+
Core Concrete Interactions	+
Ex-Vessel Fission Product Release	+
Retention in Pools	
Ex-Vessel Debris Cooling	
Containment Fission Product Transport	+
Heat and Mass Transfer Models	+
TMI-2 Accident	
PWR Applications	+
BWR applications	

Table 3.2-5 -2 Status of experimental and other benchmarking studies of integrated PSA and accident management computer codes

Models	THALES/ART and THALES-2
Thermal/Hydraulics	+
Natural Circulation	-
Blowdown Model	(Moody's and an orifice models used)
Cladding Oxidation	+
Melting/Relocation	-
Debris Quenching	-
Vessel Failure	-
In-Vessel Fission Product Release	-
In-Vessel Transport	+
HPME/DCH	-
Light Gas Transport and Mixing	-
Hydrogen Combustion	-
Core Concrete Interactions	-
Ex-Vessel Fission Product Release	-
Retention in Pools	-
Ex-Vessel Debris Cooling	-
Containment Fission Product Transport	+
Heat and Mass Transfer Models	-
TMI-2 Accident	+
PWR Applications	+
BWR applications	+

3.2.5 References

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3.3 Documentation of the Use of Severe Accident Computer Codes in Selected Level 2 PSAs for Nuclear Power Plants

The referencing to publications in this section refers to the list in subsection 3.3.9.

3.3.1 Introduction

In section 3.2 severe accident computer codes are reviewed with respect to

- types of severe accident computer codes
- range of applicability
- assessment status of the codes

In this section the use of severe accident computer codes in the examined PSAs is discussed. This includes discussions whether correlation exists between the quantitative results at the various levels of the analysis and the use of certain severe accident computer codes. A more detailed discussion of the subject is provided in reference 7.

3.3.2 Severe Accident Computer Codes Used in the Examined PSAs

In tables 3.3-1 to 3.3-6 the severe accident computer codes used in the examined PSAs are compiled. A detailed description of the codes is provided in section 3.2.

The tables are organised by PSA relevant issues and phenomena; listed are the computer codes applied to the various items in the examined PSAs.

In the following, qualitative and quantitative aspects of the use of the codes are discussed. For each item a short discussion of the phenomenological context is provided.

3.3.3 Analysis of containment loads from in-vessel phenomena at PWR plants

3.3.3.1 Arrest of core melt progression, temperature induced hot leg/surge line/SGT rupture.

Core melt progression can be arrested if injection to the RPV can be re-established. Besides recovery of injection by operator action, relevant passive recovery scenarios involve

- failed steam generator feeding, high pressure in the reactor system resulting from operator failure to depressurise, inability to inject to the RPV, beginning core heat-up. Superheated steam flow is from the core through hot leg, surge line, pressuriser, out of the power operated relief valve (scenario 1).
- depressurisation by temperature induced passive failure of hot leg, surge line or steam generator heating tubes. Once the system is sufficiently depressurised, injection may be recovered. Whether or not RPV integrity can be maintained depends on the timing of the depressurisation.

Three different assessment bases exist in the examined PSAs:

1. In the NUREG-1150 analyses, the quantification of probabilities of passive depressurisation was based on an expert opinion elicitation process. The experts based their quantification on results of calculations with the computer codes MELPROG, TRAC/MELPROG, CORMLT/PSAAC, RELAP5/SCDAP and MAAP, as well as on evaluation of pertinent experiments.

   For scenario (1) a conditional probability of temperature induced hot leg failure of about 0.99 is obtained from the aggregation of the expert's quantifications. If in scenario (1) a seal LOCA is induced (scenario 2) the probability of hot leg failure is much lower, i.e. about 0.15. The figures are taken for all PWRs included in the NUREG-1150 analyses. The underlying probability distribution functions generated by the expert team for the cases without and with seal LOCA are shown in Figures 3.3.3.1-1 and 3.3.3.1-2.

In all other examined studies but Beznau HSK/ERI, the assessment is based on calculations with the MAAP code. For Sizewell B, the probabilities were based on the results of calculations using MAAP. This was supplemented by using SCDAP/RELAP5 for the station blackout sequence and account was taken of information from other studies including NUREG 1150. For a scenario similar to the one described above, the following quantifications are made:

- Robinson IPE: ~0.9 (point value)
- Maine Yankee IPE: ~0.75 (point value)
- Beznau PLG: ~0.99 (same data as in NUREG-1150)
- Ringhals 2: ~0.8
- Borssele: ~0.73
- Sizewell B: 0.9 (point value) which comprised 0.8991 for induced hot leg failure and $9x10^{-4}$ for induced SG tube failure

2. In the Beznau HSK/ERI study the assessment is based on MELCOR results, plant specific calculations with RELAP5/SCDAP and assessment of the TMI accident.

   The conditional probability for hot leg failure in scenario 1 was estimated to be 0.75, and 0.0 in scenario 2.

The estimates of the conditional probabilities of temperature induced depressurisation have significant impact on the fraction of core damage sequences remaining at high pressure at time of RPV bottom head failure. An exact correlation can not be established because of the differing shares of relevant sequences and scenarios.

In the examined PSAs, the following percentages of high pressure core melt sequences are reported:

Plant/PSA	Percentage of high pressure core melt sequences
Surry	3%
Zion	2%
Robinson	22%
Maine Yankee	16%
Beznau HSK/ERI	10%
Ringhals 2	12%
Borssele PSA-3	6%
Borssele PSA-97	?
Sizewell B	4%

3.3.3.2 In-vessel hydrogen generation

The amount of hydrogen generated in the in-vessel phase of core degradation and meltdown is proportional to the fraction of zirconium oxidised. The oxidation process is the result of complex interactions of thermo-hydraulic and chemical phenomena.
Basis for the assessment in the examined PSAs are:

- in the NUREG -1150 studies:

  Calculations with the program systems MELPROG, SCDAP, CORMLT, MAAP, MARCH, as well as evaluations of experiments and of the TMI-accident. A number of typical cases have been defined, characterised by various pressure ranges and time scales, with or without flooding of the core.
  Experts who had experience with several of the computer codes rated MAAP and MARCH lower than the others, because MAAP was considered to underestimate zirconium oxidation, and MARCH to overestimate it.

  The available information was assessed by a formalised expert opinion elicitation process. Subjective probability distribution functions for the amount of oxidised zirconium have been aggregated to one distribution function, which then was used in the quantification process.
  For the investigated cases, the median values of the aggregated distribution functions are between 30% and 50% zirconium oxidation.

- In the IPE-studies:

  Results of calculations with the program MAAP that were adapted to the special circumstances at the plant and evaluation of separate effect tests and of the TMI-accident.

  In the Robinson study, point values are being used, which are in good agreement with the median values in NUREG-1150. In the Maine Yankee study the point values are in the in the upper range of the distribution functions of NUREG 1150. They are generally higher that in the other PWR studies.

In the HSK/ERI analysis of the Beznau plant the assessment is based on MELCOR calculations. The range for the fraction of oxidised zirconium is 40 - 50%. As point value, 44% is used.

In the Ringhals 2 and Borssele analyses the assessment is based on MAAP calculations. The range for the fraction of oxidised zirconium is 30% - 52 %.

- For Sizewell B, the analysis of in-vessel hydrogen generation was done using MAAP, supplemented by the insights gained from other PSAs. For a PDS with an intact circuit in which hot leg failure does not occur, the point value for the in-vessel generation of hydrogen before vessel failure is that 45% of the Zirconium in the core will be oxidised.

The uncertainties in the modelling of in-vessel zirconium oxidation are large. However, a significant influence on early containment failure can only be identified in the Maine Yankee analysis. For this plant, with the fuel loaded at the time of the analysis, the ratio “amount of zirconium in the core/ containment volume” is much larger than for the other plants, thus making the containment vulnerable to hydrogen generation. At the other plants examined in this study, the threat from hydrogen burn is insignificant.

3.3.3.3 In-vessel steam explosion

In all studies, the assessment of the impact of in-vessel steam explosions is based on expert judgement. Input to the expert judgement are investigations performed by the USNRC Steam Explosion Review Group (NUREG 1116) (reference 1.), Corradini (reference 2.), Theofanus (reference 3.), Turland et al. (reference 4.). In all examined studies, the potential of in-vessel steam explosions to fail the containment is considered small relative to other containment failure modes. The quantified conditional probabilities for containment failure due to in-vessel steam explosions, given core melt, are in the range 10^{-3} to 10^{-2} for low pressure sequences, and in the range 10^{-4} to 10^{-3} for high pressure sequences. The Sizewell B analysis was based on the methodology developed by Theofanous. The conditional failure probabilities were estimated to be 6 10^{-4}, 2 10^{-3} and 4 10^{-4} for low, intermediate and high RCS pressure respectively. A recent reassessment by the USNRC Steam Explosion Review Group presented in reference 5. suggests that these figures are pessimistic estimates.

3.3.3.4 Bottom head failure

Important questions are: mode of bottom head failure (HPME, pour or dump); temperature, mass and fraction of metal in the ejected material.

- In the NUREG-1150 analyses the assessment is based on expert judgement. Input to the expert judgement are calculations with the codes MELPROG and MAAP and evaluations of the TMI accident. The investigations covered three cases:

  1. high pressure in the reactor system, no accumulator discharge, no upper head injection,
  2. intermediate pressure in the reactor system, no accumulator discharge, only partial upper head injection,
  3. low pressure in the reactor system, partial accumulator discharge, upper head injection functioning.

The aggregated answers of the experts are shown in the table below.

Mode of bottom head failure				
		Failure mode (fraction)		
Case	RCS pressure (psia)	HPME	Pour	Dump
1	2500	79%	8%	13%
2	2000	60%	27%	13%
3	200-1200	60%	27%	13%

- For Sizewell B, the CET node which considers failure of the reactor pressure vessel also considers the transport of the molten material through the instrument tunnel into the containment atmosphere. The event that is considered is the dispersal of at least 80% of the debris available out of the reactor cavity. The analysis considers the ability of the structures in the instrument tunnel to retain molten material.
- Other studies: Comparable information on this issue is not provided in the examined studies.

3.3.4 Analysis of containment loads from ex-vessel phenomena at PWR plants.

3.3.4.1 Loads at vessel breach

According to the quantifications made in the examined PSAs, the loads at vessel breach primarily result from „direct containment heating“ (DCH), which is a superposition of several physical phenomena, most notably:

- blowdown of steam and hydrogen,
- combustion of hydrogen,
- interaction of core debris with water on the containment floor and in the cavity,
- transfer of heat from dispersed debris to the containment atmosphere.

The parameters most important to DCH loads are:

- pressure in the reactor system at time of vessel breach (see the discussion in section 3.3.3.1),
- amount of unoxidised metal in the core (see the discussion in section 3.3.3.2),
- amount of ejected core debris,
- size of hole in the RPV,
- depth of water pool in the cavity,
- availability of containment spray.

The following code systems are used for the quantification of containment loads resulting from DCH:

- In NUREG-1150: CONTAIN, MAAP, HMC.
- In the IPE studies and in the studies for Beznau PLG, Ringhals-2, Borssele: MAAP.
- In the Beznau HSK/ERI study: SCDAP/RELAP5, CONTAIN, MAAP.
- In the Sizewell B analysis: MAAP, CONTAIN.

In the table below conditional probabilities related to DCH are compiled.

Conditional probabilities of DCH			
PSA	Conditional probability of high pressure in reactor system, given core damage	Conditional probability of containment failure due to DCH, given HPME	Conditional probability of containment failure due to DCH, given core damage
NUREG-1150	0.02-0.03	~ 0.2	0.004-0.006
IPE, Ringhals-2, Borssele	0.1-0.22	~ 0.1	0.01-0.025
Sizewell B POSR	0.04	0.25	0.01
Beznau HSK/ERI	< 0.1	0.13	0.013

The conditional probabilities are in good agreement for all large dry containments. For most plants DCH is the main contribution to early containment failure.

All presented results on DCH loads are accompanied by large uncertainties. but the impact of the uncertainties on the failure probabilities of the large dry containments is small. For a recent state-of-the-art report, see reference 6.

3.3.4.2 Ex-vessel steam explosion

In all studies, the assessment of the impact of ex-vessel steam explosions is based on expert judgement. Input to the expert judgement are investigations performed by the USNRC Steam Explosion Review Group (NUREG 1116) (reference 1.), Corradini (reference 2.), Theofanus /3/, Turland et al. (reference 4.). In all examined studies, the potential of ex-vessel steam explosions to fail the containment is considered small relative to other containment failure modes. The quantified conditional probabilities for containment failure due to ex-vessel steam explosions, given core melt, are in the range 10^{-4} to 10^{-3}. .A recent reassessment by the USNRC Steam Explosion Review Group presented in /5/ suggests that these figures are pessimistic estimates.

3.3.4.3 Ex-vessel generation of non-condensable gases

Non-condensable gases generated in the ex-vessel phase are:

- hydrogen resulting from unoxidised core debris reacting with water,
- hydrogen and carbon monoxide resulting from core debris/concrete interaction.

A parameter critical to the estimation of the amount of hydrogen generated from unoxidised core debris is the amount of zirconium in the core.

In the examined studies the following codes were used for the prediction of the amount of combustible gases generated:

- in NUREG-1150: CORCON,
- in the Beznau PLG study: MAAP, COMPACT,
- in the Beznau HSK/ERI study: MELCOR, COBURN,
- in all other studies: MAAP.

In the MAAP calculations it is assumed that core debris/concrete interaction is suppressed if the cavity is filled with water. This assumption is not made in other computer codes. Therefore, for situations with the cavity being filled with water, the ex-vessel generation of non-condensable gases is significantly lower for MAAP calculations than for other codes. Otherwise, predictions of the total amount of non-condensable gases generated - scaled to the amount of zirconium in the core - agree well among the various codes. However, significant uncertainties exist on the time history of generation of combustible gases.

3.3.4.4 Loads from combustion of hydrogen and carbon monoxide

Distinction is made between loads early in the accident that contribute to early containment failure, and loads late in the accident that contribute to late containment failure or - if applicable - to venting failure.
Codes used for the quantification of containment loads are MAAP, HCTOR, MELCOR and APPEAR-BURN.

Loads relevant to early containment failure depend on the amount of zirconium generated in the in-vessel phase (section 3.3.3.2). For all examined plants but Maine Yankee, the containment loads resulting from combustion of hydrogen in the early phase stay clearly below containment capacities, see typical examples, Figures 3.3.4.4-1 and 3.3.4.4-2. Therefore, the conditional probability, given core damage, of early containment failure due to combustion of gases is insignificant relative to other containment failure modes.

The majority of combustible gases is produced in the late accident phase. Thus, higher loads than in the early phase are seen in this phase, see a typical example, Figure 3.3.4.4-3, which indicates a high likelihood of containment failure due to combustion of gases at plants without venting capabilities.

3.3.4.5 Molten corium/containment structure interaction

Molten corium/containment structure interaction can lead to penetration by the core debris of the containment basemat. MAAP, CORCON and MELCOR are used for the analysis of this phenomenon. Basemat penetration is a significant contribution to containment failure at most plants, but its contribution to releases is generally low.

3.3.4.6 Containment structural response

Several well established code systems are available for quantification of containment load capacity, for example, NASTRAN, ABAQUS, DYNA3D, NEPTUNE.

In all studies, containment failure is described by cumulative probability functions. Probability of failure begins to rise from practically zero at about 5 bar. For the Surry and Ringhals-2 containments, probability of failure approaches 1 in the range 13-14 bar. In the Sizewell B analysis, the containment event tree analysis models three failure modes:

- design basis leakage,
- enhanced leakage, and
- gross failure.

Enhanced leakage is predicted to occur due to tearing of the liner at the equipment hatch and the personnel airlock. The median pressure at which this would occur is 120 psig - that is, 2.4 times the design pressure. The leakage area is estimated to be $5.4\ 10^{-4}\ m^2$.

The model of the containment considers a number of gross failure modes. The dominant one is hoop membrane failure at the cylindrical portion of the wall. The median pressure at which this would occur is 130 psig - that is, 2.6 times the design pressure. The leakage area is estimated to be $0.1\ m^2$.

Failure pressure probability distributions have been derived at a range of temperatures and this is found to be relatively insensitive to temperature. The analysis used the probability distribution for 300^oC since this represents the upper limit of the containment temperature for the sequences leading to a slow pressurisation."

For Zion NUREG-1150 and the IPE studies, probability of failure approaches 1 in the range 18-23 bar. In the examined IPE studies, higher pressures than in the other studies are required to fail the containment.

For illustration, see Figures 3.3.4.6-1 to 3.3.4.6-4.

For the steel containment of the Beznau plant, probability of failure approaches 1 in the range 9-10 bar, see Figure 3.3.4.6-5.

3.3.5 Analysis of source term issues for PWR plants

3.3.5.1 In-vessel fission product release, transport and retention

A large number of different code systems is used for predicting fission product release and transport inside the reactor system and to the containment.

For NUREG-1150 these are: STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, as well as evaluations of experiments. For Beznau HSK/ERI, MELCOR is used, and MAAP for all other plants.

The agreement among predictions of releases and retention inside the reactor system is reasonably good for noble gases and volatile fission products. For the refractory aerosols, there is more disagreement of predictions.

3.3.5.2 Scrubbing of fission products in water filled steam generator or in water pool

For mitigating releases from an unisolated defective steam generator, severe accident management procedures have been put in place at several plants. Essentially, the procedures are aimed at filling up a defective steam generator with a water column in which fission products are retained. For the quantification of the scrubbing effect The MAAP code system is used in the analyses for Ringhals-2 and Borssele, and the MELCOR code system in the Beznau HSK/ERI analysis.

The releases are predicted to be reduced by factors in the range 10 - 100. However, more clarification of this important issue is needed

In the Robinson and Maine Yankee IPEs, reductions of Cs releases by a factor 20-100 are reported for situations in which core debris on the containment floor is covered by an overlying water pool. However, this is to be attributed to two effects:

1. suppression of the core debris/concrete interaction (an assumptions that is made in MAAP, but not in the other codes),
2. fission product scrubbing by the water pool.

The information provided does not permit to differentiate between the two effects.

3.3.5.3 Fission product release, transport and retention inside containment

These issues are controlled by several phenomena which are not well understood, most notably thermophoresis, Brownian diffusion, aerosol agglomeration, aerosol plate out on surfaces, settling under influence of gravity. Most of these processes are governed by the aerosol particle size distribution which is not well known. Another important factor influencing deposition and plate out is the time history of the convection processes.

A large number of different code systems is in use for predicting fission product release and transport inside the containment.

For NUREG-1150 these are: STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, as well as evaluations of experiments. For Beznau HSK/ERI, MELCOR is used, and MAAP for all other plants.

Differences among predictions are large and difficult to compare and interpret.

3.3.5.4 Releases to the environment

For the calculation of source terms the XSOR suite of codes is used in the NUREG-1150 analyses, MELCOR and ERPRA are used in the Beznau HSK/ERI analysis, and MAAP in all other examined studies.

Suitable measures for comparing releases at different plants are the

- conditional probability of exceeding 10% Cs release, given one of the containment failure modes „early containment failure“ (ECF), „containment bypass“ (Bypass) or „isolation failure“ (ISF).

– conditional probability of exceeding 1% Cs release, given core damage.

In the table inserted below, these conditional probabilities are compiled for the plants examined in this section.

Conditional probabilities for Cs releases		
PSA	Conditional probability of exceeding 1% Cs release, given core damage	Conditional probability of exceeding 10% Cs release, given ECF or bypass or ISF
Surry, NUREG-1150	0.15	0.39
Zion, NUREG 1150 update	0.08	0.66
Maine Yankee IPE	0.06	0.19
Robinson IPE	0.1	0.23
Beznau HSK/ERI	0.03	0.05
Ringhals-2	0.01	0.03
Sizewell B	0.36	0.25, given LRCF mode 0.99, given late overpressurisation 0.22, given core damage

The calculated conditional probabilities reflect the combined effect of all issues discussed above, including the associated uncertainties. In view of the described differences, the agreement among plants with comparable retention capabilities is satisfactory.

Among the US PSAs, the IPE studies calculate lower large releases than the NUREG-1150 studies, but the source of the discrepancy is difficult to identify.

The Beznau, Borssele, Sizewell B and Ringhals-2 plants have implemented severe accident management procedures for filling up a defective steam generator with water, however, in the Sizewell analysis no credit is taken for this in the PSA. The retention capability of the water column is reflected by the significant reduction, relative to the other plants, of the conditional probability of exceeding 10% Cs release, given ECF or bypass or ISF.

Also, at the two plants high capacity filtered containment venting is available and severe accident management equipment and procedures are in place that permit to flood the containment using external water sources. This feature is reflected by the reduction, relative to the other plants, of the conditional probability of exceeding 1% Cs release, given core damage.

3.3.6 Analysis of containment loads from in-vessel phenomena at BWR plants

3.3.6.1 Arrest of core melt progression

Core melt progression can be arrested if injection to the RPV can be re-established. The relevant scenarios involve

- recovery of AC power if the accident was initiated by loss of AC power,
- operator actions to depressurise the reactor and align low pressure injection systems in situations with failed high pressure injection and failed automatic depressurisation.

3.3.6.2 In-vessel hydrogen generation

The amount of hydrogen generated in the in-vessel phase of core degradation and meltdown is proportional to the fraction of zirconium oxidised. The zirconium oxidation is the result of complex thermo-hydraulic and chemical interactions. Basis for the assessment in the examined PSA s are:

- In the NUREG -1150 studies:

  Calculations with the program systems MELPROG, SCDAP, CORMLT, MAAP, MARCH, BWRSAR and APRIL, as well as evaluations of experiments and of the TMI-accident. A number of typical cases have been defined, characterised by various pressure ranges and time scales, with or without flooding of the core.

  Experts who had experience with several of the computer codes rated MAAP and MARCH lower than the others: MAAP was considered to underestimate zirconium oxidation, and MARCH to overestimate it.

  The available information was assessed by a formalised expert opinion elicitation process. Subjective probability distribution functions for the amount of oxidised zirconium have been aggregated to one distribution function, which then was used in the quantification process.
  For the investigated cases, the median values of the aggregated distribution functions are between 10% and 25% zirconium oxidation.

- In the IPE-studies:

  Results of calculations with the program MAAP that were adapted to the special circumstances at the plant and evaluation of separate effect tests and of the TMI-accident.

  In the Browns Ferry and Perry studies, nominal values are being used, which are in good agreement with the median values of the distribution functions of NUREG-1150.

  In the HSK/ERI analysis of the Mühleberg plant the assessment is based on MELCOR calculations. The range for the fraction of oxidised zirconium is 21% - 25%.

3.3.6.3 In-vessel steam explosion

In all studies, the assessment of the impact of in-vessel steam explosions is based on expert judgement. Input to the expert judgement are investigations performed by the USNRC Steam Explosion Review Group (NUREG 1116) (reference 1.), Corradini (reference 2.), Theofanus (reference 3.), Turland et al. (reference 4.). In all examined studies, the potential of in-vessel steam explosions to fail the containment is considered small relative to other containment failure modes. The quantified conditional probabilities for containment failure due to in-vessel steam explosions, given core melt, are below 10^{-3}.

3.3.6.4 Bottom head failure

Important questions are: mode of bottom head failure (HPME, pour or dump); temperature, mass and fraction of metal in the ejected material.

- In the NUREG-1150 analyses the assessment is based on expert judgement. Input to the expert judgement are calculations with the codes BWRSAR and MAAP and evaluations of the TMI accident. Of the investigated cases, three are presented here:

1. high pressure in the reactor system, no injection,
2. low pressure in the reactor system, no injection
3. low pressure in the reactor system, LPI injection restored, no recriticality after LPI restoration.

The aggregated answers of the experts are shown in the table below.

Mode of bottom head failure				
		Failure mode (fraction)		
Case	RCS pressure	HPME	Pour	Dump
1	high	80%	20%	
2	low		75%	25%
3	low		74%	26%

- Other studies: Comparable information on this issue is not provided in the examined studies.

3.3.7 Analysis of containment loads from ex-vessel phenomena at BWR plants

3.3.7.1 Loads at vessel-breach

The pressure rise at vessel breach primarily results a superposition of several physical phenomena, most notably:

- blowdown of steam and hydrogen,
- combustion of hydrogen,
- interaction of core debris with water in the pedestal area,

- transfer of heat from dispersed debris to the containment atmosphere,
- impulse loads.

The parameters most important to DCH loads are:

- pressure in the reactor system at time of vessel breach,
- amount of unoxidised metal in the core,
- amount of ejected core debris,
- size of hole in the RPV,
- depth of water pool in the pedestal area,
- availability of containment spray.

The following code systems are used for the quantification of the pressure rise at vessel breach:

- In NUREG-1150: CONTAIN, MAAP, HMC.
- In the IPE studies and in the studies for Barsebäck, Forsmark 3: MAAP.
- In the Mühleberg PLG study: BWRSAR/CONTAIN.
- In the Mühleberg HSK/ERI study: MELCOR.

Loads due to vessel breach are among the dominant containment failure modes in all examined PSAs.

3.3.7.2 Ex-vessel steam explosion

In all studies, the assessment of the impact of ex-vessel steam explosions is based on expert judgement. Input to the expert judgement are investigations performed by the USNRC Steam Explosion Review Group (NUREG 1116) (reference 1.), Corradini (reference 2.), Theofanus (reference 3.), Turland et al. (reference 4.). In all examined studies, the potential of ex-vessel steam explosions to fail the containment is considered small relative to other containment failure modes. In all examined studies the quantified conditional probabilities for containment failure due to ex-vessel steam explosions, given core melt, are below 10^{-3}.

3.3.7.3 Ex-vessel generation of non-condensable gases

Non-condensable gases generated in the ex-vessel phase are:

- hydrogen resulting from unoxidised core debris reacting with water,
- hydrogen and carbon monoxide resulting from core debris/concrete interaction.

A parameter critical to the estimation of the amount of hydrogen generated from unoxidised core debris is the amount of zirconium in the core.

In the examined studies the following codes were used for the prediction of the amount of combustible gases generated:

- in NUREG-1150: CORCON,
- in the Mühleberg PLG study: MAAP,
- in the Mühleberg HSK/ERI study: MELCOR,
- in all other studies: MAAP.

In the MAAP calculations it is assumed that core debris/concrete interaction is suppressed if the debris is covered by a overlying pool of water. This assumption is not made in other computer codes. Therefore, for situations with the cavity being filled with water, the ex-vessel generation of non-condensable gases is significantly lower for MAAP calculations than for other codes. Otherwise, predictions of the total amount of non-condensable gases generated - scaled to the amount of zirconium in the core - agree well among the various codes. However, significant uncertainties exist on the time history of generation of combustible gases.

3.3.7.4 Combustion of hydrogen and carbon monoxide

Distinction is made between loads early in the accident that contribute to early containment failure, and loads late in the accident that contribute to late containment failure or - if applicable - to venting failure.

Codes used for the quantification of containment loads are MAAP, HCTOR, MELCOR and BWRSAR/CONTAIN.

Loads relevant to early containment failure depend on the amount of zirconium generated in the in-vessel phase (section 3.3.6.2). For examined plants with Mark III containments, which are not inerted, hydrogen combustion is a dominant contribution to containment failure. At all plants with inerted containment, the conditional probability, given core damage, of early containment failure due to combustion of gases is practically zero.

3.3.7.5 Molten corium/containment interaction

Molten corium exiting the reactor pressure vessel can erode the pedestal structure. In many Mark I containments the in-pedestal sump volume is too small to accommodate the molten core. Therefore, molten corium may spill over to the drywell floor and lead to drywell shell meltthrough and subsequent erosion of the concrete drywell structure.

In the NUREG-1150 studies, expert elicitation was performed for both pedestal erosion and drywell attack. The results provided by the experts differed widely. Research results that have since become available have removed some of the discrepancies, as they indicate that concrete erosion and drywell shell attack can be reduced by the presence of water. This issue was controversial at the time of the expert elicitation.
Input to the predictions by the experts were calculations with CORCON.

- In the Mühleberg PLG study, the CONTAIN is used, and in the Mühleberg HSK/ERI the TEXAS code is used.
- In all other studies, predictions are based on MAAP calculations.

- For US plants with Mark I containments there still may be significant contributions to early containment failure from molten corium/containment interaction. For all other plants, only insignificant contributions are reported.

3.3.7.6 Containment structural response

Several well established code systems are available for quantification of containment load capacity, for example, NASTRAN; ABAQUS, DYNA3D, NEPTUNE.

In all studies, containment failure is described by cumulative probability functions. For plants with Mark III containments, probability of failure begins to rise from practically zero at about 4-5 bar, and probability of failure approaches 1 in the range 7-8 bar. For plants with Mark I containments, probability of failure begins to rise from practically zero at about 7-8 bar, and probability of failure approaches 1 in the range 18-20 bar.

For illustration, see Figures 3.3.7.6-1 and 3.3.7.6-2.

3.3.8 Analysis of source term issues for BWR plants

3.3.8.1 In-vessel fission product release, transport and retention

A large number of different code systems is used for predicting fission product release and transport inside the reactor system and to the containment.

For NUREG-1150 these are: STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, as well as evaluations of experiments. For Mühleberg PLG, BWRSAR(RMA), CORSOR-M, and for Mühleberg HSK/ERI, MELCOR is used, For all other plants, MAAP is used.

The agreement among predictions of releases and retention inside the reactor system is reasonable for noble gases, CsI and CsOH. For Te and the refractory aerosols, there is considerable disagreement of predictions.

3.3.8.2 Scrubbing of fission products in water pool

In most accident sequences, gas mixtures containing aerosol particle pass through the pressure suppression pool where very effective scrubbing of fission products takes place. For the quantification of the scrubbing effect STCP with the SPARC module is used in NUREG-1150, MELCOR is used for Mühleberg HSK/ERI, and MAAP for the others. Reported fission product reduction factors typically are in excess of 1000.

3.3.8.3 Fission product release, transport and retention inside containment

These issues are controlled by several phenomena which are not well understood, most notably thermophoresis, Brownian diffusion, aerosol agglomeration, aerosol plate out on surfaces, settling under influence of gravity. Most of these processes are governed by the aerosol particle size distribution which is not well known. Another important factor influencing deposition and plate out is the time history of the convection processes.

A large number of different code systems is in use for predicting fission product release and transport inside the containment.

For NUREG-1150 these are: STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, as well as evaluations of experiments. For Mühleberg PLG, CONTAIN is used, for Mühleberg HSK/ERI, MELCOR and ERPRA are used, and MAAP for all other plants.

Differences among predictions are large and difficult to compare and interpret.

3.3.8.4 Releases to the environment

For the calculation of source terms the XSOR suite of codes is used in the NUREG-1150 analyses, MELCOR and ERPRA are used in the Mühleberg HSK/ERI analysis, and MAAP in all other examined studies.

Suitable measures for comparing releases at different plants are the:

- conditional probability of exceeding 10% Cs release, given one of the containment failure modes „early containment failure“ (ECF), „containment bypass“ (Bypass) or „isolation failure“ (ISF),
- conditional probability of exceeding 1% Cs release, given core damage.

In the table inserted below, these conditional probabilities are compiled for the plants examined in this section.

Conditional probabilities for Cs releases		
PSA	Conditional probability of exceeding 1% Cs release, given core damage	Conditional probability of exceeding 10% Cs release, given ECF or bypass or ISF
Peach Bottom, NUREG-1150	0.46	0.54
Grand Gulf, NUREG 1150	0.36	0.58
Browns Ferry IPE	0.25	0.22
Perry IPE	0.33	0.26
Mühleberg HSK/ERI	0.1	0.13
Barsebäck-2	0.13	0.36
Forsmark 3	0.0038	0.36

The calculated conditional probabilities reflect the combined effect of all issues discussed above, including the associated uncertainties. In view of significant differences in the quantifications in the various analysis steps, the agreement among plants with comparable retention capabilities is satisfactory.

Among the PSAs for US plants, the IPE studies calculate lower large releases than the NUREG-1150 studies, but the source for this discrepancy can only be speculated on. One reason may be the different treatment of core debris covered by water by the MAAP code in which core concrete interaction is practically suppressed in such situations, see section 3.3.7.3.

At Mühleberg and , Forsmark 3 (and all other Swedish plants), high capacity filtered containment venting is available and severe accident management equipment and procedures are in place that permit to flood the containment using external water sources. This feature is reflected by the reduction, relative to the other plants, of the conditional probability of exceeding 1% Cs release, given core damage.

Mühleberg has the lowest conditional probability of exceeding 10% Cs release, given a LRCF mode. In relation to US plants, this can be explained by the much larger in-pedestal sump volume which can easily accommodate the whole molten core. This practically eliminates the drywell attack problems seen in US plants with Mark I containment.

In relation to Forsmark 3 and Barsebäck which do not have the drywell attack problem, the low exceedance frequency at Mühleberg can be explained by additional retention in the strong reactor building which acts as a secondary containment and from which the release path for sequences bypassing the filter is through an outer water filled torus.

3.3.9 References

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2. Corradini and D. V. Swenson, "Probability of Containment Failure Due to Steam Explosions Following a Postulated Core Meltdown in an LWR," NUREG/CR-2214, SAND80 Sandia National Laboratories (1981).
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6. OECD/NEA/CSNI/R(96)25, December 1996.
7. W.F. Werner, Documentation of the use of severe accident computer codes in selected level-2 analyses for nuclear power plants. OECD/NEA HR/U3//96/854/AN/AMH, November 1996.

3.3.10 Figures

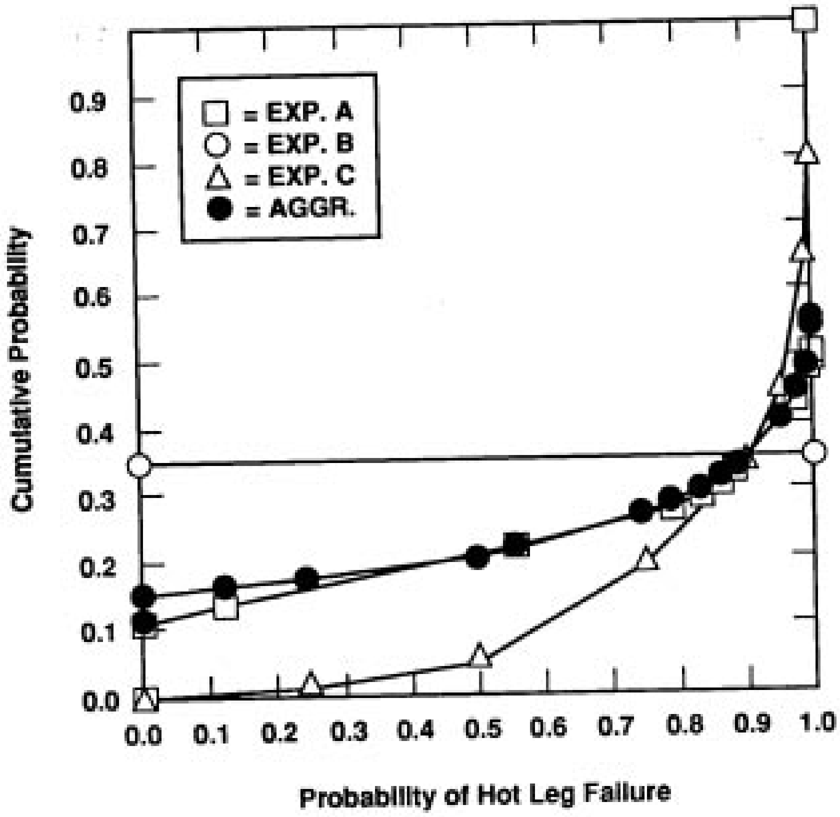

Figure 3.3.3.1-1. Induced hot leg failure in PWRs, scenario 1 (from NUREG/CR-4551)

Figure 1-1. Case 1: Induced Hot Leg Failure in PWRs.

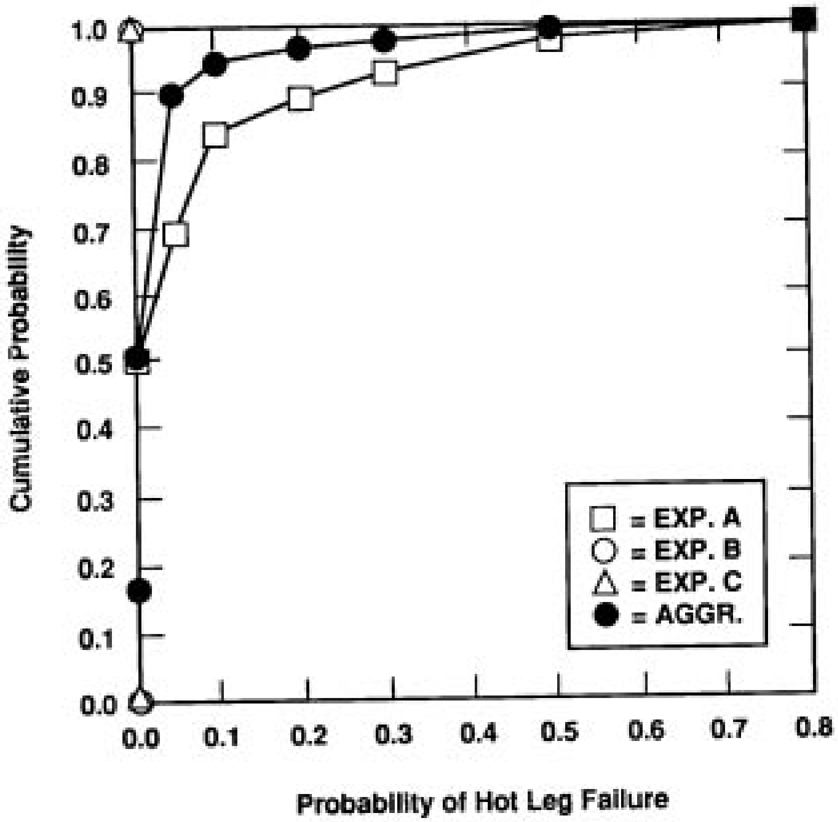

Figure 3.3.3.1-2. Induced hot leg failure in PWRs, scenario 2 (from NUREG/CR-4551)

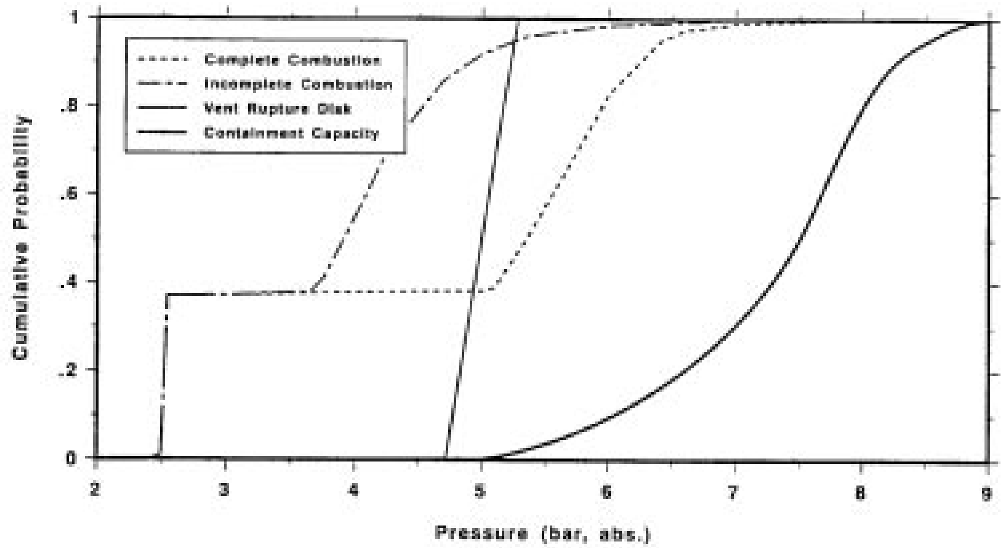

Figure 3.3.4.4-1. Distribution of loads due to hydrogen combustion at vessel breach for a high pressure scenario at Beznau (from ERI/HSK 94-301, Vol. 2)

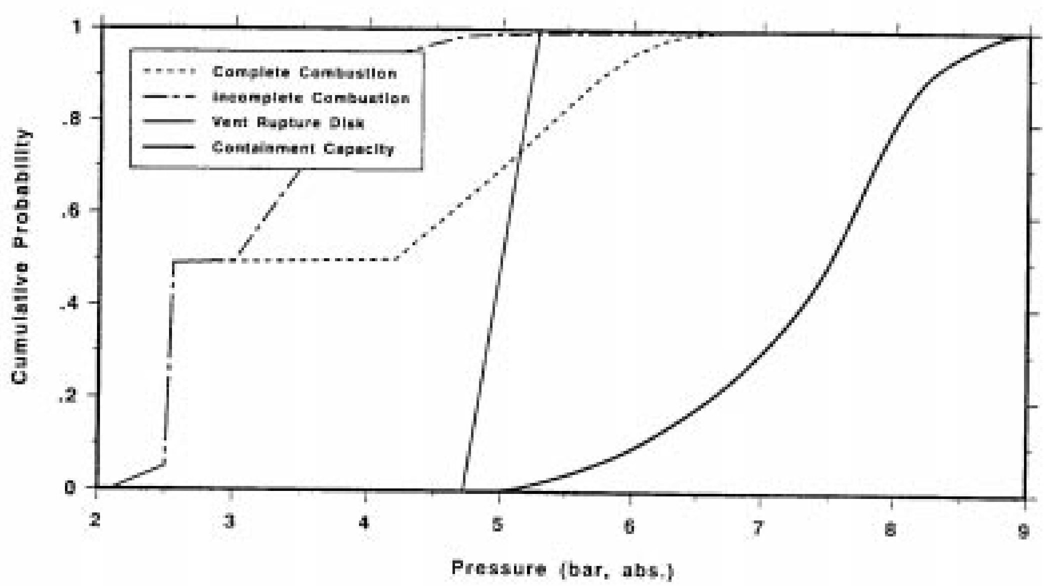

Figure 3.3.4.4-2. Distribution of loads due to hydrogen combustion at vessel breach for a low pressure scenario at Beznau (from ERI/HSK 94-301, Vol. 2)

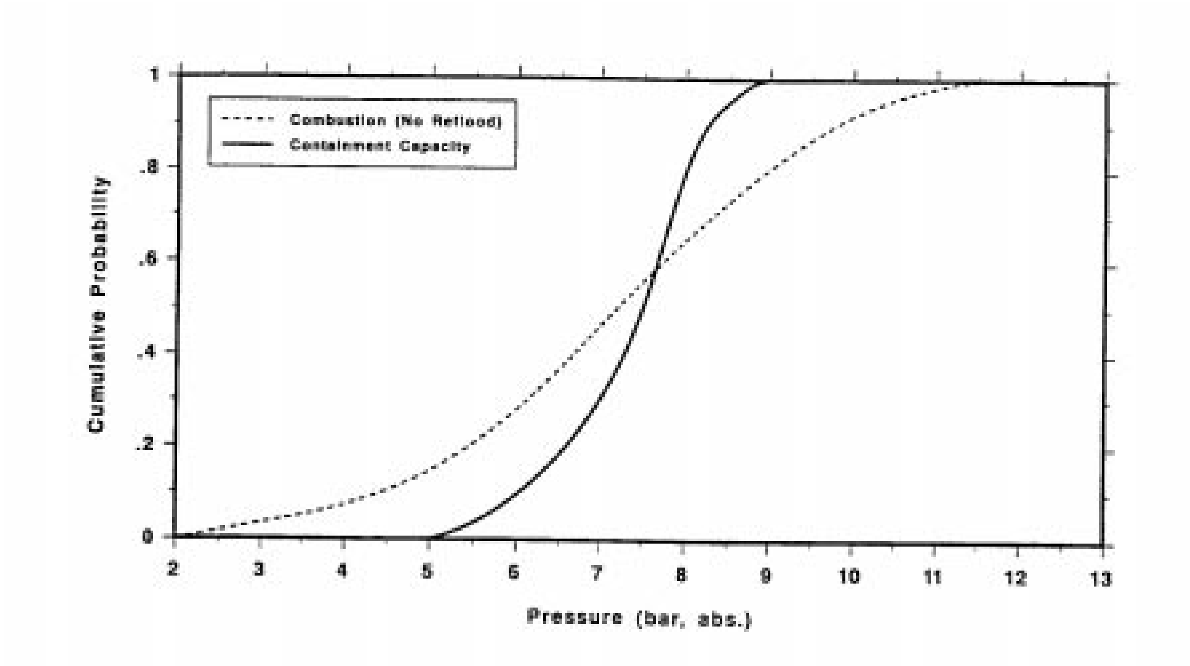

Figure 3.3.4.4-3. Distribution of loads due to hydrogen and carbon monoxide in the late stages of an accident at Beznau (from ERI/HSK 94-301, Vol. 2)

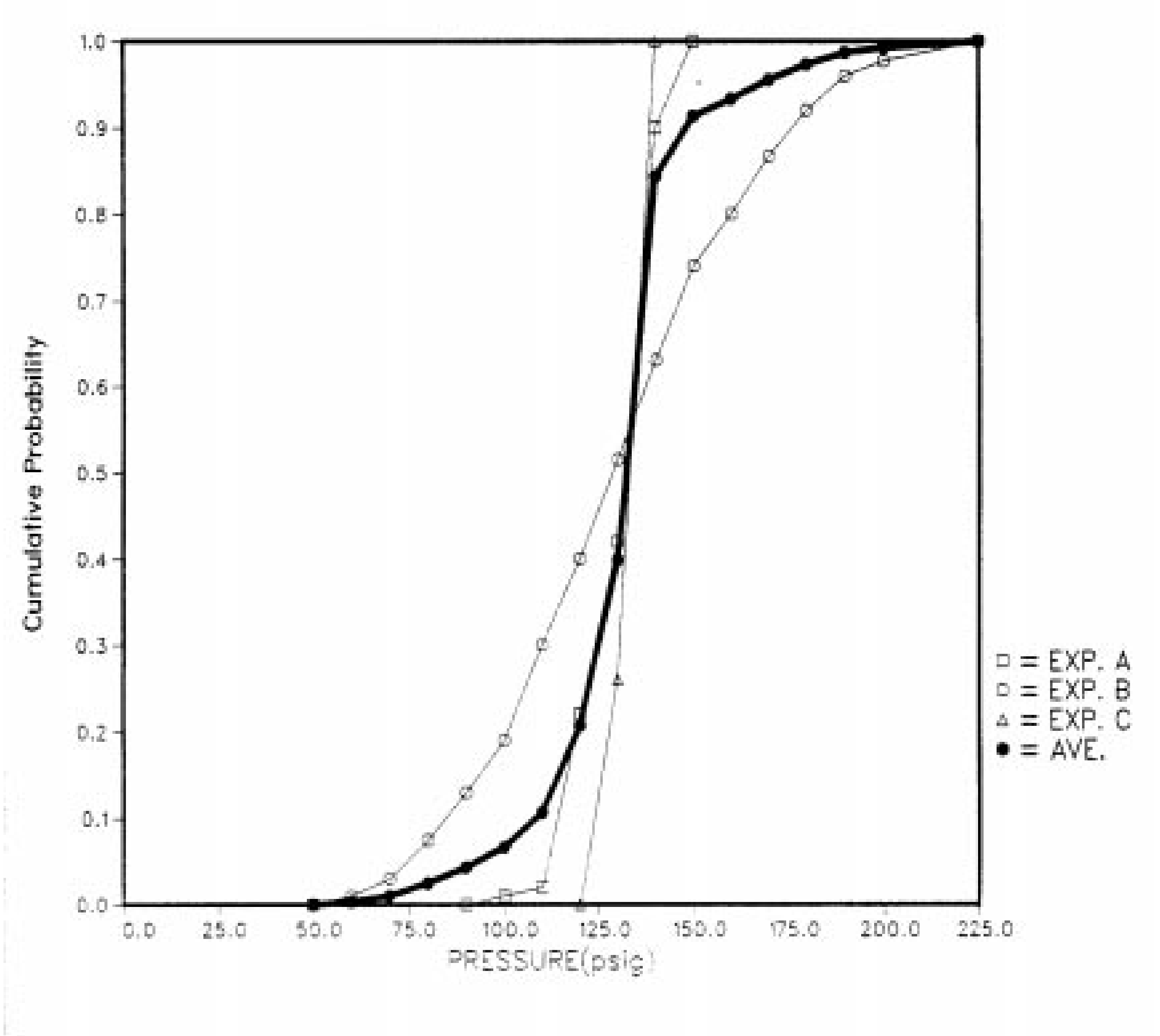

Figure 3.3.4.6-1. Probability distribution of the failure pressure for the Zion containment (from NUREG/CR-4551)

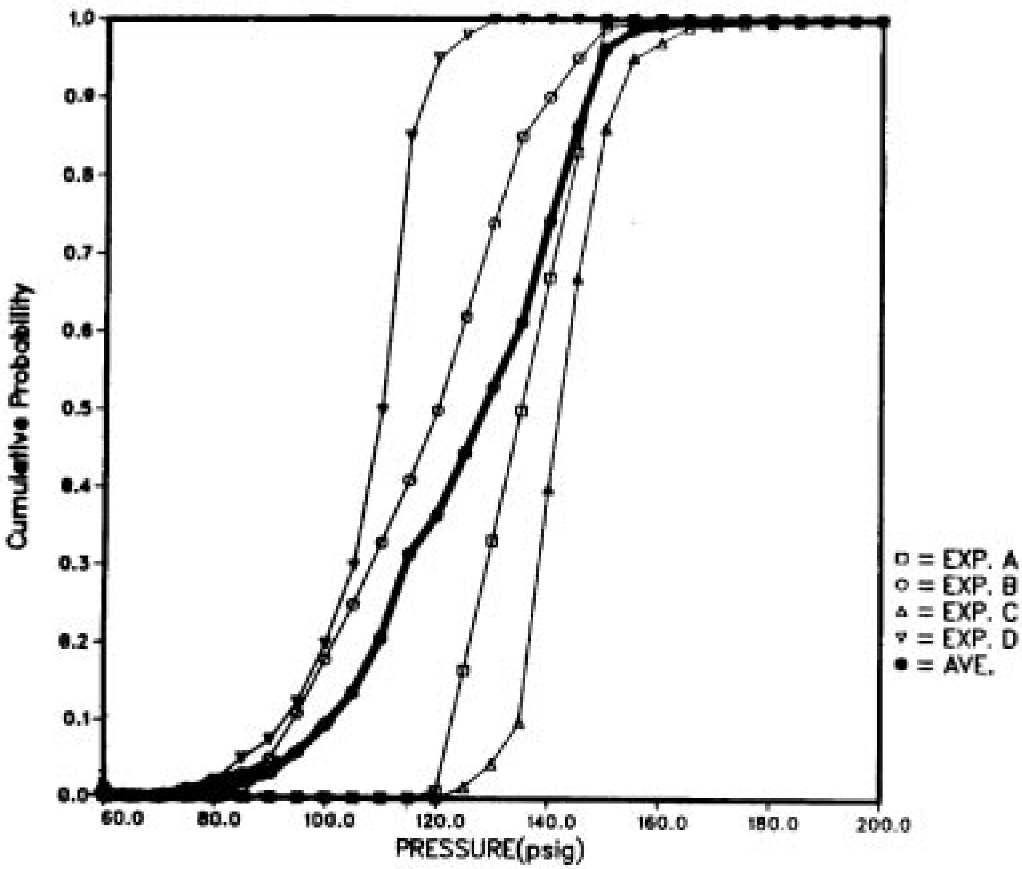

Figure 3.3.4.6-2. Probability distribution of the failure pressure for the Surry containment (from NUREG/CR-4551)

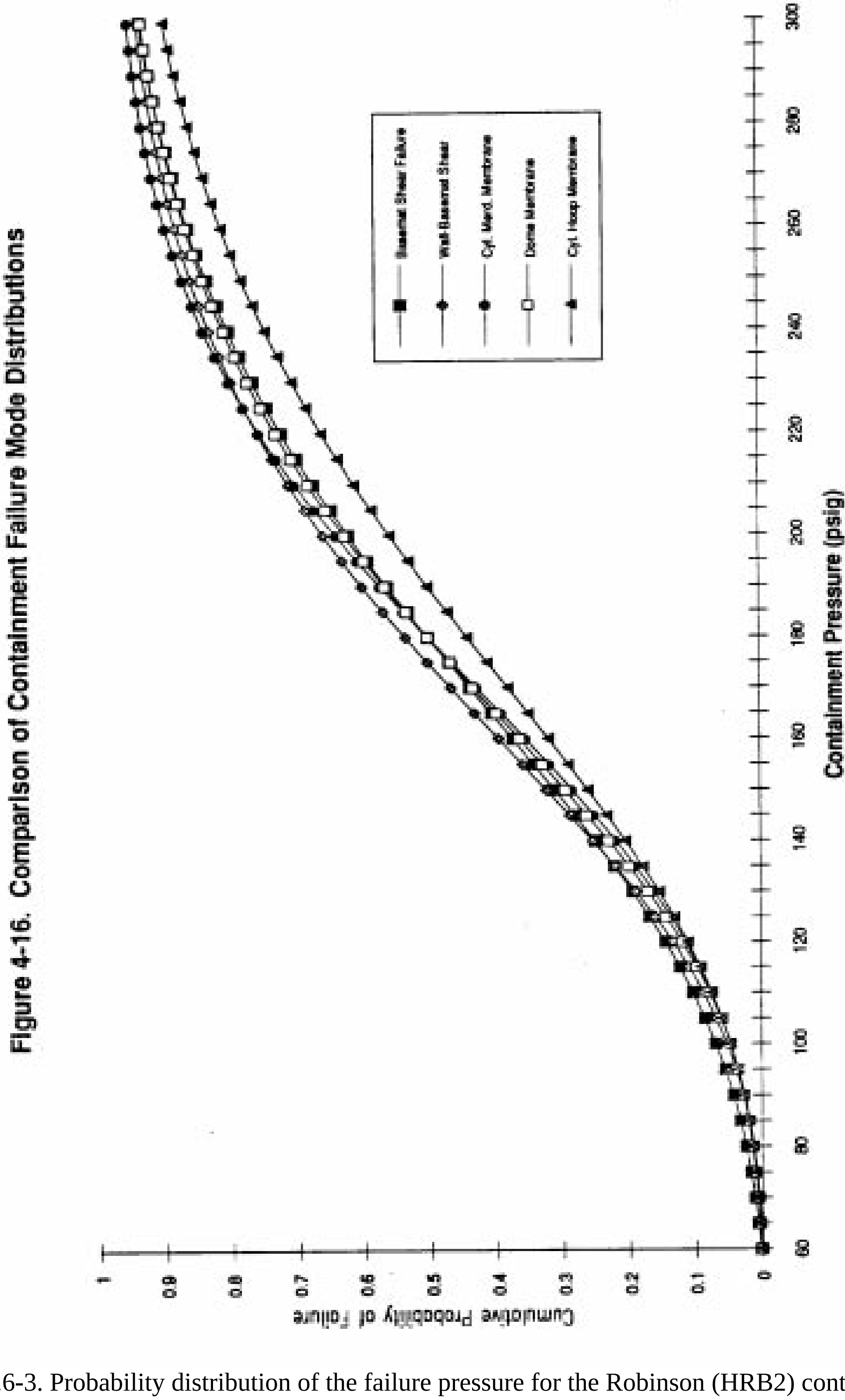

Figure 3.3.4.6-3. Probability distribution of the failure pressure for the Robinson (HRB2) containment (from Robinson IPE)

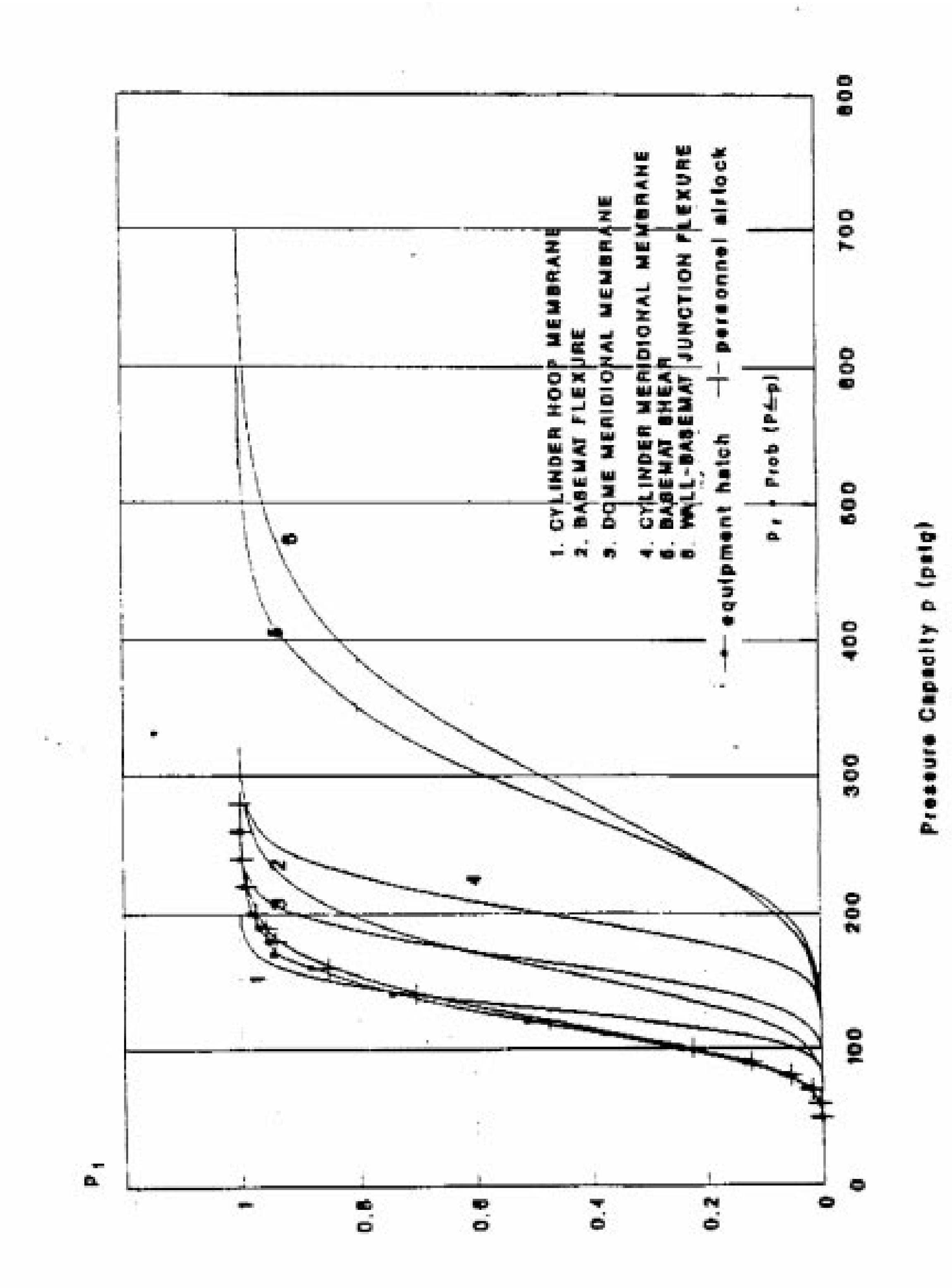

Figure 3.3.4.6-4 Probability distribution of the failure pressure for the Sizewell B containment

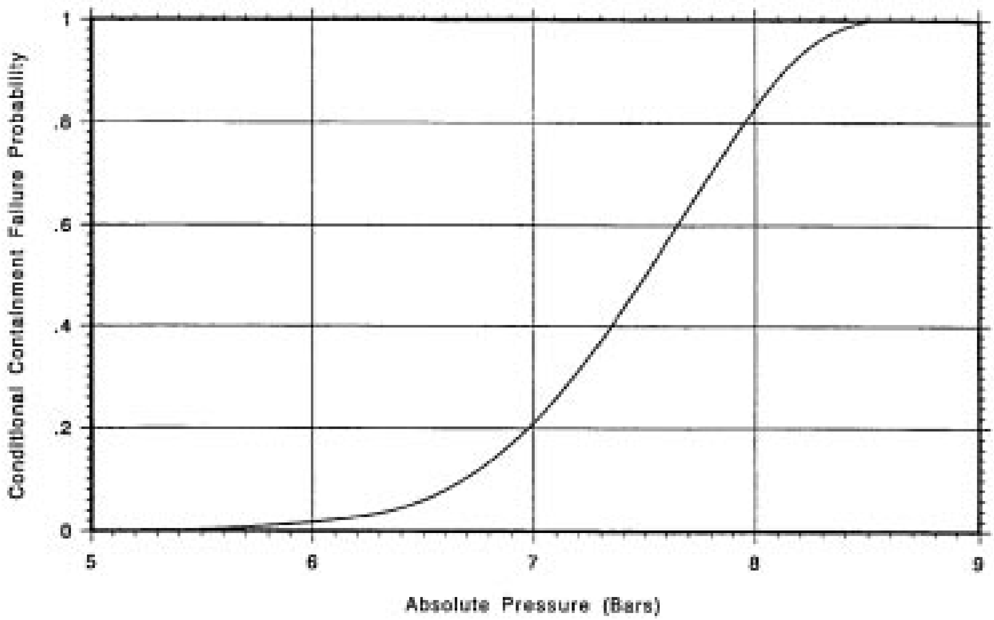

Figure 3.3.4.6-5. Probability distribution of the failure pressure for the Beznau containment (from ERI/HSK 94-301)

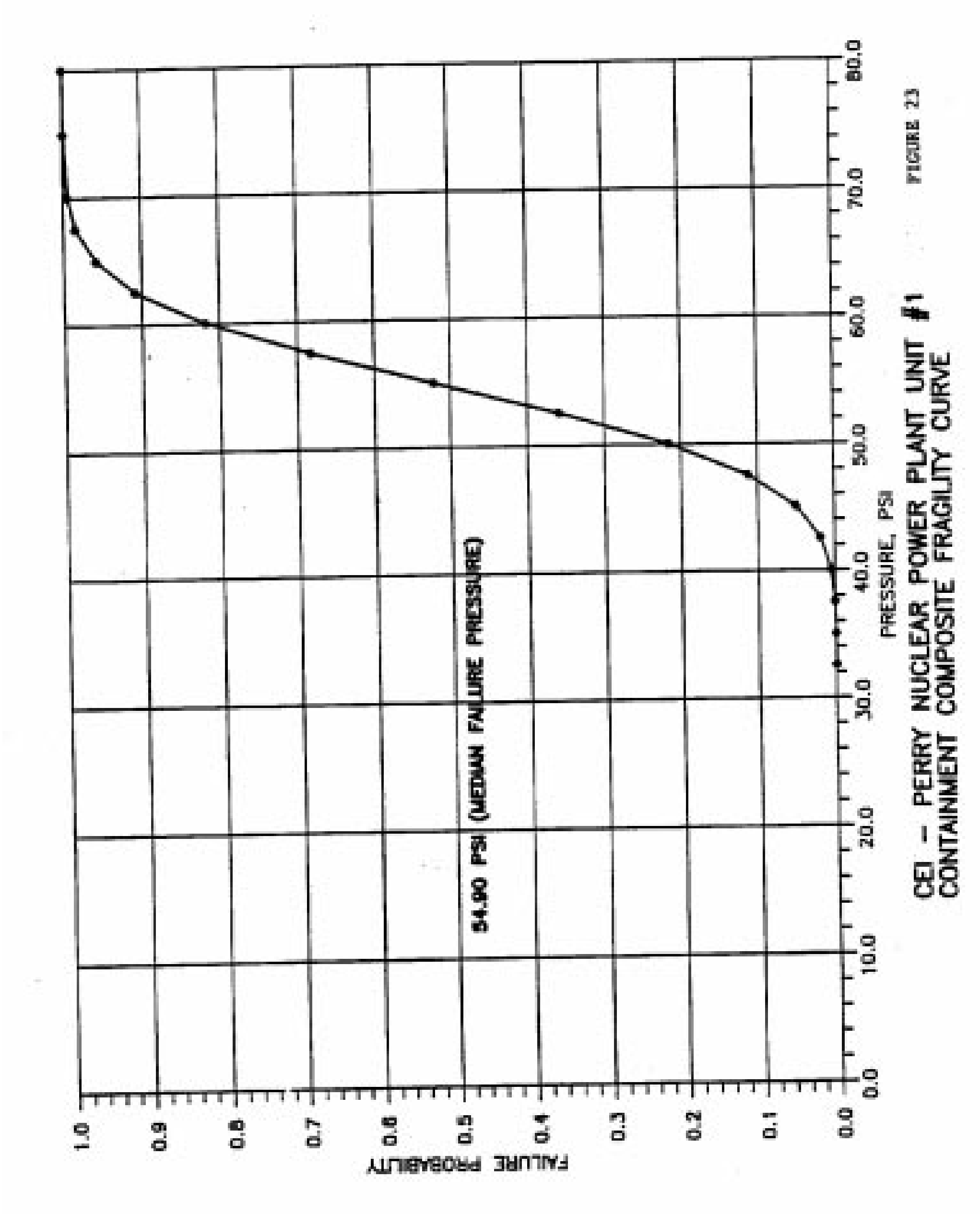

Figure 3.3.7.6-1. Probability distribution of the failure pressure for the Perry containment (from Perry IPE)

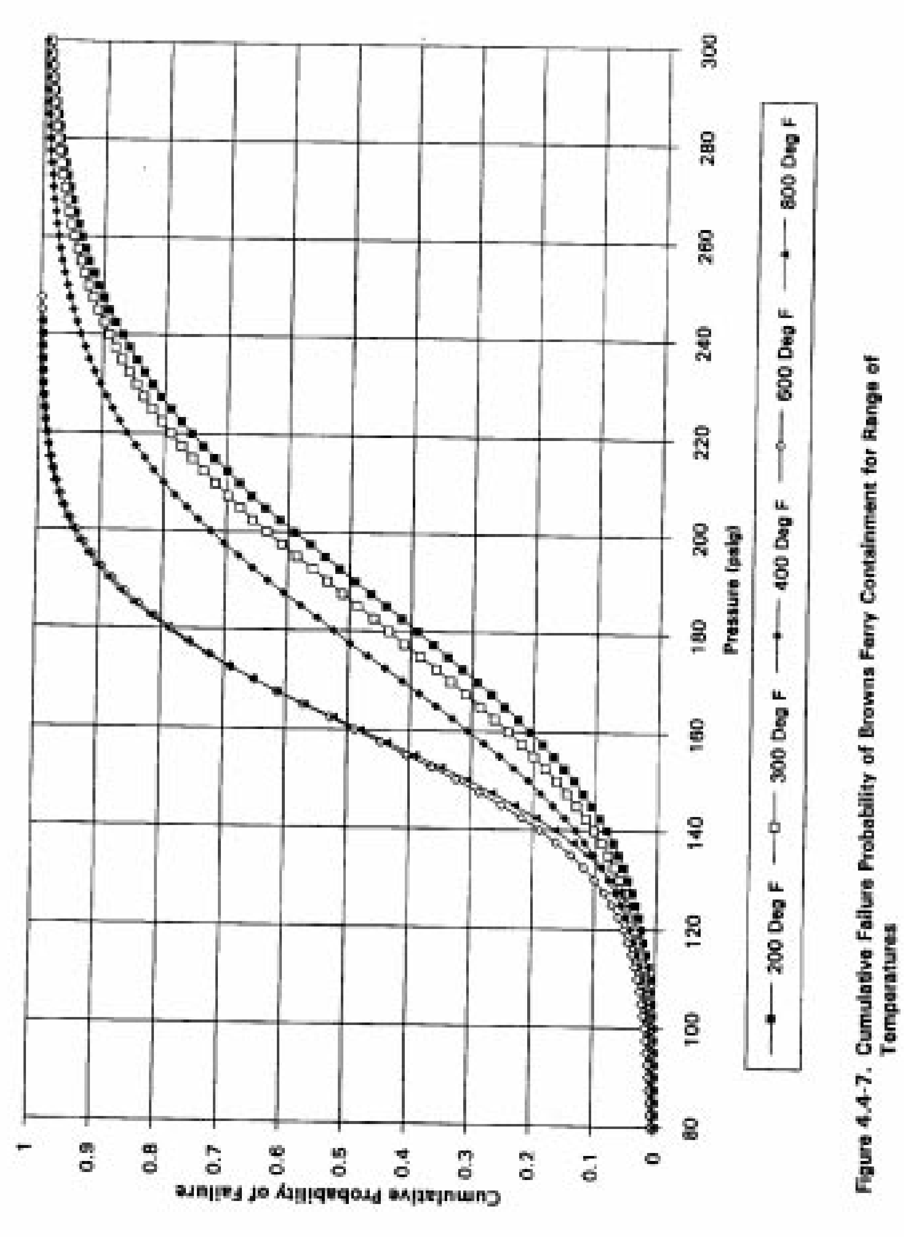

Figure 3.3.7.6-2. Probability distribution of the failure pressure for the Browns Ferry containment (from Browns Ferry IPE)

Table 3.3.3-1. Computer codes used in the examined PSAs for analysis of containment loads from in-vessel phenomena in PWRs

Phenomena PSA	Arrest of core melt progression	Temperature induced hot leg/surge line/SGT rupture	In vessel hydrogen generation	In-vessel steam explosion	Bottom head failure
NUREG-1150 Surry	Depends on • Level 1 systems analysis parameters • Likelihood of passive depressurisation mechanisms (column 2) • Rate of accident progression (MELCOR calculations)	Hot leg/surge line rupture MELPROG, TRAC/MELPROG, CORMLT/PSAAC, RELAP5/SCDAP, MAAP used by expert panel members	MELPROG SCDAP CORMLT MARCH MAAP used by expert panel members	Expert judgement, based on USNRC Steam Explosion Review Group (NUREG 1116) /1/, Corradini /2/, Theofanus /3/, Turland et al. /4/	MELPROG, MAAP analysis of TMI accident, used by expert panel members
NUREG-1150 Zion	Depends on • Level 1 systems analysis parameters • Likelihood of passive depressurisation mechanisms (column 2) • Rate of accident progression (MELCOR calculations	Hot leg/surge line rupture MELPROG, TRAC/MELPROG, CORMLT/PSAAC, RELAP5/SCDAP, MAAP used by expert panel members	MELPROG SCDAP CORMLT MARCH MAAP used by expert panel members	Expert judgement based on USNRC Steam Explosion Review Group (NUREG 1116) /1/, Corradini /2/, Theofanus /3/, Turland et al. /4/	MELPROG, MAAP analysis of TMI accident, used by expert panel members
Robinson IPE	Depends on • Level 1 systems analysis parameters • Likelihood of passive depressurisation mechanisms (column 2)	MAAP	MAAP	Expert judgement	MAAP
Maine Yankee IPE	Depends on • Level 1 systems analysis parameters • Likelihood of passive depressurisation mechanisms (column 2)	MAAP	MAAP	Expert judgement	MAAP
Beznau PLG	MAAP	MAAP	MAAP	Expert judgement	MAAP
Beznau HSK/ERI	MELCOR	SCDAP/RELAP5, NUREG-1150 results, TMI evaluation	MELCOR	Expert judgement, based on work by Theofanus /3/ Corradini/2/, and HSK sponsored analyses.	MELCOR
Ringhals 2	MAAP	MAAP	MAAP	Expert judgement, based on work by Theofanus /3/ Corradini (2)	MAAP
Borssele	MAAP	MAAP	MAAP	Expert judgement	MAAP
Sizewell B	MAAP	MAAP, SCDAP/RELAP5 Larson-Miller creep rupture model	MAAP CONTAIN	Expert judgement Theofanus method	MAAP CORDE

Table 3.3.4-1. Computer codes used in the examined PSAs for analysis of containment loads from ex-vessel phenomena, PWRs

Phenomena PSA	Loads at vessel breach	Ex-vessel steam explosion	Ex-vessel generation of non-condensable gases	Combustion of hydrogen and carbon monoxide	Molten corium /containment structure interaction	Containment structural response to pressurisation
NUREG-1150 Zion	CONTAIN, MAAP	Expert judgement based on NUREG 1116 /1/, Corradini /2/, Theofanus /3/, Turland et al. /4/		Expert judgement, HECTR		Structural analysis codes
Robinson IPE	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Maine Yankee IPE	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Beznau PLG	MAAP	Expert judgement	MAAP	COMPACT, MAAP		Structural analysis codes
Beznau HSK/ERI	SCDAP/RELAP5, CONTAIN, MAAP	Expert judgement, based on Theofanus /3/ Corradini /2/, and HSK sponsored analyses.	MELCOR	MELCOR, ERPRA-BURN	MELCOR	Structural analysis codes
Ringhals 2	MAAP	Expert judgement based on NUREG 1116 /1/, Corradini /r2/, Theofanus /3/, Turland et al. /4/	MAAP	MAAP	MAAP	Structural analysis codes
Sizewell B	MAAP CORDE	Expert judgement	MAAP	MAAP CONTAIN	MAAP	Structural analysis codes Scale model test
Borssele	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes

Table 3.3.5-3. Computer codes used in the examined PSAs for analysis of source term issues, PWRs

Phenomena PSA	In-vessel fission product release, transport and retention	Scrubbing in water filled steam generator or in water pool	Fission product release, transport and retention inside containment	Environmental release
NUREG-1150 Surry	STCP (CORCON-MOD2/ VANESSA, NAUA), MAAP, ASTEC, CONTAIN, experiments,		STCP (ICEDF, CORCON-MOD2/ VANESSA, NAUA), MAAP, ASTEC, CONTAIN, experiments,	SURSOR
NUREG-1150 Zion	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, experiments,		STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2, VANESSA, NAUA, experiments,	ZISOR
Robinson IPE	MAAP	MAAP	MAAP	MAAP
Maine Yankee IPE	MAAP		MAAP	MAAP
Beznau PLG	MAAP		MAAP	MAAP
Beznau HSK/ERI	MELCOR	MELCOR	MELCOR	MELCOR, ERPRA
Ringhals 2	MAAP	MAAP	MAAP	MAAP
Borssele	MAAP	MAAP	MAAP	MAAP
Sizewell B	MAAP	MAAP	MAAP	MAAP

Table 3.3.6-1. Computer codes used in the examined PSAs for analysis of containment loads from in-vessel phenomena, BWRs

Phenomena PSA	Arrest of core melt progression	In vessel hydrogen generation	In-vessel steam explosion	Bottom head failure
NUREG-1150 Peach Bottom		MELPROG, SCDAP, CORMLT, MAAP, MARCH, BWRSAR and APRIL	Expert judgement based on USNRC Steam Explosion Review Group (NUREG 1116) /1/, Corradini /2/, Theofanus /3/, Turland et al. /4/	BWRSAR Expert judgement
NUREG-1150 Grand Gulf		MELPROG, SCDAP, CORMLT, MAAP, MARCH, BWRSAR and APRIL	Expert judgement based on USNRC Steam Explosion Review Group (NUREG 1116) /1/, Corradini /2/, Theofanus /3/, Turland et al. /4/	BWRSAR Expert judgement
Browns Ferry IPE		MAAP	Expert judgement	MAAP
Perry IPE		MAAP	Expert judgement	MAAP Expert judgement,
Mühleberg PLG	BWRSAR	BWRSAR/CONTAIN	Expert judgement	MAAP
Mühleberg HSK/ERI		MELCOR	Expert judgement	Expert judgement
Forsmark 3		MAAP	Expert judgement	MAAP
Barsebäck 1/2		MAAP	Expert judgement	MAAP

Table 3.3.7-1. Computer codes used in the examined PSAs for analysis of containment loads resulting from ex-vessel phenomena, BWRs

Phenomena PSA	Loads at vessel breach	Ex-vessel steam explosion	Ex-vessel generation of non-condensable gases	Combustion of hydrogen and carbon monoxide	Molten corium/containment interaction	Containment structural response
NUREG-1150 Peach Bottom	CONTAIN, MAAP, HMC	Expert judgement	CORCON	Expert judgement, HECTR, MELCOR	CORCON	Structural analysis codes
NUREG-1150 Grand Gulf	CONTAIN, MAAP, HMC	Expert judgement		Expert judgement, HECTR, MARCH-HECTR, MELCOR		Structural analysis codes
Browns Ferry IPE	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Perry IPE	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Mühleberg PLG	BWRSAR/CONTAIN	Expert judgement	MAAP	BWRSAR/CONTAIN	CONTAIN	Structural analysis codes
Mühleberg HSK/ERI	MELCOR	TEXAS, expert judgement	MELCOR	MELCOR	MELCOR, TEXAS	Structural analysis codes
Forsmark 3	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes
Barsebäck 1/2	MAAP	Expert judgement	MAAP	MAAP	MAAP	Structural analysis codes

Table 3.3.8-1. Computer codes used in the examined PSAs for analysis of source term issues, BWRs

Phenomena PSA	In-vessel fission product release and retention	Scrubbing in suppression pool	Ex-vessel fission product release, transport and depletion inside containment	Environmental release
NUREG-1150 Peach Bottom	STCP (CORCON-MOD2/ VANESSA, NAUA), MAAP, ASTEC, CONTAIN, experiments,	STCP	STCP (SPARC, CORCON-MOD2/ VANESSA, NAUA), MAAP, ASTEC, CONTAIN, experiments,	PBSOR
NUREG-1150 Grand Gulf	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2/VANESSA, NAUA, experiments,	STCP	STCP, MAAP, ASTEC, CONTAIN, ICEDF, CORCON-MOD2/VANESSA, NAUA, experiments,	GGSOR
Browns Ferry IPE	MAAP	MAAP	MAAP	MAAP
Perry IPE	MAAP	MAAP	MAAP	MAAP
Mühleberg PLG	BWRSAR (RMA); CORSOR-M	BWRSAR	CONTAIN	MAAP
Mühleberg HSK/ERI	MELCOR	MELCOR, ERPRA	MELCOR, ERPRA	MELCOR, ERPRA
Forsmark 3	MAAP	MAAP	MAAP	MAAP
Barsebäck 1/2	MAAP	MAAP	MAAP	MAAP

4. SEVERE ACCIDENT MANAGEMENT

The referencing to publications in this section refers to the list in subsection 4.6.

4.1 Background and Objectives

Main objective of Severe Accident Management (SAM) at nuclear power plants is to provide hardware and establish the organisational and procedural framework for coping with accidents beyond design basis. Principal aim of SAM is to return the plant to a stable and controlled state and to minimise plant internal and off-site consequences.

The provisions to be made at a plant for establishing SAM capabilities and their probabilistic background will be described in this chapter.

4.2 Evolution of an accident from the operators perspective

Accident management has become one of the key levels of protection which, together, constitute an overall strategy for utilising safety measures and features of nuclear power plants (defence in depth concept). Defence in depth is effected primarily by means of successive barriers preventing the release of radioactive material (fuel matrix, cladding, primary coolant boundary and containment) and the protection of these barriers by a set of measures, including conservative design, quality and status surveillance, operating limits and highly reliable safety systems. Should vital safety functions or barriers fail, then accident management has to provide the final level of protection. Two types of strategies are distinguished:

- Achieving a stable and controlled state of the reactor coolant system (AM for prevention of core damage). Typically, this is accomplished by using alternative equipment to back-up failed systems or safety functions.
- Protecting the containment against excessive severe accident loads; minimisation of fission product release to the environment if containment integrity can not be maintained (SAM for mitigation of accident consequences). Typically, hardware that has been installed specifically for SAM purposes is used for this type of strategies

Specific instructions must be provided for using equipment, also non safety-related systems, in severe accidents. As the numerous combinations of failures result in a very large number of different scenarios an efficient approach is the development of general symptom based guidance that tries to account for as many as possible representative generic situations.

Given that hardware capabilities are available for performing SAM, the first action that operators and the supporting technical staff must take in an accident situation that has gone beyond their EOPs is

- assess the current status of core cooling, reactivity, RCS and containment status, fission products release status,
- depending on the specific situation and following pre-planned procedures (some kind of diagnosis flow chart) choose an appropriate strategy,
- assess before its implementation the positive and negative impacts,
- select the most suitable strategy.

Once this strategy has been implemented it is of great importance to keep track of the plant state, in order to verify that it evolves as expected and will lead to a stable and controlled end state. This may be difficult because of the potential for further equipment and instrumentation failures. For this reason it is strongly recommended to have well analysed and documented pre-planned strategies available at the plant.

Plant specific details must be taken into account in the identification and selection of the most suitable SAM actions. Plant details vary quite widely between different types of nuclear power plants (e.g. type of fuel, pressure of the coolant, size and strength of the containment, safety systems; and alternative systems for back-up of safety systems). Therefore, SAM-strategies or single SAM-measures can not simply be transferred between plants.

Whenever plant protection systems are actuated, operators follow predefined procedures which are set out in documents designated, for example, as EOPs. These documents are used to verify the automatic operation of safety systems, to diagnose the situation by following a predefined logical process for selecting the appropriate procedure and to take action as prescribed by this specific procedure. It is important that these procedures provide systematic and adequate guidance from the beginning of an event or transient. This enables operators to initiate the appropriate response without having to rely on memorised responses when facing a complex situation. Effective procedures should assist operators in focusing attention on the most important information and developments; they must help to prevent or overcome possible confusion caused by numerous simultaneous alarms and prevent misdirection of attention to matters of lesser importance.

In the past the general approach was to provide a set of EOP's covering a series of individual initiating events taken into account in the design.

In recent years, experience with plant operations and specially the TMI-2 accident have shown the need to take account of more complex situations in which it is difficult or even impossible to diagnose the initiating event and to take the prescribed corrective action to regain control of the plant. In order to cover a broader range of accidents, and to take into account errors in diagnosis or inadequacy of operator interventions, more general EOP's have been developed. These procedures are based on the idea that it is generally not necessary to know the chronology of the past events and actions that have led to an actual situation in order to be able to identify the required actions. This approach is based on a set of generic symptom (or function or state) oriented procedures with only a few safety objectives to be fulfilled. The symptoms and plant parameters associated with them have to be identified for the different mechanisms that could occur in the plant. Possibly, several strategies can be identified for preventing or mitigating the mechanism that causes the safety function challenges

It is generally thought that symptom oriented guidance and procedures should be available for SAM. Also, an effective organisational structure is required: the "decision making process" needs to be clearly stated, good communications between operators and the supporting technical staff and training of the organisation staff involved in the SAM.

In specific cases when a vulnerability is detected that cannot be covered by available hardware, it may be necessary to implement suitable modifications.

4.3 Safety objectives for the development of SAM Guidance

The criterion for the development of the SAMG is the preservation or achievement of a stable and controlled state of core, containment and fission products in a severe accident.
The safety objectives related to the above items are the following:

Core:

- reactivity control
- availability of heat sink
- inventory in RCS and containment

The core is in a stable and controlled state when its temperature is below the point where chemical or physical changes can take place and when a "long term" heat sink is available.

Containment :

- pressure in RCS
- inventory in steam generators and containment
- availability of heat sink
- containment isolation
- hydrogen control

The containment is in a stable and controlled state when there is a way to transfer all the energy released to the containment to a "long term" heat sink, the containment is correctly isolated and structurally intact, and the conditions in the RCS and the containment will not lead to sudden changes.

Fission products :

- control of the releases
- minimise the release to the environment

The fission products are considered in a stable and controlled state when the containment integrity is maintained, any leakage through the containment barrier can be controlled and the inventory of volatile fission products in the containment atmosphere can be reduced.

In Table 4.3-1 the top-level safety objectives for severe accident management are related to the plant conditions that must be monitored in order to achieve the objectives.

Table 4.3-1 Plant conditions to return to a controlled and stable state

SAFETY OBJECTIVE	PLANT CONDITION
Core in stable, controlled state	Core cooling
	Reactivity
Containment in stable, controlled state	Heat sink in containment
	Inert Containment
	Low pressure at RCS
Fission products release in stable and controlled state	Inventory in steam generators
	Containment Isolation
	Low pressure in containment
Heat sink to avoid rapid changes	Low pressure at RCS
	RCS subcooling
	Containment subcooling

These top-level objectives for SAM are related, via top down guidance or procedures, to plant conditions and from there to actions. Guidance and procedures are to be used to assure that the objectives are obtained in a severe accident situation.

An accident will pose challenges to the safety functions with the potential to cause violation of acceptable limits.

The symptoms and plant parameters associated with these symptoms, for the different mechanisms that could occur in the plant, have to be identified. Finally, several strategies can be identified for preventing or mitigating the mechanism that cause the safety function challenges.

4.3.1 Development of SAM Strategies and Guidance's

A "strategy" is a set of methods that should be used to carry out a specific function that prevents a potential challenge to occur and/or mitigate its potential consequences in a fixed time and situation, for a specific challenge during an accident with core damage.

Each strategy includes three elements: the potential challenge to be mitigated, the equipment to be used and the action(s) to be taken.

4.3.1.1 SAM Development and Assessment

When developing a SAM-strategy for a specific plant design, one may start by identifying the most obvious challenges to the containment integrity, and the strategies to deal with the phenomena that may develop. The goal is to prevent loss of containment integrity and, more generally, severe environmental consequences. Such a pragmatic, top-goal-oriented approach for SAM development can be used especially if the benefits are obvious and they clearly outweigh any potential concerns associated with the proposed SAM strategies. Since many severe accident phenomena are strongly governed by the sequence of events that had led to core damage, knowledge of the most probable accident sequences - PSA Level 1 work- is required.

Yet there usually remain phenomena and issues that require a detailed, systematic assessment. This is particularly so if the proposed SAM actions could interfere with the priority of preventing core damage. In addition, some severe accident phenomena may require conflicting actions and one should compare the advantages and disadvantages involved. Finally, the spectrum of severe accident sequences and phenomena identified can give rise to new safety issues, and one should be able to assess the importance by comparing their probability and importance to total risk. The level of detail of such PSA-based or PSA-informed SAM development and assessment work may vary considerably.

4.3.1.1.1 Identification of critical PSA sequences at Level 1

Some severe accident sequences are more critical than others. Some examples: containment integrity can be lost in the beginning of an accident (pre-existing openings, isolation failure, steam generator tube rupture, interfacing system LOCA, annual refuelling maintenance), or ATWS sequences in BWR plants might fail the containment before severe core damage. In both cases the main SAM interest could shift from protecting the containment to minimising the radioactive release to the environment. Depending on the critical sequence in question, a secondary containment and related safety systems - if available and efficient - may possess some capability to mitigate the releases.

The overall objective formulated from the PSA-based severe accident management perspective is as follows: If the environmental radioactive releases of a critical accident sequences is expected to be high due to impaired containment retention capability, the likelihood of the critical sequence should be low.

4.3.1.1.2 Identification of critical phenomena at Level 2

After identification of the most important severe accident phenomena, one should assess the effectiveness of the proposed mitigation strategies for protecting containment integrity. In general, the phenomena that could fail the containment early in the course of an accident are the most important. Although the potential for radioactive releases may be also high for some late containment failure modes, there is more time for implementation of mitigative actions in the plant and for protective off-site measures. Typically, severe accident challenges to containment integrity, in an approximate time (not severity) order, are the following: hydrogen combustion, in-vessel steam explosions, high-pressure melt ejection, direct containment heating, ex-vessel steam explosions, direct melt attacks on containment boundaries, slow containment pressurisation due to gas generation, slow degradation of containment leak tightness and basemat erosion.

The overall PSA objectives are the following: (1) If the fission product releases to the environment due to the critical severe accident phenomena were high, the conditional probability of the critical phenomena to occur should be low, and (2) if SAM actions are available to prevent the phenomena or mitigate its consequences, the effectiveness of the SAM systems/actions should be assessed and demonstrated.

4.3.1.1.3 Identification of human intervention

The SAM actions referred to in the preceding section require the assessment of the reliability of human actions needed for performing the actions. The times available play a crucial role in human intervention. In addition, the operator instructions and the help from a support team are particularly important, keeping in mind the complexity of severe accident situations

The overall PSA objective is to estimate the human reliability in taking the appropriate SAM actions.

4.3.1.1.4 Identification of information needs

In order to take the required SAM actions, one needs to know the plant status, most importantly the status of the core and coolant system. When using all the plant instrumentation, the information flow may be enormous in an accident situation and due to erroneous measurements caused by harsh plant conditions, some indications may be contradictory. It may be appropriate to quickly concentrate on the most important SAM actions and the most essential measurements and their reliability.

The overall PSA objective is to estimate the reliability of obtaining the information needed when selecting and initiating the required SAM actions.

4.3.1.1.5 Accident management for generic plant states (PWG2)

In 1990, PWG2 established a Working Group on Accident Management and gave the group the following task:

Identify possible generic plant states leading to core melt for pressurised water reactors (PWR) and possible approaches to appropriate strategies for dealing with them.
The group considered only Accident Management measures which serve to prevent core damage, to stop any initiated core damage and to retain the core inside the reactor vessel, maintain containment integrity for as long as possible and to minimise off-site releases for both design-basis and beyond-design-basis accidents.

Only PWRs with vertical U-tube steam generators were considered.

Eight PSAs from six different countries (Biblis B (FRG), Konvoi (FRG), PUN (Italy), 900 MWe (France), 1100 MWe (Japan), Surry (USA), Sequoyah (USA), Ringhals (Sweden)) were examined to identify initiating events with a contribution to core damage frequency (CDF) higher than or equal to 0.5%. The examination was restricted to full power operating states. Fifteen initiating events were then identified and combined with 12 hypothetical failures of individual safety systems resulting in a matrix of 180 sequences. Altogether 32 sequences were recognised as ultimately leading to core melt.

Table 4.3-2 ACCIDENT SEQUENCES LEADING TO CORE MELT

1.	TOTAL LOSS OF FEEDWATER
1.a	during a transient (due to internal/external hazards fire, earthquake, aircraft crash .)
1.b	with a SMALL BREAK (primary)
1.c	with a break on the secondary
1.d1	with SG tube rupture 1 tube
1.d2	with SG tube rupture 10 tubes
1.e	Annulus flooding during a Transient
2.	LOSS OF HIGH PRESSURE INJECTION ON DEMAND following:
2.a	SMALL LOCA, reactor coolant loops
2.b	SMALL LOCA, vessel
2.c	MEDIUM LOCA
3.	LOSS OF LOW PRESSURE INJECTION ON DEMAND following:
3.a	SMALL LOCA
3.b	MEDIUM LOCA
3.c	LARGE LOCA

4.	LOSS OF CONTROLLED SECONDARY SIDE STEAM RELIEF
4.a	DURING A TRANSIENT
4.b	in case of SMALL LOCA
4.c1	in case of SG-Tube Rupture 1 tube
4.c2	in case of SG-Tube Rupture 10 tubes
5.	LOSS OF RHRS HEAT EXCHANGER SUMP WATER SUCTION CONTAINMENT COOLING following:
5.a	SMALL LOCA
5.b	MEDIUM LOCA
5.c	LARGE LOCA
6.	STATION BLACKOUT with:
6.a	Total Loss of Feed Water
6.b	Stuck open secondary side safety valve(s)
6.c	Small Break on the primary
6.d1	SG-Tube Rupture 1 tube
6.d2	SG-Tube Rupture 10 tubes
7.	FAILURE OF SHUTDOWN RODS (Failure to Scram)
7.a	Total loss of feedwater
8.	FAILURE OF HIGH PRESSURE INJECTION SHUTDOWN during:
8.a	SG-Tube Rupture 1 tube
8.b	SG-Tube Rupture 10 tubes
9.	FAILURE OF SG ISOLATION during:
9.a	SG-Tube Rupture 1 tube
9.b	SG-Tube Rupture 10 tubes
9.c	Steam Line Break outside of Containment followed by SG-Tube Rupture
10	Loss of Containment Isolation with:
10a	Pipe break (small) of interfacing system outside containment
10b	Pipe break (medium) of interfacing systems outside containment

The time histories of these sequences were analysed in detail in terms of thermodynamic and other parameters (for example, availability of power or coolant).

Based on the results each sequence has been divided into characteristic plant state intervals according to safety goal objective challenges. In this way 141 characteristic plant state intervals have been defined. The plant state intervals are described by plant parameters. At least 16 (in case of SG-tube rupture 18) plant parameters are necessary to clearly define a plant state interval with regard to challenges and available resources for AM-measures.

Table 4.3-3 Definition of Plant Parameters

Safety Objective	System Function	Plant-Parameter
Subcriticality	Scram, Boration	Core Power Boron-Concentration
Core Cooling Core inventory Primary Pressure Control Heat Removal	Primary water injection Make Heat sink available	RPV-Inventory (3 ranges) RPV-Outlet-Temp. (3 ranges) Sump-Level * RWST-Level* RPV-Pressure (4 ranges) SG-Pressure (4 ranges SG-Level (3 ranges) AFWT-Level*
Activity Retention Containment Integrity Steam Generator Integrity Primary Side Integrity	Containment Isolation Heat Removal from Containment Steam Generator Isolation Closure of valves	Containment Pressure Activity at Secondary Activity outside containment Nuclear Auxiliary Building* Level Nuclear Auxiliary Building* Temperature
Availability of additional Utilities	Electrical power supply Component cooling	Voltage of all non-battery buffered systems --

* = required for AM-measures

Many plant state intervals were found to be similar for the various sequences. To all plant state intervals (141) suitable accident management measures were allocated. As expected, different accident sequences contain identical plant state intervals. These were grouped into Generic Plant State Intervals.

The number of descriptive plant parameters and generic plant states can be substantially reduced, if only challenges to safety objectives such as "core cooling" are desired. In this case, not more than 4 plant parameters are needed. Four plant parameters lead to only 14 generic plant states. But besides the indication of safety objective challenges the four plant parameters do not provide any information about available resources for AM-measures. This would be very important for selecting the most effective AM-measures.

4.3.1.2 Assessment of vulnerabilities and capabilities

A necessary step in SAM planning is to identify the vulnerabilities of the plant, which cause possible challenges to the safety functions, and to identify the mechanisms by which the barriers to the release of radioactive material can be challenged.

Another important step is the determination of the capabilities of the plant. This involves safety systems as well as non safety systems that can be used to implement any specific strategy.

The assessment of vulnerabilities should be based on analyses of the plant's response to accidents beyond the design basis. This should be done in a realistic way using best-estimate and should include plant situations other than full power operation, taking into account the compatibility with current EOP's and Critical Safety Functions procedures established in the specific plant.

This assessment should be supplemented by the following inputs

- Safety research into severe accident phenomena.
- Study of operational experience and precursors events.
- Generic studies and analysis done for similar or reference plants.
- Review of existing procedures to assess reliability their limitations.
- Evaluation of instrumentation behaviour and limitations.
- Evaluation of utility capability for emergency situations.

Level -2-PSAs provide a systematic plant specific framework to integrate all this information, to assess the relative importance of each issue, to model plant modifications and AM-measures and finally to describe the level of protection in terms of remaining uncontrolled sequences and their frequencies and consequences.

It is not necessary to investigate all possible accident sequences and to determine their probabilities of occurrence, but it is important to assess the limitations of the PSA done for the plant once it is decided to develop the SAM Guidance (SAMG). The development of a severe accident management program would therefore, in general, use the PSA as a basis. But even if a PSA is not available or not yet completed, it should still be possible to start to develop the accident management guidance and strategies.

The investigation of capabilities and vulnerabilities would then have to be based on analysis and experience from other similar plants and from the plant under investigation, supported to the extent possible by plant specific information and analyses and, possibly, by external experts who have analysed similar plants. In this case, plant specific differences should be carefully considered. Plants of similar design can have operating and hardware characteristics which can result in different behaviour and different ways of managing severe accidents. It is recommended that a PSA be carried out as soon as possible, and the findings of that analysis be used to update and complete the severe accident management program.

4.3.1.3 Identification of Guidance and Strategies

The objectives of the guidance should be specified and related to the basic safety functions to be maintained or restored. One of the first activities in developing guidance should establish criteria based on identified parameters associated to physical states. This helps to prioritise the action levels and determine the various parameter values or thresholds needed as input to specific guidance in a logical decision making process by operators and/or technical support staff.

Following the defence in depth principle, the failure or not implementation of guidance to achieve the objectives at one level should still leave options for achieving the objectives at a later time.

A guidance could include one or more strategies like those listed in sections 4.3.3 and 4.3.4.

Typically the following priorities and associated guidance’s, included in Table 4.3-4, based on fission' products release control, should be established once the plant leaves the domain of applicability of EOP's and enters in a severe accident progression and management

Table 4.3-4 Generic Guidance’s and Objectives

GUIDANCE	OBJECTIVES
Inject into the steam generators.	Establish heat sink for RCS
	Avoid fission product release in SGTR
Depressurise the RCS	Injection to the RCS by low pressure sys.
	Avoid RCS failure at high pressure
Inject into RCS	Recover cooling of the core
Inject into containment	Prevent vessel failure
	Assure NPSH for the ECCS pumps
Reduce fission product releases	Reduce FPR if there is a leak in containment
Control containment conditions (pressure, temperature). Depressurise if necessary	Minimises fission product release
	Obtain margin to cont. failure pressure
Reduce containment hydrogen. Control hydrogen flammability	Avoid hydrogen combustion
	Return containment to controlled, stable state
Flood containment	In and ex-vessel cooling of debris
	Reduce fission products release

4.3.1.4 Investigation of Information Needs and Instrumentation

In order to assess the plant conditions and the specific plant parameters related to these conditions for use in SAMG, it is necessary to review the plant instrumentation that gives information about these parameters, taking into account not only the direct measurement, if existent, but also alternate instruments or methods of measurement that together could provide the adequate value or trend of this parameter.
In Table 4.3-5, there is an example for a PWR of this relationship

Table 4.3-5 Plant conditions and associated parameters

PLANT CONDITION	PARAMETER
Core cooling	Core Temper.; RCS Temperature; Vessel
	Inventory
Reactivity	Boron in RCS, Nuclear Instrument
Heat sink at containment	Pressure and Temperature in containment
Inert Containment	Hydrogen and pressure in containment
Low pressure at RCS	Pressure in RCS
Inventory in steam generators	Level in steam generators
Containment Isolation	Containment Isolation status, radiation
	level in and out
Low pressure in containment	Containment pressure
Low pressure at RCS	Pressure in RCS
RCS subcooling	PIT in RCS
Containment subcooling	PIT in containment

The correct interpretation of instrumentation indications is fundamental to achieve correct diagnosis, control, decisions, implementation of the strategies and evaluation of their effectiveness.

It has to be analysed, whether the implemented instrumentation in the LWRs is sufficient to fulfil these requirements.

4.3.1.5 Assessment of Measures

Analyses with best-estimate codes have to be performed to assess the effectiveness of the severe accident management strategies and to answer questions with respect to minimum equipment requirements, timing of actions, influence of uncertainties and human actions in different plant conditions as well as the positive or negative impacts expected. This analysis should be done as a preparation of the specific SAMG as well as for normal training in severe accidents for the operators and the staff in the plant.

The feasibility assessment considers equipment and human performance under severe accident conditions and availability of information. A further important aspect is the evaluation of the accessibility of equipment which has to be operated or repaired. An examination has to be conducted to determine whether the equipment concerned can be operated/ repaired without exposing the plant staff to excessive radiation, temperature, etc.

For some severe accident management strategies, plant personnel must be allowed to deactivate functions of safety-related Instrumentation and Control. It has to be assessed whether the actions can be performed in time and whether the administrative controls to prevent inadvertent execution are effective.

4.3.2 Implementation of Strategies

For the implementation of the specific strategies the following steps are important

4.3.2.1 Development of Procedures and Guidance

On the basis of the technical assessment of strategies, as well as the analysis done in Level 2, if available, and the plant specific capabilities; final procedures and guidance can be developed.

The procedures in the prevention area are plant-specific, like the guidance or procedures for the mitigation regime, but somewhat more generic conditions could be used for the entry conditions in each guidance, for example, generic conditions supplied by the corresponding owners groups, depending on the plant technology.

The severe accident guidance should be symptom based, with few exceptions related to accidents that evolve very quickly, like ATWS or large LOCA.

The interfaces or the new guidance/procedures with the existing EOP have to be carefully analysed, specially for those actions that could be at variance with the EOP and the SAMG (e.g. start/stop containment spray, start/stop hydrogen recombiners, etc.). It is recommended that the transition from EOP to SAMG be clearly stated and based on a small number of parameters, basically the core exit temperature provided by the vessel thermocouples or an equivalent parameter that signals the onset of severe accident.

Integration of the large variety of information requires many different skills and it must be done by a multidisciplinary team with involvement of: operating personnel, technical personnel knowledgeable in severe accident phenomena, emergency plant personnel, training personnel, etc., to increase the acceptance of the guidance/procedures.

4.3.2.2 Plant organisation and Decision Making Process

Equally important for the plant capability to cope with severe accidents are organisational aspects. In this area, the severe accident management program must state clearly "where" the responsibility is, "who" must take the final decision, "who" will evaluate the best option, "who" will implement the strategies, "who" will declare the plant or parameters in a controlled and stable state-, and so on. It is strongly recommended that the definition of responsibilities should be stated in the administrative procedures of the plant before starting the severe accident management program that could be officially implemented at a specific site.

Special care should be taken when analysing, by the responsible staff, the strategy to be implemented in order to evaluate the positive as well as the negative impacts that this strategy could cause in the plant, in order to take the most favourable decision.

Once the strategy has been implemented, it is necessary to validate whether the plant behaviour would be as expected and to monitor in the long term the parameters that are to assure that a stable and controlled state is reached for a given physical situation.

The availability of information at the place where the decisions are to be taken must also be assured.
It is important for the decision making process to have diagrams and flow charts available at the place where the decisions are to be taken. These should include the priorities and, therefore, the guidance that should be implemented once a safety function is challenged.

In general it is considered that for severe accidents beyond the EOP, the decision process must be carried out by the technical support staff, instead of operators, once the technical support centre has been declared operative. Thus, it is important that a good communication between operators at the control room and the technical support centre be established to guarantee the success of severe accident management.

4.3.2.3 Validation

The guidance and procedures have to be validated from the point of view of their usability, technical accuracy, scope and function. Functional validation includes demonstration of the compatibility of the procedures with the plant and control room lay-out and the accessibility to rooms where actions have to be performed during severe accidents, as well as to verify the capability of the decision making process: responsibilities, good communications, etc.

Use of an interactive simulator specific for severe accident analysis may not be required but in some cases could be useful. In view of the limitations of the normal simulators capability to represent severe accident behaviour, desk or table top validations on the basis of code results can be made, completed if necessary with plant walkdowns.

Training workshops with plant personnel including high level management, delivers valuable feedback to the validation process. The validation considers the large uncertainties in understanding severe accident and ensures that there is sufficient flexibility in the procedures to accommodate potential uncertainties.
The feedback from the initial validation process should be introduced in the severe accident management guidance before these have been declared to be "officially" implemented.

4.3.2.4 Training

Because the success of accident management relies heavily on manual action and capability of evaluations of the plant state and behaviour during the severe accident conditions, training is of special importance to overcome the degradation of human performance during stressful situations and to reduce the potential for human errors. Therefore, appropriate training requirements have to be determined to develop an integrated training program. Comprehensive training must be provided to plant personnel to ensure a common understanding of the concept and contents of guidance and procedures as well as of the roles and responsibilities of the involved personnel.

4.3.2.5 Periodic Exercises

Periodic exercises are recommended to be performed in the plant in order to maintain the capability and guidance usability demonstrated in the first validation carried out in the plant.

The periodic exercises should complement the normal emergency exercises that each plant does periodically, but the objectives are slightly different. It is not necessary to verify all the steps that normal and official exercises require, but just the correct organisation for severe accident situations as well as the technical feasibility of the guidance. It is suggested the overall guidance to be tested in a small number of periodical severe accident exercises.

Candidates for SAM measures and their potential

4.3.3 Potential strategies for PWR

"A" Strategies (preventive strategies):

Maintain Coolant Inventory:

- refill RWST with borated water or CST with condensate
- reduce containment flow rate to conserve water for core injection
- use charging pumps for core injection
- use alternate injection for RCP seal cooling
- fast secondary side cool-down to utilise water sources for low pressure systems

Maintain Decay Heat Removal:

- use condenser or start-up pumps for feedwater injection
- enable emergency connection of feedwater to rivers, reservoirs or municipal water systems
- enable emergency cross-tie of service water and CCW to feedwater
- use diesel driven pumps for injection to containment spray or steam generators
- initiation of RHR system outside normal ranges

Reactivity Control:

- ensure an abundant supply of borated water

Maintain Support Systems:

- conserve battery capacity by shedding non-essential loads
- use portable battery charger to recharge batteries
- enable emergency cross-tie of AC power between two units or to onsite gas turbine generator

"B" Strategies (mitigative strategies):

Prevent Vessel Failure:

- use RCP pumps to force flow through the core
- depressurise and inject coolant into the RCS
- remove RCS heat using steam generators (secondary feed and bleed)
- remove RCS heat using PORV (primary feed and bleed)
- flood cavity to cool vessel

Prevent Containment Failure by Slow Overpressurisation:

- use containment sprays to remove containment heat
- use fan coolers to remove containment heat
- flood cavity before or after vessel failure to delay or prevent core/concrete interaction
- use recombiners or ignitors to control combustible gases
- vent containment to relieve pressure

Prevent Containment Failure by Rapid Overpressurisation:

- depressurise RCS to prevent direct containment heating
- flood cavity before or after vessel failure to break up and cool core debris
- vent containment to control combustible gases (pre-vessel failure and/or post-vessel failure)

Prevent Basement Melt-Trough:

- flood cavity to cool core debris before vessel failure and/or after vessel failure

Mitigate Fission Product Release:

- Control Transport Out of RCS
  - use auxiliary pressurised spray to scrub fission product before they are released through the PORV
  - flood cavity before and/or after vessel failure
- Control Transport Outside Containment:
  - flood leak location
  - re-establish containment isolation
  - depressurise containment to reduce driving forces across leak
  - depressurise RCS (steam generator tube rupture)
  - flood steam generator secondary (SGTR)
  - flood break location/interfacing system (LOCA)

4.3.4 Potential Strategies for BWR Plants

"A" Strategies (preventive strategies):

Maintain Coolant Inventory:

- refill condensate storage tank (ST)
- extend ECCS availability by switching pump suction
- use control rod drive (CRP) pumps for core injection
- use fire pump for core injection

Maintain Heat Removal:

- re-open Main Steam Isolation Valves (MSIV's) and turbine bypass valves to regain main condenser
- use fire pump for supplying water to the containment spray
- cross-tie service water to RHR

Maintain Support Systems:

- shed non-essential DC loads
- use portable battery charger to recharge batteries
- cross tie AC power between units or to a gas turbine

"B" Strategies (mitigative strategies):

Prevent Vessel Failure:

- use CRD pumps for vessel injection
- use of fire water for vessel injection
- water pool underneath the core without supply from external source
- external water injection system
- filtered containment venting
- combination of containment flooding with filtered containment venting

4.4 Examples of implemented provisions for mitigative SAM (level 2) and of their effectiveness

4.4.1 SAM implemented at PWRs examined in this report

4.4.1.1 Containment Spray

Heat removal by containment spray is available at all plants but Biblis-B. Its operation in the event of severe accidents may have positive and negative effects:

- Positive: Removal from the containment atmosphere and deposition in the containment of fission products released to the containment during severe accidents.
- Negative: Reduction by steam condensation of the steam inertisation of the containment, thereby increasing the likelihood of hydrogen combustion. With the fuel loaded at the time of the studies, this is a significant concern only at the Combustion Engineering plant Maine Yankee which is vulnerable - due to its high amount of zirconium in the core - to hydrogen combustion. At the other plants it could become a concern if reloaded fuel had thicker fuel rod cladding.

At Borssele, containment spray is only used for fission product depletion in severe accidents.

Additional external injection from fire trucks for backing up the water supply for the containment spray system is provided at Beznau and at all Swedish PWRs. For Sizewell B, the definition of the plant damage states used in the level 2 PSA includes an identifier which relates to operation or failure of the containment spray system and the containment fan coolers. However, sensitivity studies were carried out to model the effectiveness of restoration of either of these two systems to reduce containment temperature, pressure and activity levels. Since the containment spray system is located outside the containment, the chance of restoring it within 24 hours before late containment failure is considered to be high. Since the heat removal rate that is required from the fan coolers to prevent containment failure is relatively low, it has been shown that it would be sufficient to provide cooling water to the heat exchanger coils without operation of the fans. These actions significantly reduced the probability of late containment failure and have both been included as accident management measures in the Station Operating Instructions.

4.4.1.2 Hydrogen Control

Hydrogen control by a combination of igniters and catalytic recombiners is foreseen at Biblis-B (and other German PWRs). Presently, the design, localisation and composition of these devices is being optimised. At the Swedish PWRs, catalytic recombiners that are qualified for severe accident environment are available, although these plants are not particularly vulnerable to hydrogen combustion. At Beznau, early containment venting for removal from the containment atmosphere of hydrogen and oxygen will be implemented. At Borssele early venting is under study, as well as combinations of recombiners with igniters, and with post- accident inertisation. For Sizewell B, hydrogen control is achieved by mixing the hydrogen that is produced in the containment atmosphere using the hydrogen mixing fans. Operation of the containment spray and the fan coolers also provides a mixing effect. In the longer term, the hydrogen recombiners can be used although their capacity is only sufficient for post-LOCA hydrogen generation. If all hydrogen recombining capacity is lost, the Station Operating Instructions allow the use of the hydrogen venting system in the last resort if the activity levels within the containment are sufficiently low (although this is unlikely following a core melt). No credit is taken for this in the PSA.

Recombiners for controlling the hydrogen generated in design basis accidents are available at all plants

4.4.1.3 Additional Water Injection to the Containment

Additional water injection to the containment can be used for prevention of core damage and for mitigation of the consequences of core damage.

- Prevention: Backup water sources for
  - low pressure injection/recirculation (Swedish PWRs, using fire trucks (CWIS))
  - containment spray (Beznau and Swedish PWRs, with injection from fire trucks)
  - cooling of containment fan coolers (Beznau, using river water and mobile pumps)
- Mitigation: Water supply for flooding of the containment when core damage and possible RPV failure is imminent.
  - By having a deep water pool underneath the reactor vessel, the extent of basemat attack by molten core debris can be reduced or basemat attack may even be prevented, thus reducing or eliminating the production of combustible gases, as well as the likelihood of basemat penetration.
  - To avoid late overpressurisation failure of the containment due to steam production, this strategy is likely to require the availability of high capacity filtered containment venting.

Procedures and hardware for implementing containment flooding and filtered containment venting are available at Beznau and at the Swedish PWRs. For Sizewell B, water can be added to the reactor cavity using the containment spray system, the containment fire suppression system or by gravity drain from the RWST. The fire suppression system is separate from the normal safety systems, has its own diesel driven pumps and its own spray lines and nozzles inside the containment. Sensitivity studies have shown that the operation of this system will lead to a significant reduction in the probability of both late containment failure and basemat failure. The gravity draining of the RWST into the reactor coolant system is possible when the RCS is open - for example, during mid-loop operation. This can provide make-up if active heat removal is lost via the RHR and can provide water additional water to cool the corium and thus avoid basemat melt-through. These have all been adopted as accident management measures in the Station Operating Instructions.

On the negative side of the containment flooding strategy is the possibility, although very remote according to present understanding, of containment failure due to massive ex-vessel steam explosions resulting from the interaction of the molten core debris with the water. Conditional probabilities, given RPV melt-through, are estimated to be below 10^{-3} for such events.

4.4.1.4 Depressurisation of the RCS for Prevention of High Pressure Melt Ejection.

Depressurisation of the RCS for prevention of high pressure melt ejection (HPME) is available at the plants for at least some sequences involving loss of steam generator feeding; at some plants for nearly all such sequences. By the application of the depressurisation procedure it is intended to prevent DCH phenomena that could threaten the containment integrity. The strategy is particularly beneficial at plants with high "power containment volume" ratio and low estimated containment failure pressure. For Sizewell-B, the RCS can be depressurised using the pressuriser PORVs, the pressuriser spray or by opening the upper head vent and this has been adopted as an accident management measure in the Station Operating Instructions. No credit is taken for this in the PSA.

4.4.1.5 Filtered Containment Venting

In the event of pressure build-up due to the ex-vessel production of steam and non-condensable gases, and the combustion of flammable gases, the failure pressure of the containment may be exceeded in the late phase of an accident. By filtered containment venting prior to critical pressure build-up, catastrophic failure of the containment can be avoided. For conducting filtered containment venting, provisions have to be made for avoiding hydrogen detonations in the filter and its connecting lines. The likelihood of such events could become significant due to condensation phenomena that reduce steam inerting. At Beznau, design modifications of the venting system have been implemented that are believed to reduce this likelihood to insignificant. Filtered containment venting is implemented at all German and Swedish PWRs, at Beznau and Borssele. It is very beneficial in combination with the strategies for having large quantities of water available for debris quenching (available at Beznau and the Swedish plants). For Sizewell B, provision was made in the design to include a filtered containment venting system and the PSA was used to consider the benefit in term of risk reduction from the system. It was not incorporated since it was concluded that it would not be cost-effective.

4.4.1.6 Use of Primary Side Bleed/Feed in the Event of Steam Generator Tube Rupture

In the event of core damage involving steam generator tube rupture with unisolated steam generator, the release to the environment of the volatile fission products, including noble gases, can be significantly reduced if primary side bleed/feed (PB/F) is applied: through the split-up of the mass flow between the broken steam generator tube (few cm^2) and the open pressuriser valves (40 to 60 cm^2), most of the fission products released from the core are directed to the containment. Calculations performed for DRS-B have shown the potential for significant reduction of the releases. Further analysis are needed for the scenario after vessel failure.

The strategy is available and was credited in DRS-B and is in place at the Swedish PWRs, Beznau, Borssele, Sizewell B and many US PWRs. The strategy is included as an accident management measure in the Sizewell B Station Operating Instructions. However, no credit is taken for this in the PSA.

4.4.1.7 Filling with Water of an Unisolated Steam Generator in the Event of Steam Generator Tube Rupture

The releases from a ruptured tube in an unisolated steam generator can be drastically reduced by scrubbing of gases in a column of water in the defective steam generator. This option is available at Beznau, Borssele, Sizewell-B and at the Swedish PWRs. At Beznau, Borssele and Sizewell-B, fire water can be used to fill up the defective steam generator. At the Swedish plants, the optimal strategy is still under investigation. The strategy is included as an accident management measure in the Sizewell B Station Operating Instructions.

However, no credit is taken for this in the PSA.

The scrubbing effect strongly depends on the height of the water column above the break. For U-tube steam generators, analyses consistently show that the likelihood of leaks is highest in the lower part of the steam generator.

In the Ringhals-2 analysis, the overall reduction of caesium releases for events with unisolated steam generator amounts to a factor about 100. In the Beznau analysis a reduction by the factor 10 - 100 of iodine and caesium releases is assumed for sequences with unisolated SG. In DRS-B, the achievable reduction is estimated to be in the same range**.**

4.4.2 Probabilistic effectiveness of SAM implemented at PWRs

Specific quantifications of the probabilistic effectiveness of SAM measures are difficult to find in the literature. However, by comparing level 2 PSA results for different reactor designs some conclusions as to the probabilistic effectiveness are possible:

- For internal events a dominant contribution to large releases comes from unmitigated steam generator tube rupture events with unisolated steam generator. The most significant reduction of this contribution is obtained by fill-up with water of an unisolated steam generator in the event of steam generator tube rupture as described above in section 4.4.1.7. The effect in terms of the conditional probability of exceeding 10% Cs release can be seen in Tables 4.4.2-1 and 4.4.2-2 below and in Figure 2.5-1 in paragraph 2: At Beznau and Ringhals 2 where this strategy is implemented, the conditional probabilities of exceeding 10% Cs release, given core damage, are about one order of magnitude smaller than at plants without this feature, and the conditional probabilities of exceeding 10% Cs release, given a large release containment failure (LRCF) mode, are about one and a half orders of magnitude smaller. Also the absolute values of the exceedance frequency are significantly lower than for the other plants.
- The importance of primary side bleed/feed is not directly visible in Tables 4.4.2-1 and in Figure 2.5-1 because this SAM measure is available at all plants included in this comparison. Table 4.4.2-1 shows that the conditional probability, given core damage, of early containment failure due to HPME is in the range 0.004 to 0.025. The fraction of high pressure scenarios is in the range 2% to 20% of the total CDF, with the majority in the few percent range, see the discussion in section 3.3.4.1 in paragraph 3. The importance of primary side bleed/feed with respect to reduction of CDF is low to moderate. Among 10 examined level 1 PSAs for PWR plants (reference 3.) values from 2% to 27% were found.

  Sensitivity studies show that the fraction of high pressure scenarios would be from 30% upward if primary side bleed/feed was not available. Correspondingly, the contribution to early containment failure due to HPME would be in the 5% to 30% range. Thus, importance of primary side bleed/feed with respect to reduction of containment failure due to HPME is very high (90% and higher). Without primary side bleed/feed, sequences with core melt under high pressure would be as important for large releases as SGTR events with unisolated steam generator.

- Filtered containment venting, in particular if combined with the possibility for external water injection to the containment, is very effective for reducing releases. For Swedish reactors, reduction by a factor up to 100, relative to releases from late containment failure, is reported. However, this is not visible in Tables 4.4.2-1 and 4.4.2-2 and in Figure 2.5-1 due to the overwhelming contribution of other event classes to releases. For Sizewell B, the PSA considered the reduction in risk from the provision of a filtered containment vent and from the use of the containment fire sprinkler system. For the filtered containment vent, there was no reduction in the individual risk and the societal risk (defined as the frequency of more than 100 fatal cancers) was reduced from $7.0x10^{-6}$ per year to $4.9x10^{-6}$ per year. For the use of the containment fire sprinkler system, the individual risk was reduced from $1.9x10^{-7}$ per year to $1.7x10^{-7}$ per year and the societal risk to $3.6x10^{-6}$ per year. The cost of incorporating a filtered containment venting system is very high compared to using a system that is already in place. In view of this, the incorporation of a filtered containment venting system was rejected on cost-benefit grounds.
- For Sizewell B, sensitivity studies were carried out to determine the effectiveness of water addition to the containment by either recovering the containment spray system or by the use of the fire sprinkler system. This can be illustrated by fault sequences in which the RCS pressure is initially high with the containment sprays and fan coolers having failed. With no water addition, 79% of the fault sequences would lead to late containment failure and 21% would lead to enhanced leakage due to tearing of the liner. If it is assumed that the probability of being able to add water to the containment within 24 hours is 0.9, these percentages would be reduced to 8% and 3% respectively, so that in 89% of the sequences the containment would be intact.

Table 4.4.2-1. PWRs with Large Dry Containments. Conditional probabilities of containment failure modes, given core damage Dominant phenomena and their relative contributions

		Containment-Failure mode					
Plant	**Total CDF**	**Early Containment Failure**	**Late Containment Failure**	**Containment-Bypass**	**Isolation Failure**	**Successful containment venting**	**Containment intact**
Surry	4.0 E-5, 3% at high pressure	0.007, DCH >90%	0.06, BMP	0.12, SGTR ~60%	-		0.81, RPV intact 57%:
Zion	6.5 E-5, 2% at high pressure	0.005, DCH >90%	0.24, BMP	0.02, SGTR ~90%	-		0.73
Maine Yankee	7.4 E-5, 16% at high pressure	0.08, H_2 burn ~70%, DCH ~ 30%	0.47, Overpressure	0.02, SGTR ~70%	-		0.43, RPV intact :30%
Robinson	2.4 E-4, 22% at high pressure	0.016, DCH>90%	0.07, Overpressure	0.02, SGTR ~70%	0.13		0.77
Beznau	4.4 E-6, <10% at high pressure	0.016, DCH>80%	0.19, includes vent failure due to hydrogen burn: ~22%, lower after modification of vent line	0.11, SGTR >90%		0.54	0.15
Biblis-B	2.9 E-6, 9% at high pressure	n.a.	n.a.	< 0.04 , V-Seq. <80% (conservative estimate), SGTR,>20%			
Sizewell-B, conservative	2.2E-5	< 0.01	0.19 overpressure	0.09, SGTR~92%	< 0.01	-	0.71
Ringhals 2	2.0 E-5, 12% at high pressure	< 0,01	0.11, BMP > 95 %	0.08, SGTR >90%	0.01	0.3	0.5
Borssele PSA-3	3.6 E-5	< 0.01	0.07	0.01, V-seq. 50%, SGTR 15%	< 0.01-	0.65	0.26
Borssele PSA-97	1.7 E-6	0.01	0.05	0.05, SGTR: 60%, V-seq.: 40%	< 0.01	0.72	0.21
Japan 1100 Mwe PWR	1.9 E-6, 18% at high pressure	0.01, DCH 50%, H_2 burn 40%	0.08	0.34, SGTR 80%	< 0.01	-	0.56

Table 4.4.2-2. PWRs with Large Dry Containments. Frequencies and conditional probabilities of significant and large Cs releases. Dominant phenomena and their relative contribution

	Frequency/a of		Exceedance frequency/a for		Conditional probability of exceeding	
Plant	Total CDF	ECF +Bypass+ISF	1% release	10% release	1% release, given core damage.	10% release, given ECF + Bypass + ISF
Surry	4.0 E-5,	5.1 E-6	6 E-6	2 E-6, SGTR >90%	0.15	0.39
Zion	6.5 E-5,	1.5 E-6	5,5 E-6	1 E-6, SGTR ~30%, DCH: ~70%	0.08	0.66
Maine Yankee	7.4 E-5,	7.4 E-6	4.4 E-6	1.4 E-6, SGTR: ~20%, H_2 burn: ~80%	0.06	0.19
Robinson	2.4 E-4,	8.6 E-6	2 E-5	2 E-6, SGTR ~50%, DCH: ~50%	0.1	0.23
Beznau	4.4 E-6,	5.3 E-7	1.2 E-7	3 E-8, SGTR ~45%, DCH ~55%	0.03	0.05
Biblis-B. Releases were quantified only for SGTR with low RCS pressure. Frequency of high pressure SGTR sequences: $1\ 10^{-8}$	2.9 E-6,		< E-8, PB/F with scrubbing in SG >E-8 otherwise (only SGTR)	<<E-8, with scrubbing in SG > E-8 otherwise (only SGTR)		
Sizewell-B, conservative	2.2 E-5	2 E-6	8 E-6	5 E-6 late overpressurisation: 80%	0.36	0.25, given LRCF mode 0.99, given late overpressurisation 0.22, given core damage
Ringhals 2	2.0 E-5,	1.8 E-6	2E-7, ECF ca. 50%, isolation failure ca. 50%	5 E-8, ECF > 90%	0.01	0,03
Borssele PSA-3	3.6 E-5	8 E-7	8 E-7	3 E-7, V-seq.: 70%	0.02	0.37
Borssele PSA-97	1.7 E-6	1.1 E-7	1.5 E-7	1 E-7, V-seq.: 70%, ISF: 15%	0.08	0.6
Japan 1100 Mwe PWR	1.9 E-6,	7 E-7	7.4 E-7	6.9 E-7	0.39 (without credit to SAM)	0.36, given core damage (without credit to SAM)
Pickering A	1.3 E-4		< 1 E-7	< 1 E-8	< 8 E-4	?

4.4.3 SAM implemented at BWRs examined in this report

4.4.3.1 Containment Pressure Relief System

Use of the high capacity containment pressure relief system without filtering for alternate heat removal in sequences that include failure of the pressure suppression system, but operability of the normal ECCS systems. As the request for pressure relief will occur when the core is not yet damaged, filtering is not required The system is implemented at most Swedish and US BWRs plants. In addition to using this system for alternate heat removal it is also used for overpressure protection.

4.4.3.2 Water Pool Underneath the Core Without Supply from External Source

To avoid attack by core debris of the drywell liner or of the concrete containment structure that could ultimately lead to the penetration of the containment barrier, the availability of a water pool of sufficient depth under the RPV would be beneficial in the event of RPV failure. Provisions have been made at many Swedish BWRs for utilisation of the freshwater reservoir and/or the firewater system, or the water volume of the condensation pool as water source for flooding of the reactor containment in the event of a severe accident.

At the Mühleberg plant a very large in-pedestal sump volume is available by design (about 5 times as large as in US plants with Mark I containment), which can accommodate twice the debris volume. Additional provisions for flooding of the containment are also available at this plant.

4.4.3.3 External Water Injection System.

External water supply for safety systems and for accident mitigation is available at the Swedish BWRs and at Mühleberg. It can be used to provide additional suction sources for

- high pressure auxiliary feedwater (Ringhals 1)
- addition of water to the containment spray system (all Swedish BWRs), Mühleberg
- and for accident mitigation by
- flooding of the containment to the upper core level, ensuring stable terminal cooling of core material (all Swedish BWRs), Mühleberg

4.4.3.4 Filtered Containment Venting

To avoid a breach of containment integrity due to a slow pressure increase following an accident, systems for filtered containment venting have been installed at all Swedish and German BWRs, and at Mühleberg The filtered containment venting system is used for alternate heat removal (preventive AM) and for overpressure protection of the containment (mitigative AM).

4.4.3.5 Combination of Containment Flooding and Filtered Containment Venting

The combination of

- having or making available large quantities of water underneath the core in the event of a severe accident (to avoid attack of the drywell liner or other containment structures) and
- of having available high capacity filtered containment venting

are reported to be beneficial for avoiding large releases to the environment, see Table 8 and Figure 1. This table and the illustration show that Mühleberg, Forsmark 3 and Barsebäck (which are equipped with this feature) have the lowest conditional probabilities, given core damage, for Cs releases of all examined plants (at Mühleberg, this is also due to the large in-pedestal sump volume which practically eliminates drywell liner attack in the event of debris pour)

4.4.4 Probabilistic effectiveness of SAM implemented at BWRs

Specific quantifications of the probabilistic effectiveness of SAM measures are difficult to find in the literature. However, by comparing level 2 PSA results for different reactor designs some conclusions as to the probabilistic effectiveness are possible:

- At many BWRs with Mark I containment the in-pedestal sump volume is much smaller than the core volume, see Table 2.3-2 in paragraph 2. In the event that a substantial fraction of the molten core exits the RPV bottom head, it can not be accommodated by the sump volume and spill-over to the drywell with subsequent attack of drywell liner and structures will result. This scenario is responsible for the main contribution to large early releases at BWRs with this design feature. At many of these plants, SAM measures have been implemented for flooding the drywell if RPV failure is imminent. Reduction by factors 2 to 5 are reported for the conditional probability, given core damage, of the described failure mode .
- Containment pressure relief systems with or without filters in the vent line have been implemented at most BWRs, primarily as means for alternate heat removal, but there is also the level 2 aspect of protecting the containment against overpressure failure. Such systems are credited with a reduction by the factor 5 to 10 of the CDF contribution of sequences involving loss of the engineered heat removal systems. Explicit quantifications of the level 2 effect are not available.

Swiss and Swedish BWRs are equipped with high capacity filtered containment venting systems and systems for external water injection to the containment. The effect in terms of the conditional probability of exceeding 10% Cs release can be seen in Tables 4.4.4-1 and 4.4.4-2 below and in Figure 2.5-1 in paragraph 2: At Mühleberg, Forsmark 3 and Barsebäck where this strategy is implemented, the conditional probabilities of exceeding 10% Cs release, given core damage, and the conditional probabilities of exceeding 10% Cs release, given a large release containment failure (LRCF) mode, are smaller than at plants without such SAM measures. Also the absolute values of the exceedance frequency are significantly lower than for the other plants.

Table 4.4.4-1 BWRs. Conditional probabilities of containment failure modes, given core damage state Dominant phenomena and their relative contributions

			Containment failure mode					
Plant	Total CDF	Containment type	Early containment failure	Containment bypass	Early containment drywell-failure without suppression pool bypass MK III	Late containment failure	Containment - venting	Containment intact
Peach Bottom	4.3 E-6	MK I	0.56, Liner failure ~60%, DCH ~5%%		-	0.05, Overpressure >90%	0.11	0.27, RPV-intact ~40%
Browns Ferry	4.8 E-5	MK I	0.46, Liner failure ~60%, Overpressure ~8%		-	0.26		0.28, RPV-intact ~90%
Grand Gulf	4.1 E-6	MK III	0.21, H_2-burn ~75%		0.22, Overpressure >90%	0.28	0.04	0.23, RPV-intact ~75%
Perry	1.2 E-5	MK III	0.16, H_2-burn >90%		0.07, Overpressure >90%	0.07	0.31	0.39, RPV-intact 70%
Mühleberg	3.5e-6	MK I	0.26, Overpressure		-	0.07, overpressure	0.66	-
La Salle	4.4 E-5	MK II	0.33, Overpressure, CCI			0.1	0.46	0.11
Barsebäck 1/2, (draft)	3.9 E-6	ASEA	0.1 CCI, 40% Reactor overpressure, impact of vessel head failure, 40%	0.05, Isolation failure of main steam line with reactor at high pressure >90%		<0.01	0.84	
Forsmark 3,	7.2 E-6	ASEA, IV	2 E-3	5 E-4		8 E-5	0.5	0.48
Dodewaard	5.5 E-5	Humbold Bay (pre-MKI)	0.25, Overpressure: 95%, Ex-vessel steam explosion: 2.5%			0.36, CCI or liner attack 90%, thermal drywell failure 3%		0.38
Japan 1100 Mwe BWR	7.6 E-7	MK II	0.51, Overpressure failure before core melt 70%	0.03, V-seq.	-	0.29, Overpressure > 40%	-	0.16

Table 4.4.4-2. BWRs. Frequencies and conditional probabilities of significant and large Cs releases, dominant phenomena and their relative contribution

Plant PSA	Frequency/a of		Exceedance frequency/a for		Conditional probability of exceeding	
	Totality of containment failure modes	ECF + bypass + ISF	1% release	10 % release	1% release, given core damage	10% release, given ECF + Bypass + ISF
Peach Bottom, NUREG-1150	4.3 E-6	2.4 E-6	2 E-6	1.3 E-6, liner failure	0.46	0.54
Browns Ferry, IPE	4.8E-5	2.2 E-5	1.2 E-5	5 E-6, liner failure	0.25	0.22
Mühleberg, HSK/ERI	3.5 E-6	9 E-7	3.3 E-7	1.2 E-7, early overpressure failure	0.1	0.13
LaSalle	4.4 E-5	1.5 E-5	1.4 E-5	3.6 E-6, overpressure, CCI	0.32	0.24
Grand Gulf, NUREG-1150	4.1 E-6	8.6 E-7	1.5 E-6	5 E-7, hydrogen burn	0.36	0.58
Perry, IPE	1.2 E-5	1.9 E-6	4 E-6	5 E-7, hydrogen burn	0.33	0.26
Barsebäck 1/2 (Draft)	3.9 E-6	3.9 E-7	5.4 E-7	1.4 E-7, steam line isolation failure, CCI, impact of vessel head failure	0.13	0.36
Forsmark 3	7.2 E-6	2.4 E-8	2.7 E-8	5 E-9 containment bypass	0.0038	0.2
Dodewaard	5.5 E-5	1.4 E-5	3.2 E-5	7.8 E-6, early overpressure failure of wetwell	0.58	0.55
Japan 1100 Mwe BWR	7.6 E-7	3 E-8	6.5 E-7	4.2 E-7	0.86 (without credit to SAM)	0.56, given core damage (without credit to SAM)

4.5 Identification of Recovery and SAM Actions in the Level 1 Domain that can influence SAM in the Level 2 Domain. Some Examples

Success or failure of a number of recovery and accident management actions in the level 1 domain is relevant for the accident progression analysis in the level 2 domain. It is therefore important that the information on such actions is available for the level 2 analysis by proper inclusion in the plant damage states. Below is a list of actions that were found to be important in PWR level 2 analyses.

4.5.1 Pressurised Water Reactors

- Maintaining the availability of the service water system for containment spray recirculation during loss of offsite power events at Surry. Surry has a gravity fed service water system which relies on the head difference between the intake and discharge canals. The intake canal is resupplied with water by the recirculating service water pumps. These are unavailable during loss of offsite power.

  In the event that a condenser fails to isolate, the outflow through the condenser is greater than the makeup capability of the diesel driven emergency service water pump, potentially leading to canal drainage before the restoration of offsite power. To maintain service water availability, the condenser(s) can be isolated by manually closing the isolation valves(s).

- Realignment of the auxiliary feedwater system to the townswater reservoir.
- Restoration of the bus powering the containment residual heat removal train pumps., following loss of offsite power events at Surry. This bus must be manually reconnected after load shed during loss of offsite power events
- Initiation of bleed/feed (all plants)

  Failures to initiate bleed/feed are relevant in the level 2 domain because of their impact on the possibility to mitigate high pressure core melt sequences. By the application of the depressurisation procedure it is intended to prevent DCH phenomena that could threaten the containment integrity.

- Initiation of secondary side bleed (to reach conditions for secondary side feed) at Maine Yankee
- Diagnosis of steam generator tube rupture and identification of the ruptured steam generator (Beznau, Surry, Sequoyah)
- Initiation of primary side bleed/feed in the event of steam generator tube rupture. In the event of core damage involving steam generator tube rupture with unisolated steam generator, the release to the environment of the volatile fission products, including noble gases, can be significantly reduced if primary side bleed/feed is applied: through the split-up of the mass flow between the broken steam generator tube (few cm^2) and the open pressuriser valves (from 20 cm^2 upward), most of the fission products released from the core are directed to the containment. (Surry, Zion, Ringhals PWRs, Borssele, Beznau)
- Filling of ruptured steam generator with water the releases from a ruptured tube in an unisolated steam generator can be significantly reduced by scrubbing of gases in a column of water in the defective steam generator. The scrubbing effect strongly depends on the height

of the water column above the break. (already installed in Beznau, Borssele, Ringhals PWRs, Sizewell B, N4)

- Controlled containment spray actuation, as a function of containment pressure, to extend the time to initiation of recirculation
- Use of hydrostatic test pump to inject to the seals of reactor coolant pumps as an alternative to charging pumps to avoid RCP seals failure and induced small LOCA.
- Recovery from CCF of the containment spray heat exchanger valves by locally opening or repairing the valves at Surry
- Diesel generator recovery (Surry, Sequoyah, other US plants)
- Establish alternate ESF pump room cooling by using portable coolers and fans (Sequoyah)
- Locally open the SWS motor operated valves to containment spray system heat exchangers (Sequoyah)
- External water injection to the containment.

  Additional water injection to the containment can be used for prevention of core damage and for mitigation of the consequences of core damage.

- Prevention: Backup water sources for
  - low pressure injection/recirculation (Beznau and Swedish PWRs, using fire trucks (CWIS))
  - containment spray (Beznau and Swedish PWRs, with injection from fire trucks)
  - cooling of containment fan coolers (Beznau, using river water and mobile pumps)
- Mitigation: Water supply for flooding of the containment when core damage and possible RPV failure is imminent. By having a deep water pool underneath the reactor vessel, the extent of basemat attack by molten core debris can be reduced or basemat attack may even be prevented, thus reducing or eliminating the production of combustible gases, as well as the likelihood of basemat penetration.

4.5.2 Boiling Water Reactors

- Diesel generator recovery (Peach Bottom, Grand Gulf, Browns Ferry, Perry, Muhleberg)
- Offsite power recovery (Peach Bottom, Grand Gulf, Browns Ferry, Perry, Muhleberg)
- Manual start of standby liquid control system in ATWS situation (all examined BWRs)
- Manual alignment and actuation of backup injection, for example, fire water system, control rod drive system, high pressure service water system, river water, condensate system with feedwater pump bypass, after failure of the engineered injection systems, (all examined BWRs)
- Manual depressurisation of RPV to reach conditions for low pressure injection after failure of high pressure injection and automatic depressurisation. (all examined BWRs).

- Containment venting for RHR. In some plants the same venting system is used for residual heat removal and for overpressure protection of the containment (Peach Bottom, Grand Gulf, Browns Ferry, Perry, Swedish BWRs)
- Manual actuation of late RPV injection for pedestal cavity flooding (Perry)
- Manual alignment and actuation of the external water injection system for flooding of the pedestal area.
- Manual actuation of flooding of the pedestal cavity from the wetwell (Swedish BWRs)
- Use low pressure injection/recirculation to the vessel (and therefore to the cavity) after vessel failure.

4.6 References

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5. AVAILABLE METHODOLOGY FOR QUALITATIVE LEVEL 2 ANALYSIS

5.1 Level 1/2 Interface

5.1.1 Introduction

Level 2 PSAs examine the responses of the containment and of its engineered safety systems to the loads attending core damage accidents. Results of level 2 PSAs are expressed in

- containments failure states and their frequencies of occurrence,
- the releases to the environment of radioactive substances depending in the related containment failure states.

The input to the analysis is produced in a preceding level 1 PSA. It consists of the set of failure states, including frequencies of occurrence, of the reactor system and of active containment systems that are of importance for to the containment analysis. The elements of this set are called plant damage states (PDS).

They provide the interface between level 1 and level 2 analyses by defining the initial and boundary conditions for the level 2 analysis. Criterion for the definition of the plant damage states is the similarity of the failure states with regard to the

- further progression of the accident in the reactor system
- functionality of the active containment systems
- response of the containment structure to the attendant loads

In the level 2 analysis it is necessary to take into account the influence of accident management actions that are considered in the level 1 analyses. Such actions can significantly influence the characteristics of plant damage states, the further progression of the accident as well as the feasibility, effectiveness and success probabilities of severe accident management measures in the level 2 domain. Therefore, the inclusion of failed preventive (level 1) accident management in the development of the plant damage states is an important aspect of the definition of the level 1/level 2 interface. An important aspect of the back end of the level 2 analyses, is whether the characteristics used for the grouping of the end states of the accident progression analysis permit to trace the influence of accident management actions all the way to the release categories In this report, the described issues will be examined on basis of the PSAs and related literature listed in the sub-section.

5.1.2 *Plant Damage State Definition and Analysis*

A level 1 analysis identifies the dominant event sequences that lead to core damage. The final stage of the event tree analysis process maps the dominant core damage sequences into plant damage states (PDSs).

The plant damage state analysis involves the identification of detailed PDS categories using multi-state indicators. The resultant number of plant damage states is usually large and difficult to manage in the containment event tree quantification process. Therefore, the plant damage states are grouped into more manageable plant damage state groups.

In the NUREG-1150 analyses, the ERI/HSK analysis for Beznau, and in the Perry IPE, the frequencies of the multi-state indicators are passed on to the containment analysis in which they are used to calculate the split fractions in the containment event tree quantification process. In other studies, condensed information that is representative of the plant damage state groups, is passed on to the containment event tree quantification process

In the event trees developed for the systems analysis stage, only those events and systems failures are examined that are needed to determine whether or not the accident sequences would lead to core damage. This includes failures of containment systems that can put the plant in a core vulnerable state in which core damage can be caused indirectly as a consequence of the containment failures. For these examinations, it is not necessary to know which containment system (or which combination of containment systems) failed - only that some form of failure occurred. Thus, the status of the containment systems is simplified by using only a single top event to track all forms of containment heat removal failure. Additional event tree headings are added to further distinguish containment spray and containment heat removal systems failures.

In the plant damage state analysis in NUREG-1150 and in the Perry IPE, event trees are developed that generate the information needed to assess the degree to which the containment systems remain operable as a means of preserving containment integrity and preventing or reducing the amount of radionuclide release following core damage. The plant damage state event trees include such headings as

- containment spray injection, containment heat removal by the containment spray recirculation system and/or the high/low pressure recirculation system, (for PWRs)
- containment heat removal with RHR spray loop, containment heat removal with RHR suppression pool cooling, containment heat removal with venting, late RPV depressurisation, late RPV injection for pedestal cavity flooding. (for BWRs)

In the ERI/HSK analysis for Beznau, the information needed in the containment analysis is provided by a fault tree linking code that links the fault tree data for a safety system with relevance for containment response to the general plant damage state cutset files. This includes information related to containment spray injection, containment heat removal by the containment spray recirculation system, the high/low pressure recirculation system, external water supply to the containment sump.

5.1.3 *Definitions of the Plant Damage State Indicators*

Below, examples are presented for plant damage indicators. A detailed description covering nine different PSAs is provided in reference 1 (section 5.1.3.3).

5.1.3.1 NUREG-1150 Analysis for Surry and Sequoyah Plants

Seven indicators are used to characterise a plant damage state. They address the following issues:

- Status of RCS at onset of core damage
- Status of ECCS
- Status of containment heat removal capability
- Status of AC power
- RWST injection capability
- Steam generator heat removal capability
- Status of RCP seal cooling

Containment isolation failures are considered negligible for the PWR plants examined in the NUREG-1150 studies; therefore, they are not included among the plant damage state indicators (in contrast to other PSAs)

Each of the seven indicators is discussed below.

- Status of RCS at Onset of Core Damage

  This indicator provides information on the pressure of the reactor coolant system, and its integrity at the time of vessel failure. as the expected RCS pressure is related to RCS integrity. Eight categories of the RCS integrity status are identified and related to the initiating events, as shown below

  T no break (transient)
  A large LOCA (6 to 29)
  S1 medium LOCA (2 to 6)
  S2 small LOCA (1/2 to 2)
  S3 very small LOCA (< 1/2)
  G steam generator tube rupture with steam generator integrity
  H steam generator tube rupture without steam generator integrity
  V interfacing LOCA

  The first character in the PDS designator is commonly referred to as the initiating event, however, the way it is used in the containment event tree (CET) analysis is to indicate the integrity of the RCS at the onset of core damage. Hence, the first character in the PDS designator may differ from the sequence initiating event. For example, if the initiating event is a transient with a RCP seal failure occurring before the onset of core damage, then the CET would treat this case as a small break in classifying the status of the RCS.

- Status of ECCS

Indication of the past and present status of high and low pressure injection or recirculation cooling. Five categories are identified relative to the ECCS, as shown below

A operated in injection only
B operated in injection, not operating in recirculation
R not operating, but recoverable
N not operating and not recoverable
L HPI failed, but LPI operable if pressure is reduced

– Status of Containment Heat Removal Capability

Indication of whether or not containment heat removal is available. For plant damage state definition, this is defined to be the availability of at least one containment spray train (at Surry) or at least one containment spray or LHR train (at Sequoyah) in the recirculation mode, with service water being supplied to the heat exchanger. The alternate means of containment heat removal (via AFW) included in the systems analysis stage event tree would not be available after vessel failure. Four categories are used for this indicator, as shown below. For this indicator it is not always possible to identify a unique state from the sequence outcome. Split fractions were developed to partition containment failure states into plant damage states.

Y operating or operable if/when needed
R not operating, but recoverable
N never operated, not recoverable
S sprays operable, but no CHR (no SW to HXs)

– Status of AC Power

Indication of whether or not the AC power needed for safety systems is available. Two status categories are identified for this indicator.

Y available
R not available, but recoverable
N not available, not recoverable

– RWST Injection Capability

Indication of whether or not the reactor cavity is full of water. In order to assure that the cavity is full of water, the RWST must be fully injected into the containment. No partial credit is taken for RWST injection. Three categories are identified:

Y fully injected into containment
R not fully injected, but could be injected with power recovery
N not fully injected, cannot be injected in future

– Steam Generator Heat Removal Capability

Indication of the status of the AFW system and its ability to provide steam generator heat removal. Six status categories were used for this indicator.

X at least one AFWS operating, SGs not depressurised
Y at least one AFWS operating, SGs depressurised
C steam driven pump operated until battery depletion, electric driven pump recoverable with power recovery - SGs not depressurised
D steam driven pump operated until battery depletion, electric driven pump recoverable with power recovery - SGs depressurised
S steam driven pump failed at beginning, electric driven pump recoverable with power recovery
N no AFWS operating, no AFWS recoverable

- Status of RCP Seal Cooling

Indication of the availability of cooling to the RCS pump seals, which provides a direct measure of the ability to preserve the reactor coolant pressure boundary at the reactor coolant pump seals. Three status categories were used for this indicator.

Y operating
R not operating, but recoverable
N not operating and not recoverable

With the number of attributes possible for each of the seven PDS indicators, there are potentially 25,920 different plant damage states.

All core damage sequences greater than l E-7/yr. are assigned to the appropriate plant damage state, and. all PDSs with frequencies greater than l E-7/yr. are retained for containment event tree analysis. If any PDS between l E-9 and l E-7 represents a substantially more severe containment state than any of the PDSs above l E-7/yr., it is also retained for further analysis.

5.1.3.2 NUREG-1150 Analysis for Peach Bottom

Sixteen indicators are used to identify a plant damage state. They address the following issues:

- Initiating event
- Status of external electrical power supply
- Availability of AC-power
- Availability of DC power
- Status of the safety valves in the reactor coolant system
- Status of high pressure injection
- Status of mechanical rod drive system
- Pressure in the reactor vessel
- Status of low pressure injection
- Status of heat removal from the reactor vessel

- Status of condensate system
- Status of high pressure service water system
- Status of containment spray system
- Status of the containment venting system
- Containment leakage
- Location of eventual containment leakage

Each of the sixteen indicators is discussed below

- Initiating Event

A	Large LOCA
S1	Medium LOCA
S2/3	Small/very small LOCA
T	Transient
TC	Transient without scram (ATWS)
IORV	Inadvertent open relief valve

- Status of External Electrical Power Supply
  - Seismic induced LOSP (not relevant for internal events analyses)
  - Internal event or random LOSP
  - No LOSP
- Availability of AC Power
  - Internal event or random LOSP and loss of all diesel generators
  - At least one diesel generator available
- Availability of DC Power, given Station Blackout
  - All DC power is failed
  - At least one DC train is available
- Status of the Safety Valves in the Reactor Coolant System
  - At least one SRV sticks open
  - No stuck open SRV
- Status of High Pressure Injection
  - Both HPCI and RCIC are initially failed
  - Either HPCI or RCIC is initially working
- Status of the Mechanical Control Rod Drive System
  - CRD is failed
  - CRD actuation failure

  - CRD operable
- Pressure in the Reactor Vessel
  - High - ADS has failed
  - High - operator does not depressurise after failure of ADS
  - Low - Depressurisation by ADS or manual, or by LOCA, or transient with stuck open SRV
- Status of Low Pressure Injection
  - Both LPCI and LPCS have failed and can not be recovered
  - Both LPCI and LPCS are currently not available but can be recovered
  - One pump is running, but no injection due to high pressure in the reactor vessel
  - Either LPCI or LPCS is working
- Status of Heat Removal from the Core
  - All RHR modes are failed
  - All RHR modes are currently unavailable, but can be recovered
  - One RHR mode is available
  - Status of Condensate System
  - Condensate system is failed
  - Condensate system is recoverable
  - Condensate system is available but not injecting
  - Condensate system is working (not possible given core damage states)
- Status of High Pressure Service Water System
  - HPSW is failed
  - HPSW is recoverable
  - HPSW is available, manual line-up and actuation required
  - HPSW is working (not possible given core damage states)
- Status of the Containment Spray System
  - CSS is failed
  - CSS is recoverable
  - CSS is available, manual line-up and actuation required
  - CSS is working
- Status of the Containment Venting System
  - Containment is not vented
  - Drywell is vented

  - Drywell is vented in ATWS, but pressure is still high
  - Wetwell is vented in ATWS, but pressure is still high
  - Wetwell is vented
- Containment Leakage
  - No leakage exceeding technical specifications
  - Leak occurs after accident
  - Rupture occurs after accident
  - Leak or isolation failure occurs before accident
  - Rupture or large isolation failure occurs before accident
- Location of Containment Leakage
  - Containment intact
  - Drywell leakage
  - Drywell head leakage
  - Wetwell leakage

5.1.3.3 Reference

1. Documentation of the Treatment of the Level 1/2 Interface in PSAs, with Emphasis on Accident Management Actions (CSNI/NEA/R(97)xx -to be determined later-).

5.1.4 Impact of the Plant Damage State Indicators and their Attributes on the Treatment of Severe Accident Management in the Containment Event Trees

Severe accident management actions can significantly influence accident progression in the reactor system and containment following a core damage accident. In many cases, such actions depend on the success or failure of recovery and preventive accident management actions which are object of the examinations in the level 1 analyses. For an adequate treatment of the severe accident management actions it is important that all relevant information is available to the level 2 analysis. The following sections provides an overview of how this accomplished in various PSAs.

5.1.4.1 NUREG-1150 Analysis for Surry and Sequoyah

All information on the level 1 event tree split fractions needed for the quantification of severe accident management actions in the containment event trees is passed on to the level 2 analysis. In the analysis for Surry, for example, such information is input to the quantification process for 31 of the 71 questions (branch points) of the containment event tree. For 11 questions, the input is explicitly provided by the attributes of the plant damage states, and for 20 others, the input contains information inherent in the plant damage state attributes.

5.1.4.2 ERI/HSK Analysis for Beznau

In the Beznau analysis, fault tree linking provides the coupling of the information on the level 1 event tree split fractions to the containment event tree nodal questions. For 11 of the 31 containment event tree questions, the input is explicitly provided by the attributes of the plant damage states, and for three others it contains information inherent in the plant damage state attributes.

5.1.4.3 Maine Yankee IPE Analysis

Information on the level 1 event tree split fractions can be partly lost in the condensation process that generates the plant damage state groups. Input to the containment event tree analysis is one sequence for each key plant damage state. The selected sequence is considered to be representative of the dominant sequences included in the corresponding key damage state. Thus, only information related to the selected sequence can be conveyed to the containment event tree analysis, whereas information related to the other sequences is deleted. This process can produce reasonable results, if a plant damage state group contains only similar sequences, or if a plant damage state group contains one dominant sequences, while the other sequences are insignificant. This is the case for the highest ranking plant damage state group, in which the dominant sequence accounts for more than 90% of the sequences in the group. However, other plant damage state groups contain dissimilar sequences with dissimilar split fractions, but with similar frequencies. Thus, information that could be important to the quantification of branch point probabilities, including consideration of severe accident management, may be lost when one ”representative” sequence is selected.

5.1.4.4 NUREG-1150 Analysis for Peach Bottom and Grand Gulf, and IPE Analysis for Perry

All information on the level 1 event tree split fractions needed for the quantification of severe accident management actions in the containment event trees is passed on to the level 2 analysis. In the analysis for Peach Bottom, such information is input to the quantification process for 61 of the 145 questions (branch points) of the containment event tree. For 13 questions, the input is explicitly provided by the attributes of the plant damage states, and for 48 others, the input contains information inherent in the plant damage state attributes.

In the analysis for Grand Gulf, information on the level 1 event tree split fractions is input to the quantification process for 36 of the 125 questions (branch points) of the containment event tree. For 15 questions, the input is explicitly provided by the attributes of the plant damage states, and for 21 others, the input contains information inherent in the plant damage state attributes.

In the IPE analysis for Perry, information on the level 1 event tree split fractions is input to the quantification process for 21 of the 68 questions (branch points) of the containment event tree. For 11 questions, the input is explicitly provided by the attributes of the plant damage states, and for 10 others, the input contains information inherent in the plant damage state attributes.

5.1.4.5 Sizewell B POSR

The definition of the plant damage states includes descriptors related to the operation of the containment systems - that is, the containment isolation, the fan coolers and the spray system in injection and recirculation mode. In addition, for the containment bypass sequences, there is a descriptor related to the operation of the ECCS. In the base-case analysis, it is assumed that if one of these safety systems had failed at the start of the fault sequence it would not be recovered. However, sensitivity studies have been carried out to determine the effect of recovery of the containment spray system or the fan coolers which indicate that this would significantly reduce the likelihood of a late containment failure.

5.1.4.6 ERI/HSK Analysis for Mühleberg and IPE Analysis for Browns Ferry

Information on the level 1 event tree split fractions can be partly lost in the condensation process that generates the plant damage state groups. Input to the containment event tree analysis is one sequence for each key plant damage state. The selected sequence is considered to be representative of the dominant sequences included in the corresponding key damage state. Thus, only information related to the selected sequence can be conveyed to the containment event tree analysis. whereas information related to the other sequences is deleted. This process can produce reasonable results, if a plant damage state group contains only similar sequences, or if a plant damage state group contains one dominant sequences, while the other sequences are insignificant. This is the case for the high ranking plant damage state groups in both analyses. Thus, it appears that no information important to the quantification of branch point probabilities, including consideration of severe accident management, is lost with the schemes for selecting one ”representative” sequence.

5.1.5 PDS considered in PSAs for advanced reactor concepts

The following plant damage states or accident categories nodal questions and release categories have been considered in the PSA of new reactor concepts.

Westinghouse AP 600

CET ACCIDENT SEQUENCES SUBCLASSES

- Core damage with the RCS at high pressure following transient or very small LOCA
- Core damage with no RCS depressurisation but with PRHR operating following small LOCA
- Core damage following loss of offsite power not recovered within 24 hours
- Core damage following loss of all DC supply
- Core damage with partial RCS depressurisation following transient
- Loss of containment integrity and core damage due to loss of containment water inventory
- Core damage with the RCS at high pressure following ATWS or MSL break inside containment
- Core damage following events with full RCS depressurisation and CMT and accumulator failure
- Core melt arrested by normal RHR injection following Medium
- LOCA without CMT and accumulator
- Core damage following LOCA or other events with full RCS depressurisation
- Core damage at long term following failure of water recirculation to RPV after successful gravity injection
- Core damage following LOCA (except large) with partial RCS depressurisation

- Core damage following vessel rupture
- Core damage following LOCA (except large) with partial RCS depressurisation
- Core damage following ATWS events
- Core damage with containment already bypassed, except steam generator tube rupture sequences
- Core damage following SGTR. The containment is bypassed and core damage occurs early
- Core damage following SGTR. The containment is bypassed and core damage occurs at long term following failure of water recirculation gravity injection

General Electric SBWR

CET ACCIDENT SEQUENCES SUBCLASSES

- Sequences of core damage following transients in which either the reactor is not depressurised or it is partially by the opening of only SRVs (DPV fail to open). Containment is not significantly pressurised (most of energy is dissipated in the SP) and the cavity is essentially dry at the time of vessel failure
- Loss of coolant inventory makeup in which the reactor is high at the time vessel failure.
- Loss of offsite power with loss of coolant inventory makeup at short term. Reactor pressure is low at the time of vessel failure.
- Loss of offsite power with loss of coolant inventory makeup at short term. Reactor pressure is high at the time of vessel failure.
- Loss of offsite power with loss of coolant inventory makeup at long term. Reactor pressure is low at the time of vessel failure.
- Loss of all DC supply. Reactor pressure is high at the time of vessel failure.
- Loss of coolant inventory makeup at short term with low reactor pressure at the time of vessel failure.
- Loss of coolant inventory makeup at long term with high reactor pressure at the time of vessel failure.
- Sequences of containment failure potentially leading to the consequential failure of the core cooling function.
- Loss of containment heat removal function following transients or LOCAs.
- Loss of decay heat removal function following transients. Core is cooled at high pressure.
- Loss of containment heat removal function following ATWS events
- Loss of containment integrity due to the failure of the vapour suppression function following full RPV depressurisation either by large LOCA or DPVs opening.
- Loss of the containment integrity due to the failure of the vapour suppression function following partial RPV depressurisation either by medium or small LOCA.

- Loss of containment integrity due to the overpressurisation following recriticality during ATWS events with successful SLCS injection.
- Sequences of core damage following LOCAs or transients with reactor depressurised by DPVs actuation. Containment is pressurised and the cavity could be flooded at the time of vessel failure.
- Core damage with the RCS at high pressure following small LOCA with failure of reactor depressurisation.
- Core damage with the RCS at low pressure following large LOCA or DPVs opening and due to loss of coolant inventory makeup at short term.
- Core damage with the RCS at low pressure following large LOCA or DPVs opening and due to loss of coolant inventory makeup at short term.
- Core damage with the RCS at low pressure for the DPVs opening following Loss of Offsite power and due to loss of coolant inventory makeup at short term.
- Core damage following vessel rupture of failure of the overpressure protection function following ATWS events
- Core damage as consequence of containment overpressurisation failure due to the failure to insert negative reactivity
- Sequences involving core damage with containment already bypassed
- Core damage due to failure of the coolant inventory makeup at short term
- Core damage due to failure of the coolant inventory makeup at long term

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- Intact containment with nominal leakage, wet PCS heat removal
- Intact containment with nominal leakage, dry PCS heat removal
- Intact containment with excessive leakage
- Containment bypass
- Containment isolation failure with no CCI
- Containment isolation failure with CCI
- Early containment failure induced during dynamic phase of core relocation with no
- CCI
- Early containment failure during dynamic phase of core relocation with CCI
- Intermediate containment failure before 24 hours after the onset of core damage
- Late containment failure before 72 hours after the onset of core damage
- Potential very late containment failure (basemat) after 72 hours with CCI

5.2 Accident Progression Event Trees

5.2.1 Introduction

The accident progression event trees analysis has progressed from the simple containment event trees methodology used in the WASH-1400 study to the more complex Accident Progression Event Trees (APET) with a completed uncertainty analysis used in the NUREG-1150. Others methodologies have been discussed with different scope and process, but this to be used has to be related to the level 2 PRA analysis to be performed.

From the accident management point of view the PRA techniques can assess potential accident management strategies with the objective of preventing or delaying the fission products release time enough to avoid mitigation trough operator actions, natural process or safeguard systems.
Given an adequate detail model the PRA analysis (including Interface level 1/level 2 and containment analysis) provide a framework for performing alternative strategy analysis trough sensitivity studies or plant behaviour accuracy analysis.

The level of detail of the model for the analyst work has to be enough for assuring that all actions with significant probability within a sequence are adequately modelled. For the items previously identified a more detailed analysis could be necessary.

In the NUREG/CR-5263 risk management applications are discussed and is suggested that the uncertainty treatment is the PRA aspect with more influence in the risk management.

PRA studies (IPE) insights have to be captured and translate to improve the accident management treatment.

The IPE analysis to be helpful in the accident management need to be realistic, because conservative analysis, hypothesis or models could lead to the identification of the accident management strategies not important or prejudicial during a real accident.

That implies that the success criteria, available times have to be realistic. In some cases these analysis should be performed with separate phenomena codes.

5.2.2 Containment Event Trees definition and Analysis

The containment event trees provide a structured approach for systematic evaluation of containment capability in coping with severe accident. These are used to characterise the progression of severe accident and containment failures modes that lead to fission product releases beyond the containment boundary.

The APET/CET structure and nodal questions muss 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 APET analysis. All recovery actions prior to initiation or core damage must only be credited in level 1 PRA, while any recovery actions beyond the initiation of core damage could be credited as part of the APET with following considerations:

- The recovery actions are included as part of the Emergency Operations Procedures (EOPs) for the plant under consideration. The APET quantification is based on a realistic human reliability analysis, thus, providing adequate bases for selection of the branch probability estimates.
- The impact of severe accident environment on the survivability of active components must also be considered. For instance, recovery of power does not necessarily ensure recovery of pumps, though the initiating event may have been caused by loss-of-power, because the pumps in question could have been rendered inoperable as a result of flooding, excessive aerosol loading, and severe radiation environment, beyond the original design basis equipment qualification limits.
- Potential adverse effects of recovery must also be considered as part of the event tree quantification. For instance, water injection to a degraded core has the potential to arrest the further progression of the severe accident, however, there is also the potential for an energetic fuel coolant interaction, additional steam, hydrogen, and fission product releases.

The assessment of accident management actions is influenced by the inherent uncertainties resulting from incompleteness and modelling inadequacies:

5.2.3 Containment Event Tree indicators for PWR plants.

Severe accident progression is modelled by the Containment Event Tree (CET) methodology,), also called Accident Progression Event Tree (APET). Some examples are presented below:

5.2.3.1 Accident Progression Event Tree for Beznau.

The APET consists of 33 nodal questions.

Three time phases are represented by the APET structure including:

- Accident progression from initiation of core damage to the time of debris relocation into the lower head.
- Phenomena occurring at debris relocation into lower plenum until soon after reactor pressure vessel breach, and
- Phenomena occurring several hours after vessel breach during extensive core-concrete interactions.

All operator actions (including recovery of containment isolation failure) prior to core damage are excluded from APET consideration. These recovery actions were already credited as part of the level 1 quantification of core damage states. Operator actions that involve post-core damage initiation are credited as part of the APET structure, provided Beznau-specific Emergency Operating Procedures (EOPs) are available. Automatic system recoveries are included in the APET structure, although at the present time they are not credited due to a lack of plant specific data.

Beznau APET Nodal Questions

For very early time frame:

1. Is the containment isolated?
2. Fraction of PDS with AC power available?
3. What is the mechanical status of the sprays in very early time frame?
4. What is mechanical status of the fans in very early time frame?
5. Does RCS depressurise manually in very early time frame?
6. Does temperature-induced hot leg failure occur in very early time frame?
7. Does temperature-induced SGTR occur in very early time frame?
8. Is AC power restored or maintained in very early time frame?
9. Are sprays actuated in very early time frame?
10. Does hydrogen combustion occur in very early time frame?
11. Is filtered vent system actuated in very early time frame?
12. Is containment isolation recovered in very early time frame?
13. Does containment fail in very early time frame?

For early time frame:

14. Is core damage arrested in-vessel preventing vessel breach?
15. Does energetic FCI occur and fail reactor pressure vessel and containment?
16. What is the mode of vessel breach and core debris ejection process?
17. Does vessel rocketing occur and fail containment?
18. Is under vessel region flooded or dry at vessel breach?
19. Mode of under-vessel FCI following vessel breach?
20. Does hydrogen combustion occur at vessel breach?
21. Does containment fail at vessel breach?
22. Does filtered vent system actuate at vessel breach?

For late time frame:

23. Is AC power restored or maintained in late time frame?
24. Do sprays actuate or continue to operate in late time frame?
25. Do fan coolers actuate or continue to operate in late time frame?
26. What is the status of fans and sprays in late time frame?
27. Is core debris in a coolable configuration ex-vessel?
28. Does hydrogen combustion occur in late time frame?
29. Does containment failure occur in late time frame?
30. Does filter vent system actuate in late time frame?
31. Is containment basemat integrity maintained?
32. What is the mode of containment failure?
33. Time of core damage.

5.2.3.2 Containment event tree for Jose Cabrera NPP IPE

In the analysis a CET with the following top events was developed:

1. What is the induced RCS failure mode?
2. Is the containment isolated?
3. Is the core cooled in-vessel?
4. Does the containment fail in alpha mode?
5. What is the fraction of core ejected from the vessel?
6. Does the containment fail early?
7. Does the core coolable ex-vessel?
8. Does the containment fail in late time frame?

Some phenomena representing these top events are developed trough decomposition event trees, these are:

- RCS induced failures
- Containment isolation
- In-vessel core cooling
- Alpha containment failure
- Fraction of the core ejected from the vessel
- Early containment failure
- Ex-vessel core cooling
- Late containment failure and
- SG Tube Rupture location

Examples of parameters considered in the decomposition event trees are:

For early containment failure:

- containment pressure at vessel breach,
- containment pressure increase at vessel breach due to the primary coolant download,
- hydrogen generated in-vessel.

For ex-vessel core cooling:

- RCS pressure at vessel breach related to the core dispersion,
- cavity geometry configuration related to the corium bed thickness,
- melted core configuration,
- water available for core cooling from systems that were operable before vessel breach.

5.2.3.3 Containment Event Tree for Surry NPP

Containment Event Trees have been developed for each plant damage states. Specific events included in each CET were determined to a large extent by characteristics of the sequences in each plant damage state with which a particular CET is associated. The events/issues considered are:

Debris cooled in-vessel	Before Reactor vessel failure
In-vessel steam explosion	Before Reactor vessel failure
Mode/Time vessel failure	At/near reactor vessel failure
Direct Containment Heating	At/near reactor vessel failure
Early hydrogen burn/combustion	At/near reactor vessel failure
Debris dispersal out of cavity	At/near reactor vessel failure
Ex-vessel steam explosion/spikes	At/near reactor vessel failure
Liner melt-through	At/near reactor vessel failure
Debris cooled ex-vessel	Longer term
Late hydrogen burn/detonation	Longer term
Late containment over pressure failure	Longer term
Safeguards/auxiliary building	Longer term

Operator, recovery and mitigation actions considered are:

In-vessel injection restored	Before/after reactor vessel failure
RCS depressurised	Before reactor vessel failure
Power recovery	Before/after reactor vessel failure
Containment spray recovered	After reactor vessel failure
Containment heat removal recovered	After reactor vessel failure

5.2.3.4 Oconee Containment Event Tree

The CET consists of 11 top events with six top events are further developed with the aid of decision trees. The decision trees use success logic methodology. The following top events are considered in the CET:

1. Containment Bypass is prevented
2. Containment is isolated
3. Isolation failure size is small
4. Release is through auxiliary building
5. Early containment failure is prevented
6. Late containment failure is prevented
7. Containment failure is benign
8. Ex-vessel release of fission products is prevented
9. Containment failure from basemat melt-trough is prevented
10. Revaporization release is prevented
11. Fission product scrubbing is effective

The decision trees represent the basic events which can lead to a particular containment phenomena (early containment failure, ex-vessel fission product release, etc.). The benefit of this methodology is that the decision tree can be expanded to include more detail (i.e. basic events) without causing an expansion of the CET.

Developed top event using decision trees are:

- Early containment failure
- Late containment failure
- Ex-vessel fission product release
- Basemat melt-through
- Fission product revapourisation
- Fission product scrubbing

5.2.3.5 Examples of issue decomposition event tree

5.2.3.5.1 Temperature induced SGTR

Creep rupture failure of SG heating tubes is the result of heating the tube walls to a high temperature, which can only occur under condition of high RCS pressure and dry steam generator secondary side. Thus creep rupture failure of the steam generator tubes can be prevented by reducing the RCS pressure or maintaining an adequate SG secondary side water inventory.

Leakage can be terminated by keeping the secondary system pressure above the RCS pressure. In addition, flooding the containment to submerge RCS piping and flooding the SG to submerge the U-tubes would provide cold surfaces for fission product deposition and retention.

The SG water level is an important parameter to determine if the SG(s) are available as an RCS heat sink, and to determine if the creep rupture of the SG tubes is a potential concern. (If the SG tubes are covered, creep rupture is not a concern).

The SG pressure is another important parameter for the following reasons: 1) to determine if water can be injected into the SGs, and 2) to determine if creep rupture of the SG tubes is a potential concern. Maintaining a small or negative pressure difference from the primary to the secondary side is another method to ensure that creep rupture will not occur.

Measures are being considered for mitigating releases from SG when it is not isolated. By using primary side bleed and feed, a large fraction of the fission products is directed through the PORVs to the containment.

Guidelines proposed for depressurizing the RCS to prevent creep rupture of the steam generator tubes when the SGs are dry consider the following steps:

- Identification of the available means for depressurizing the RCS,
- Identification of the positive and negative impacts associated with depressurizing the RCS,
- Identification of the RCS depressurisation limitations,
- Determination of other mitigation actions if necessary and another RCS depressurisation path if needed,
- Identification of long term concerns due to depressurizing the RCS.

Decomposition of the induced SGTR issue is performed by considering the scenarios in which the primary system is at high pressure. It is the determined in which scenarios the SG rupture mode is likely to occur before others rupture modes, for example, vessel rupture or hot leg rupture.

The following example of shows the decomposition event tree for induced SGTR in SBO sequence:

<u>Top events.</u>

1. type of SBO
   - long-term
   - short term

2. Stuck open PORV or pump leak before core damage
   - stuck open PORV
   - Pump leak
   - None

3. Stuck open PORV or pump leak after core damage
   - None
     - Stuck PORV
     - none
   - Short-term
     - stuck PORV
     - 250 gpm leak
     - 480 gpm leak
     - none

4. Hot leg/surge line/vessel failure
   - no
   - yes

5. Induced SGTR
   - no
   - yes

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- Does the operator depressurise the reactor coolant system after core damage has occurred?
- Do the steam generator tubes remain intact? (accident class IA only)
- Does the hot leg nozzle/surge line fail due to high temperature creep rupture?
- (accident class IA only)
- Is the containment isolated?
- Is the passive containment cooling system operating?
- Is the hydrogen control system operating?
- Is the IRWST water flooding the reactor cavity?

- Does the molten core relocation to the lower plenum fail to produce a steam explosion which fails the rector vessel?
- Does the reactor vessel remain intact?
- Does the hydrogen generated in-vessel fail to burn globally?
- Does the ex-vessel core debris quench in the reactor cavity?
- Does short-term core-concrete interaction not occur as a result of core debris relocation to the reactor cavity?
- Does the containment remain intact during the dynamic phase of core relocation?
- Is containment water recirculated into the cavity for long-term debris cooling?
- Does the hydrogen in the containment fail to burn globally before 24 hours?
- Does the containment remain intact 24 hours after the onset of core damage?
- Does the hydrogen in the containment fail to burn globally between 24 and 72 hours?
- Does the containment remain intact 72 hours after the onset of core damage?
- Is the containment integrity not threatened after 72 hours after the onset of core damage?
- Does the containment not leak excessively?

5.3 Modelling of human intervention

As demonstrated by a number of PSAs, both qualitatively and quantitatively, human actions play a very important role in the safe operation of current Nuclear Power Plants (NPPs).

Therefore Human Reliability Analysis (HRA) becomes an extremely important task for the realistic assessment of the plant safety in PSAs. Unfortunately, human reliability is a very complex subject which cannot be addressed by fairly straightforward reliability models like those used for components and systems. Almost all of the methods that have been developed have been more or less criticised. In the last years, HRA experts have repeatedly stressed the need of development of second generation HRA methods. These methods should be based on a more balanced approach using a combination of experimentally derived data and insights (using both large-scale training simulators and small scale simulations) coupled with the use of formalised experts opinion elicitation methods rather than experts judgement-based quantification methods.

Examples of both types of methods with their limitations are given in detail in the Task 94-1 report.

All these methods have been developed and validated by interviewing and observing control room personnel performance when challenged by events potentially leading to plant damage states. During these events, all the operators responses are in more or less detail guided by unambiguous Emergency Operating Procedures (EOPs). So, even if uncertainties still exists in some areas, the described methods well represent the situations in which the operators are to perform preventive accident management actions.

This is not generally true for actions that can be effective in the mitigation of severe accidents, such actions are not always clearly addressed in the Emergency Procedures Guidelines (EPGs) or in the EOPs.

5.3.1 Consideration of preventive accident management

Strategies related to core damage prevention can be summarised in the following categories:

- control of reactivity
- control of RPV coolant inventory
- maintenance of coolant inventory
- maintenance of heat removal

All of these strategies can be accomplished in different ways: with combinations of systems and/or operator interventions and they are or can be well defined in the EPGs or EOPs. Under these conditions, the adoption of the HRA methods described in the Task 15 report, conservatively applied (for example considering reduced time window for the operator action in order to account for uncertainty in the phenomena) can give acceptable and at least comparable results.

5.3.2 Consideration of mitigative accident management

In most cases EOPs or EPGs cover just the first few hours of accident progression, after this time, the operator needs to be innovative and this, coupled with the incomplete understanding of phenomena (e.g. core melt arrest in vessel, hydrogen production and mixing, direct containment heating, core concrete interaction, steam explosions, etc.) increases the probability of carrying out unwanted, unnecessary or aggravating actions. This probability of error is even larger if the action requires violation of Technical Specification requirements or of the plant equipment design basis.

The mitigative strategies can be grouped in the following main categories:

- prevent vessel failure
- prevent containment failure
- limit the release to the environment

Many different opinions exist on how to best implement those strategies, specially for the first two categories, and available EOPs or EPGs do not give unambiguous, complete and correct directions for implementation using the existing equipment. Under these circumstances HRA methods developed for preventive strategies actions cannot be easily adopted to the analysis of mitigative strategies (Performance Shaping Factors can dramatically change depending on the different situations).

6. EVALUATION OF LEVEL 2 PSA MODELS AND QUANTIFICATION

A Level 2 PSA makes it possible to assess in an overall survey the special plant-specific characteristics and potentials, the phenomena determining the accident sequence, the measures taken by the personnel and to quantify the uncertainties associated with all these issues. Analyses of this kind require a great effort and can hardly be performed without simplifications. This chapter discusses the possible methods with regard to their indented application, namely for the development of accident management measures and the probabilistic assessment of their feasibility, effectiveness and reliability within an energy spectrum that is as wide as possible.

6.1 Brief description of Methods

There are several approaches to performing the accident progression portion of a Level 2 PRA, some have been used in past risk assessments and some are speculative. Nine approaches to implement the logic framework for tracing accident progression (logic PSA codes) are listed and briefly described below.

– Containment event tree (CET), also called accident progression event tree (APET) method.

  The logical structure for the modelling of the progression of the accident in the reactor system and in the containment is provided by containment event trees (CET). In some studies these are termed "accident progression trees" (APET). In a structured approach, the interdependent physical-chemical processes are traced that are relevant to the integrity and retention capability of the containment. The questions asked at the branch points of the event paths are ordered chronologically; they characterise the various possibilities of accident progression inside the containment. The quantification of the branching probabilities addresses the availability of containment systems, as well as the physical phenomena. It provides a conditional probability for each accident path, originating from a PDS, and ending at an APB. The logical structure of the CETs is analogous to that of the system event trees in level 1 analyses.

  Two main methodologies are employed for the development of the CETs:

  – The large CET, which contains virtually all top event questions regarding the specifics of severe accident modelling; and

  – The small CET method, which includes top event questions concerning the major severe accident phenomena, which are then supported by fault trees.

  In principal, neither method is more accurate or complete, but the small CET method is much better traceable, and considerably easier to review.

In the PSAs examined in the context of this report, the number of branch points differs greatly, it varies between 9 and 145. For the quantification of the branching probabilities, calculations and analyses of varying complexity are performed, using mechanistic computer codes, parametric codes and engineering judgement (for phenomenological questions), as well as systems analysis codes (for questions of availability of systems). The number of branch points, by itself, is not a measure for the depth and degree of detail of a level 2 investigation. With a compact CET having the essential questions of accident progression in the event tree, and associated fault tree-like analyses, the same analysis quality and completeness can be obtained, as with very complex and large CETs.

The questions asked at the branch points of the CETs often are of global nature, i. e. "amount of zirconium oxidised in the pressure vessel?", "amount of core material, that is released into the containment following failure of the reactor pressure vessel?", "is the molten material coolable?" There are many instances where such questions are not answered by using mechanistic computational models. Instead, engineering judgement is used to assign subjective probabilities to the branch points of the CETs. When mechanistic models are not directly used, the dependency of the results of the accident progression analysis on the underlying physical phenomena is often hidden.

The information that is available to model and quantify the progression of accidents consists of a variety of research results including numerous calculations with computer programs that model special important aspects of the accident progression, as well as experimental results.

The flexibility and the generality of the CET method makes it a powerful tool for conducting level 2 analyses. The possibility of defining questions at various analysis levels makes the CET efficient. At the same time, however, this possibility has to be used with caution, especially when it is applied to questions with a poor knowledge base.

– Fault tree method

The fault tree method is a widely used method in which the event whose frequency of occurrence is to be determined, e.g. system and component failure, is put at the top (TOP event) of a logic diagram (the fault tree) and in which the failures that can lead to such an event are identified. By representing complex interrelations through binary logic and suitable graphic representation, the fault tree analysis enables the treatment of very large systems. If applied consistently, it provides by its deductive procedure all the event combinations which lead to the undesired event. Limits are not set by the method itself but only by the know-how and the care of the user. Experience in using this method is extensive. The drawback in applying it to a Level 2 PSA is that the TOP events (i.e. the numerous damage states of the containment) would have to be defined in advance. Moreover, the possibility of taking the phenomenological uncertainties into account in the main computer programs would still have to be created. So far, this method has not been used for accident sequence analyses.

– Markov model

The Markov model is a method to trace processes which may take on different states during the course of time. For this purpose, the set of all states that have to be considered in the analysis have to be previously defined by the user, including the initial, interim and final states which the analysed system or accident sequence may take on during the course of time.

The different states are linked by the probabilities of transition from one state to another. In this context it is possible that one state that has already may be reached again. For example, it would thereby be possible to deal with the repeated activation of systems or the multiple melting and subsequent cooling of core debris. In the Markov model, this transition between the different states is represented by a matrix of probabilities which is called "stochastic transition probability matrix" (STPM). The "transition rates" are conditional transition probabilities within a given time, depending only on the previous state. Discrete states and discrete observation times form so-called Markov chains.

Mathematically, the Markov model is described by a system of linear differential equations. The coefficients are the transition rates that have to be previously defined. The solution of the system is expressed as a vector of state probabilities.

However, problems may arise in practical application due to the possibly existing large number of system states which may occur in real systems. In such a case it will be necessary either to split the system into different independent partial systems, to summarise several systems in macro states if there are any symmetries, or to define a more favourable structure of the transition matrix by prescribing a suitable order to the different states. However, the Markov model still remains a costly to handle instrument even after such effort-reducing measures.

– Influence diagrams

This method is a more recent development which can be seen as an alternative to accident sequence trees. Influence diagrams are a graphic and mathematical representation of probabilistic conclusions and decision alternatives. An influence diagram is a network which describes the structure of the model that is to be analysed, e.g. an AM measure, as a decision problem. This means that influence diagrams are directed graphs in which uncertain parameters of a decision model (one special case of decision model is the event sequence analysis) are shown as nodes and the direct dependencies between the individual nodes and the information available at the time of the decision are represented as directed arcs, with the detailed information about each variable being stored in the corresponding node. They contain two types of directed arcs, namely conditional and informal ones, which represent the conditional dependency between the chance parameters and the time of the arrival of information and generation. There are four types of nodes: decision nodes, chance nodes, deterministic nodes, and value nodes. The decision nodes describe decisions which may occur at these points. The chance nodes are linked to a set of possible terminations and the probabilities for these terminations, thus representing uncertain parameters. The deterministic nodes are the functions of values of previous nodes (they can therefore be considered as a special variety of the chance node). The value (or usefulness) nodes represent a final result. In its form of representation the influence diagram is much clearer and simpler than an event tree.

– Checklist approach

This approach would be taken when funds are not available to perform a full blown level 2 PSA. In this approach, insights and plant features important to accident progression analyses are catalogued from previous PSAs. The plant features and insights are compared to the checklist one by one for the plant in question. The features and insights believed to be important would be documented and assessed further if necessary.

- Extrapolation approach

  This approach would begin from existing models and use existing data. The models and data would be modified appropriately for the plant in question. An approach similar to this one has been recommended in the Individual Plant Examination Submittal Guidance (NUREG-1335). Theoretically, the results given by this approach could be at the same level of detail as the surrogate approach from which the models and the data are based.

- Issue rotation

  Although not part of a full scope Level 2 PRA, a methodology has been proposed to resolve specific Level 2 issues. A particular phenomenon is targeted. and is assessed using methods that combine probabilistic analyses, analytical methods, and experimental results. This method differs from the full blown detailed model method, in that one particular phenomenon is addressed and all resources are directed toward that one phenomenon.

Appendix B provides a summary description of some commonly used logic PSA codes.

6.2 Use of Expert Judgement

The referencing to publications in this section refers to the list in subsection 6.2.4.

6.2.1 Introduction

Expert judgement (EJ) is typically used when:

- relevant data are incomplete, or scarse, or exhibit high variability;
- operating experience does not exist;
- there is no generally accepted state-of-the-art;
- experts question the applicability of data and models;
- the complexity of the issues calls for a wide spectrum of expertise.

Expert judgement, used to incorporate many disparate types of knowledge into a coherent evaluation, is present at all level of the PSA analysis. Expert judgement is initially required to determine whether a problem deserves attention; then, judgement is needed to understand the dimensions of the problem, to develop alternatives, to decide what data to collect, to choose which models to build, to interpret the results of data collection efforts and of calculations, and to integrate the information needed for the analysis and solution of the problem (references 1. to 5.).

It should be remembered that the principal aim of the EJ process is to generate unbiased probability distributions for uncertainties, (best estimates are then obtained as by product). In this respect, it is recalled that the probability frame for uncertainty representation is widely used, but different, alternative formalisms are also proposed in literature (belief functions, possibility distributions, uncertainty factors, etc.). Furthermore, new lines of research point toward a reconsideration of the whole EJ problem: expert opinions are considered not as statistical data, but as the result of intensive cognitive processes, and as such represented (reference 6.).

6.2.2 The Need for Structured Expert Judgement

The EJ process, being incorporated in the PSA analysis, should obviously comply with the PSA needs of:

- accountability: all the sources of information should be clearly identifiable
- reproducibility: the results of the study should be traceable;
- scrutability: assumptions and methods should be understandable and credible, hence scientifically defensible.

Further, the EJ-process should comply with the principles of *neutrality*, i.e. the method for combining and evaluating experts opinion should encourage experts to state their true opinion, and *fairness*, i.e. all experts should be treated equally, at least in principle (reference 3.).

In order to fulfil the stated requirements, an explicit, formal and structured process should be put in place for EJ use. A structured EJ process could offer several advantages, namely (reference 4.):

"First, an explicit approach can provide the expert with means to process the multitude of information associated with complex technical questions. For example, issues can be broken into logical parts that can be more easily considered. Second, the explicit process is more likely than its implicit counterpart to use the body of research on human cognition and communication. This practice usually enhances the quality of the expressed judgements. Third, the procedures of the explicit approach provide the record of the experts' judgements, and of their rational for arriving at these judgements. This documented record allows the judgements obtained by the explicit process to be more easily updated as new information becomes available. Fourth, people other than those immediately involved can scrutinise the explicit process and its results. With the implicit approach, there is little to review and, indeed, reviews are rarely performed. Thus, the explicit approach is more likely to advance to the preferred review process."

There are some fairly advanced approaches to deal with EJ in PSA. One of the most extensive is the method applied in NUREG-1150-project, (reference 2.), and many of the other approaches are versions of that method.

In order to improve the identification of areas where EJ can be utilised and to encourage the use of proper analysis tools, the Joint Research Centre of the European Union, within the context of IV EC Framework Programme, has organised a Benchmark Exercise on Expert Judgement Techniques in level 2 PSAs (BE-EJT) (reference 7.). The methods benchmarked within phase 1 of BE-EJT were:

1) the NUREG-1150-methodology, (USA) (references 2. & 8.);
2) the methodology used by NNC (UK), based on quality assurance methods of the source of information and of the problem solving process;
3) GRS (D) methodology, (reference 9.), based on extensive use of physical analysis codes and on sensitivity and uncertainty analysis performed by means of proprietary codes,

4) the STUK-VTT (FIN) methodology, (reference 10.), based on the NUREG-1150-approach with Bayesian aggregation of experts distributions,
5) JRC-ISIS knowledge based approach, based on knowledge engineering techniques (reference 6.).

6.2.3 Procedural Framework for Expert Judgement

Although no generally accepted procedure exists for elicitation of expert judgement, the key elements of such a procedure consist of (in accordance with references 1. to 5.:

Selection of issues

Extensive lists of potential issues are prepared and submitted for discussion and screening to expert panels. The criteria used for issue selection should be:

- High impact on risk. Interest within the reactor safety community.
- High impact on uncertainty.

Selection of experts

With the intended goal of incorporating maximum expertise and, at the same time, a diversity of expertise and of points of view, the following criteria should be adopted to select experts:

- List of publications.
- Wide variety of experience, obtained in universities, consulting firms, laboratories, nuclear utilities, government agencies.
- Wide perspectives.
- Willingness to participate in the formal expert judgement process.

Elicitation training

A training session in probability theory and in techniques for eliciting probabilities should be performed in order to acquaint the experts with the concept of subjective probability and more confident in expressing their beliefs as PDFs.

Psychological aspects concerning probability assessment, the so-called "biases" (e.g., overconfidence, representativeness, overestimation, etc., references 3. & 11., should be presented to the experts as well as problems related to group behaviour.

Presentation of issues

Plant analysts present the problem at hand in order to reach a common understanding of the problem itself, and, in particular, of the relevant question. Suggestion for decomposition should be given too, and be subjected to discussion. Experts should be encouraged to search for alternative decompositions or to modify the proposed ones.

Preparation and discussion of analyses

An adequate span of time should be allocated for the experts to prepare their individual assessments and supporting documentation. To make sure that requirements of accountability and reproducibility are fulfilled, the format of reporting the individual analyses should be specified. If deemed necessary, discussions about the proposed issue decompositions and methods adopted for issue resolution could be scheduled before individual elicitation.

Elicitation

Normative experts individually elicit the panel members. Different types of elicitation questions include discrete and continuous random variables. Broadly speaking, the fixed quantile method involves assessing values of the variable at given cumulative probabilities, while in the fixed value method the values of the variable are provided as input and the probabilities associated with the intervals are assessed by the expert.

The normative experts codify the expert opinions in complete and consistent form in probabilistic terms.

Aggregation of results

It should be noted that experts often give their results in different formats, and only after their distributions are homogenised, some averaging procedure could be applied to get the aggregated opinion, in the form of CDFs. One of the averaging methodsis weighted averaging, i.e.:

$$P(x) = \sum_{i=1}^{n} w_i P_i(x)$$

or

$$P(x) = \prod_{i=1}^{n} P_i(x)^{w_i},$$

where $P(x)$ is the aggregated (consensus) distribution for the uncertain variable x,
n is the number of experts consulted
$P_i(x)$, $i=1,\ldots,n$ is the probability distribution of x according to the i-th expert
w_i, a weight ascribed to the i-th expert ($w_i>0$, $w_1+ w_2+\ldots+ w_n=1$).

The first case, based upon arithmetical averaging, is known as the linear opinion pool; the second, based upon geometric averaging, is known as the logarithmic opinion pool. Different methodologies for weight assignment can be found in the literature (e.g., reference 3.).

Alternatively, the analyst could resort to Monte Carlo methods or Bayesian approaches (see e.g. references 9., 12. & 13.).

Review

In order to correct potential misunderstandings and to ensure that elicited judgements actually reflect experts' opinions, written analyses of each issue should be re-submitted to the special issue's experts by the normative experts.

Documentation

Openness, impartiality, and tractability should guide the production of the documentation.

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13. Pulkkinen, U. *Statistical Models for Expert Judgement and Wear Prediction*. VTT Publications 181. Technical Research Centre of Finland, Espoo, 1994. 65 p.+ app. 80 p.

6.3 Uncertainty Issue Quantification Technique

The referencing to publications in this section refers to the list in subsection 6.3.3.

6.3.1 Definition and types of uncertainty

In principle, a PSA should investigate all possible accident scenarios. However, many scenarios involve phenomena that have not been studied through experiments or observations, and therefore, there exists a fundamental uncertainty in the results of a PSA. A thorough uncertainty analysis can identify areas which need further investigation or special attention (vulnerabilities). Furthermore, if the PSA generates point estimates, an uncertainty analysis may contribute to the credibility of these results.

Typically, three classes of uncertainties are identified; for details, see, for example (reference 1.):

- Parameter (data) uncertainty

  Uncertainties associated with the values of basic parameters like initiating event frequencies, component failure rates, human error probabilities included in the determination of event sequence frequencies. The uncertainties are characterised by probability density functions of the parameters. By Monte Carlo methods, frequently in the form of Latin Hypercube sampling (LHS) (reference 7.), the uncertainties are propagated through the analysis steps to generate a probability distribution for the end result.

- Model uncertainty

  Uncertainties associated with phenomenological models for the physical-chemical processes and related assumptions. Such uncertainties may be modelled similar to parameter uncertainties, or by defining multiple branching representing different (uncertain) sequence paths emanating from a nodal point.

- Completeness uncertainty

  These uncertainties reflect limitations of scope or truncation effects. In principle, such uncertainties can not be quantified within a given PSA scope, but by performing additional analyses of excluded events their insignificance can be demonstrated.

6.3.2 Treatment of uncertainty

6.3.2.1 General

There are several approaches to conduct an uncertainty analysis: The analysis may be qualitative with the prime objective of identifying and ranking the most important uncertainties. This may include limited sensitivity analyses and can be performed prior to a quantitative uncertainty propagation analysis.

Sensitivity analysis is defined as the degree of change in the results due to changes in data or in modelling assumptions, for which it is particularity suited. This analysis may rank and identify the major contributors to uncertainty. An example technique is the computation of several importance measures (e.g. risk reduction, risk increase factors).

If probability distributions are selected as the form to express the uncertainties a choice will have to be made between the classical (frequentistic) or subjectivistic (Bayesian) approach to statistics.
Furthermore, a decision has to be made if a formal uncertainty propagation analysis is conducted or just a limited uncertainty analysis i.e. only an identification of major uncertainties.

A General uncertainty analysis can be divided into the following steps:

- *Definition of the scope of the uncertainty analysis:*
  Because the number of uncertainties is very large, and the resources are limited, it is important to make a selection of the issues to be included, e.g. on the basis of the computation of importance measures, limited sensitivity analyses, and available data.
- *Characterisation/evaluation of each uncertainty issue:*
  The format and the range of the uncertainty parameters of each issue have to be defined e.g. probability distributions or just bounds. Subsequently, the impact of each issue may be evaluated.
- *Propagation/combination of the uncertainties:*
  If desired, the method of propagating all the important uncertainties through the different steps of the PSA will have to be selected (e.g. straightforward Monte Carlo or Latin Hypercube Sampling, Method of Moments (references l. & 2.). In any case, the uncertainties at the different levels have to be combined to estimate the overall uncertainty in the final results.
- *Display and interpretation of the results of the uncertainty analysis:*
  Finally, the uncertainty in the overall results can be displayed by e.g. probability distributions, or just a mean value (or median) in combination with some quantities (e.g. 5 and 95 percent percentiles). From this, the principal sources of uncertainty are not easily identified. Therefore, at the intermediate levels, appropriate uncertainty measures have to be computed or qualitatively assessed.

An excellent framework for uncertainty analysis which can be generally applied, is presented in reference 2**.**

6.3.2.2 NUREG-1150

General Approach to Uncertainty

An important characteristic of the NUREG-1150 study (reference 4.) is the extensive treatment of uncertainties and uncertainty propagation in the PSAs. Uncertainty in these analyses comes from every step in the analysis, and it can be both quantitative and qualitative in nature. Sources of uncertainty are, amongst others, an inherent random variability of data or processes, or a lack of knowledge regarding data, modelling assumptions, and completeness of the analysis. Once identified, the impact of uncertainties can be propagated through the models and through the principal steps of the risk analysis. The uncertainty analysis of the NUREG-1150 study was performed according to the steps mentioned in section 6.3.2.1. Physical parameters were treated as data uncertainties. The uncertainty analysis included:

- the definition of the scope of uncertainty analyses:
- the definition of specific uncertainties,
- the development of (subjective) probability distributions,

– the combination and propagation of the uncertainties; and

– post-sensitivity analysis, based on stepwise rank regression analyses.

For data, the parameters of interest are failure rates, component unavailabilities, initiating event frequencies, and human error probabilities, Modelling of uncertainties includes success criteria, failure logic in fault trees, and phenomenological processes with their impact on system performance. Expert panels (reference 4., 8., & 9.) have estimated the probability distributions for parameters which have a large uncertainty and are important to risk. This approach has been applied in particular to the level 2 analysis.

For uncertainty propagation, the Latin Hypercube Sampling (LHS) method has been selected. LHS, which has advantages compared to straightforward Monte Carlo sampling is outlined here. Its application in the NUREG-1150 analyses is illustrated for the Surry PSA.

Latin Hypercube Sampling

Problem Description

Consider a particular variable Y which is of interest in a risk analysis. Let Y be a function of K input variables X_1 ,X_2. X_k:

$$Y = h(X), \quad X = (X_1 ,X_2. X_k)$$

The function h may be quite complicated e.g. a mathematical or numerical model. In a PSA, the variable Y can be a core melt probability, a containment failure probability, a release fraction of a radionuclide, etc. Now, the question is: how does Y vary when the input variables X_1 ,X_2. X_k vary according the assumed joint probability distribution F(X). Here, the joint probability distribution F(X) reflects, for example, uncertainties in input data or modelling assumptions. A conventional method is straightforward Monte Carlo sampling, also called Simple Random Sampling (SRS) in which the observations for Y are sampled from the joint probability distribution F(X). An alternative method is LHS which is extensively described in reference 7., and will be outlined below.

Method Description

Suppose that the sampling of N observations of Y is required. For this purpose, LHS can be applied. LHS selects N different values for each of the K input variables X_1 ,X_2. X_k, by the following 3-step procedure:

1. the range of each X is divided into N non-overlapping intervals of equal marginal probability 1/N;

2. for each interval, one sample is selected randomly but taking into account the probability density in that interval; and

3. the N values for X_1 are paired at random with the N values for X_2 forming N pairs of values for the pair (X_1, X_2) which are combined at random with the N values for X, to form N triplets, and so on to form a set of N K-tuplets; this set is the Latin Hypercube sample.

In Figure 6.3.2.1-1, this procedure is illustrated for two variables (K=2) and five intervals (N=5). The five intervals are randomly paired: (1,3), (2,5), (3,2), (4,1), and (5,4).

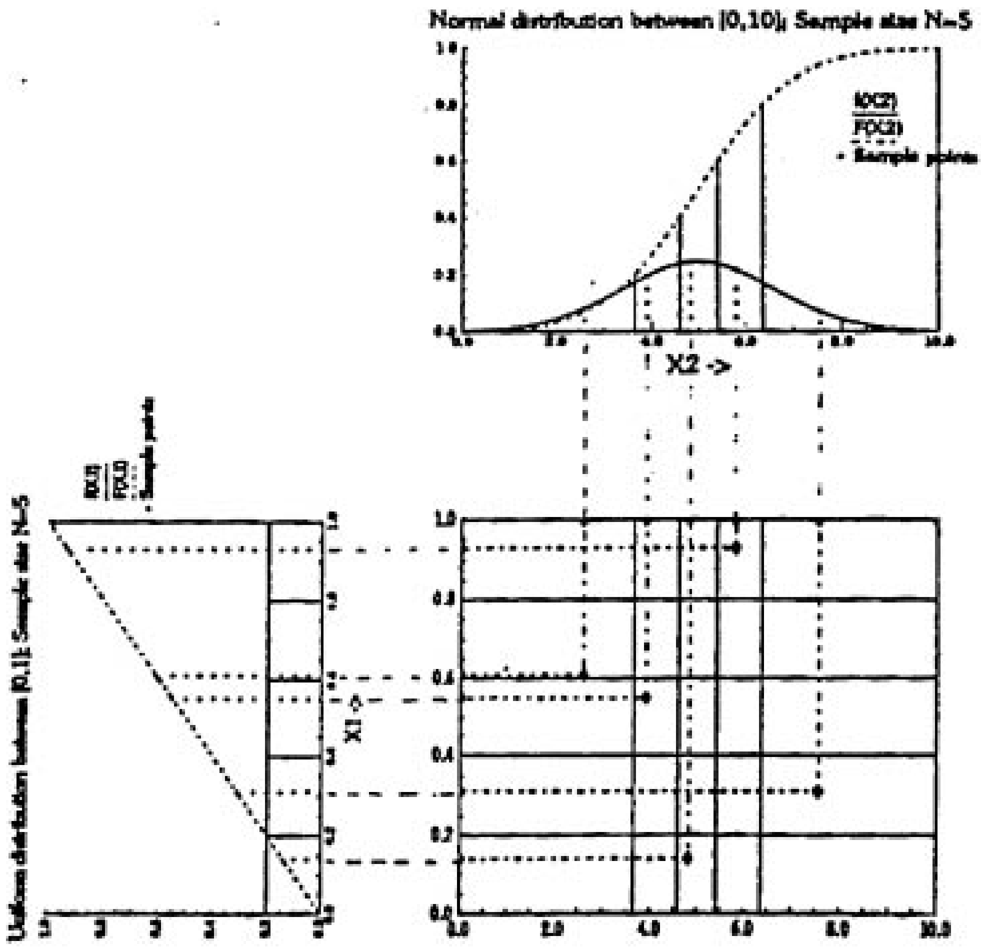


Figure 6.3.2.1-1. Example of Latin Hypercube Sampling

Two major advantages of LHS over SRS are that:

- LHS forces to include samples from the tails of the distributions, and
- in case Y is dominated by only a few components of X, LHS ensures that each of these components is represented in a fully stratified manner, no matter which components might turn out to be of importance.

To conclude this section two important properties of LHS will be given. Consider the class of estimators

$$T_g\,(Y_1, ., Y_n) = 1/N \sum_{n=1}^{N} g(Y_n),\ g\ \varepsilon\ G$$

where G is the set of all functions. It can be shown [6] that:

- if the Yn's constitute a LHS sample from the distribution of Y = h(X), then LHS yields an unbiased estimator of E(Tg),
- if Y = h(xl, x2, .' x,) is monotone in each of its arguments, and g(Y) is a monotone function of Y, then it holds that Variance(T G)>LHS Variance(T,)", thus, LHS yields estimates with smaller variance.

Other advantages of SRS over LHS are:

- the direct computability of confidence limits on estimates of mean value, variance, percentiles, etc. (see e.g. reference 3.),
- the ability to aggregate independent samples of different sizes to sequentially arrive at a sufficiently large sample, and
- less undesired correlations between the sampled variables compared to LHS.

Application of LHS in the NUREG-1150 study

Principal Steps of a NUREG- 1150 PSA

The NUREG-1150 PSAs consist of five principal steps, viz.:

(1) accident frequency (systems) analysis,
(2) accident progression analysis,
(3) radioactive material transport (source term) analysis,
(4) offsite consequence analysis- and
(5) risk integration.

The final stage of the PSA is the assembly of the outputs of the first four steps into an expression of risk as follows:

$$Risk_{In} = \sum_{h=1}^{nIE} \sum_{j=1}^{nPDS} \sum_{j=1}^{nAPB} \sum_{k=1}^{nSTG} f_n(IE_h)\ P_n(IE_h PDS_i) P_n(PDS_i APB_j) P_n(APB_j STG_k) C_{ik}$$

h=1, i=1, j=1, k=1

where:

n = the sample number in the LHS scheme;
n_{IE} = the number of initiating events;
n_{PDS} = the number of plant damage states;
n_{APB}= the number of accident progression bins;
n_{STG}= the number of source term groups;
$Risk_{IN}$ = the risk of consequence measure I for sample n (consequences/year);
$f_n(IE_h)$ = the frequency (per year) of initiating event h for sample n,-
$P_n(lE_h \text{ -> } PDS_I)$ = the conditional probability that initiating event h will lead to plant damage state i for sample n;
$P_n(PDS_I\text{->}APB_J)$ = the conditional probability that plant damage state i will lead to accident progression bin 1 for sample n
$P_n(APB_J \text{ ->}STG_k)$ = the conditional probability that accident progression bin j will lead to source term group k for sample n; and

C_{lk} = the expected value of consequence measure i conditional on the occurrence of source term group k.

The risk integration is shown in matrix formulation in Figure 6.3.2.1-2. The approximate numbers of PDSs, APBs, and STGs, and the number of consequences used in the different NUREG-1150 PSAs are 20, 1000, 50 and 8, respectively.

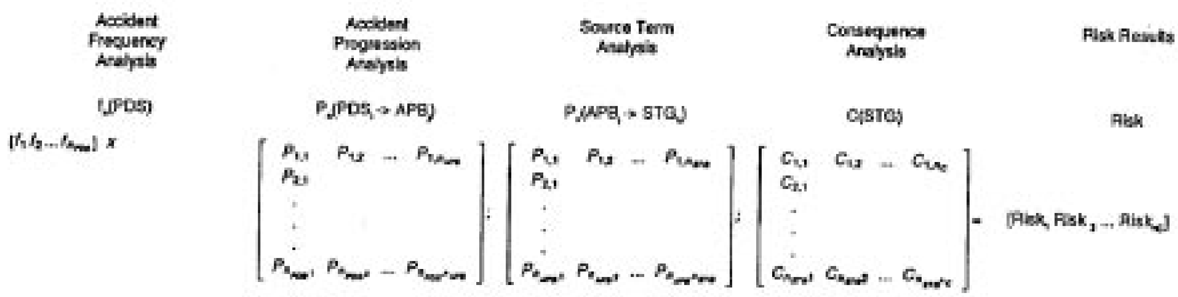


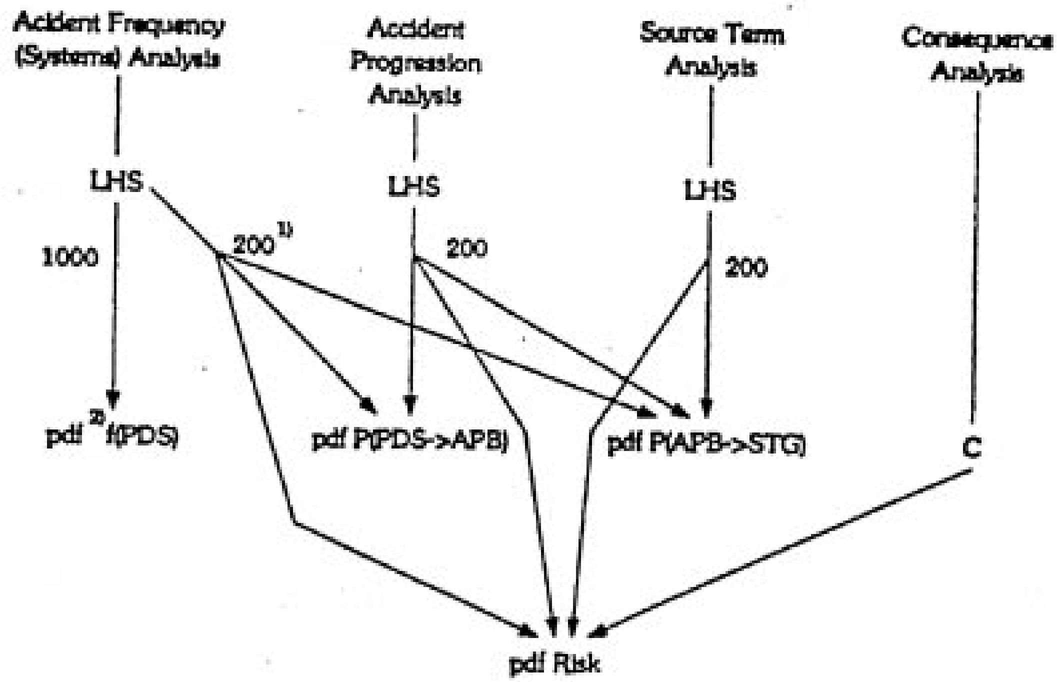
Figure 6.3.2.1-2 Example of Latin Hypercube Sampling

Propagation of Uncertainties through LHS

The uncertainties have been combined and propagated by a specialised LHS code (reference 7.) in which credit is given for statistically correlated parameters e.g. for generic categories of components or of basic events (reference 8.). This has been applied for the accident frequency analysis, the accident progression analysis and the source term analysis, in order to quantify the impact on the total risk. Expert judgement has been an important source for the probability distributions of the parameters. Except for the usual sampling of weather sequences, no probability distributions have been derived for parameters in the consequence analysis; only expected values for the consequence measures have been calculated. This sampling process is outlined in reference 4. and is described in more detail in reference 10. Figure 6.3.2.1-3 shows the propagation of uncertainties through the principal steps of the risk analysis.

Notes:

1. The numbers denote the sample size used in the Surry PSA.
2. pdf. probability distribution function, may be in the form of frequency histogram, mean, median, or quantiles.



Notes:

1) The numbers denote the sample size used in the Surry PSA.
2) pdf. probability distribution function, may be in the form of frequency histogram, mean, median, or quantiles.

Figure 6.3.2.1-3 Propagation of uncertainties through the principal steps of the NUREG-G- 1150 PSAs (Note the different levels at which LHS is applied).

Example Sample Parameters Surry PSA

The LHS process of the Surry PSA will be outlined to give an idea of the variables sampled throughout the various steps of the PSA (reference 5.). Table 6.4.1 presents an overview of these variables.

For the accident frequency analysis, most variables are sampled according to a lognormal distribution. Only two variables were correlated i.e. the probability that the diesel generator fails to run for 1 hour and for six hours. Two LHS samples have been computed: one with a sample size of 200 to be used in the risk integration and one with i sample size of 1000 to be used in the generation of the probability distribution functions of the PDSs,

For the accident progression analysis, most variables are sampled according to the aggregate (distributions derived from the expert panels. Most of the variables are correlated. In this case, also two LHS samples have been computed, both with t sample size of 200. One sample has been used for the risk calculation, while with the other sample the robustness of the sample method and sample size has, been tested. The results for the probability distributions of the APBs were fairly close.

For the source term analysis, most variables are sampled according to the aggregate distributions derived from the expert panels. The variables are uncorrelated, but the values for different radionuclide classes are completely correlated e.g. a 0.05 percentile for iodine in case of low zirconium means that also 0.05 percentiles of the distributions for other radionuclide groups are chosen. The sample size was 200.

Table 6.3.2.1-1. Variables sampled in the Surry PSA for internal events

Analysis Step (Number of variables)	**Types of variables**
Accident Frequency Analysis (46)	• Initiating event frequencies (11) • Failure probabilities diesel generators (3), scram/actuation (4), pumps (6), valves (11) • Common Cause failures (6) • Human error probabilities (5)
Accident Progression Analysis (49)	• Branching probabilities APET e.g.: – offsite power recovery – core melt arrest before vessel breach – PORV or RCS SRV sticks open – RCP seal failure – hot leg or surge line failure • Parameters of physical processes e.g.: – fraction of zirconium oxidation – fraction of core involved in high-pressure melt-ejection – Pressure rise – RCS pressure – Containment failure pressure • Mode of vessel failure
Source Term Analysis (12)	• Certain parameters of XSOR algorithm: – fractions of each fission product group • released from core to vessel, from vessel to the containment, from vessel to the steam generator, and to the environment • available for CCI • released from CCI • deposited in the RCS • released to the containment as aerosol particles at vessel breach in a DCH event – decontamination factors • pool scrubbing in interfacing systems LOCA • sprays • overlying pools of water in CCI – late iodine release

6.3.2.3 QUASAR Program

A somewhat different approach, mainly with respect to the determination of probability distributions, was taken within the QUASAR program (Quantification and Uncertainty Analysis of Source terms for severe Accidents in light water Reactors). The approach followed the general approach, with a selection of specific techniques at each step:

- Screening sensitivity analysis:
  The screening is aimed to determine the relative significance of each input parameter and to reduce the number of model parameter, for which an extensive analysis is needed.
  The sensitivity analysis described is based on regression techniques involving the statistical controlled covariation of input parameters, avoiding the more traditional technique in which parameters are varied one it a time, which cannot account systematically for the joint effects of the individual-parameters.

- Quantification of uncertainty:
  Uncertainties about the true values of parameters are quantified by treating them as random variables with appropriate probability distributions. Typical problems arise due to the fact that the sparseness of date precludes reliance on notions of experimental population variability in constructing distributions; and the subjective approach to formulating distributions gives an unwarranted impression of precision, thus understating, the degree of uncertainty. The use of expert opinion is necessary.
  The formulation of the probability distributions used in the QUASAR program is based on principles from information theory. The aggregation of the individual expert opinions was also governed by information-theoretic principles.

- Propagation of uncertainty
  The uncertainty distributions for each significant parameter is propagated through the models by Latin Hypercube Sampling.

- Analysis of output distribution sensitivity*:*
  Post-uncertainty sensitivity analyses are performed by response surface regression techniques which relate the model outputs and inputs through polynomial forms, and which result in an importance ranking.

6.3.3 References

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7. 1 Imam, M.J. Shortencarier, 'A FORTRAN 77 Program and User's Guide for the Generation of Latin Hypercube and Random Samples for Use with Computer Models", NUREG/CR-3624, SAND83-2365, March 1984.

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11. Khatib-Rahbar et al.: 'A Probabilistic Approach to Quantifying Uncertainties in the Progression of Severe Accidents'. In: Nucl. Sc. & Eng., Vol. 102, pp. 219-259, July 1989.

7. INTEGRATED AND PSA INFORMED APPROACH TO DECISION MAKING

7.1 Introduction

One of the strengths of PSA is that it allows to develop integrated models of the responses of a plant's safety systems and additional provisions for accident mitigation to a broad spectrum of challenges. The models developed can be used for the validation of design on the background of the most recent state of knowledge, for the plant specific interpretation of operational experience as well as recent safety analyses, and of current research results on safety and risk relevant phenomena. Furthermore, the models can be applied to the assessment of the significance of safety issues and vulnerabilities as well as to the judgement of proposed plant modifications and the implementation of new technological means.

As PSA provides a broader and deeper understanding of safety and risk relevant issues than deterministic methods alone, it is increasingly used for optimisation of the various levels of defence, and for the optimal allocation of available resources.

However, most of these applications have been in the context of case by case decisions.

A more recent development is to make use of PSA in a systematic integrated approach that integrates probabilistic and deterministic considerations, called „risk informed decision making": In summary its elements can be characterised as follows:

- Design criteria that implicitly involve probabilistic considerations are complemented by explicit probabilistic arguments clarifying design objectives.
- Weaknesses and vulnerabilities of a design can be identified and judged against design objectives.
- Various options available for improving safety can be quantitatively assessed and compared, also with respect to cost effectiveness.
- Decisions concerning reliable assurance of safe operation and control of risk can be based on additional justification.

The application of risk informed decision making requires that certain quality standards are obeyed by the PSA studies, and that guidance is available on how to resolve potential conflicts between probabilistic and deterministic considerations.

This chapter summarises

- Recent activities and publications in the field.
- Important attributes of PSAs used in the context of risk informed decision making.

- Current national positions on risk informed decision making.
- Specific examples of risk informed decisions.

Some of the decisions were made before the introduction of the concept of risk informed decision making. In several cases decisions were made on basis of PSA results for similar plants or on basis of generic probabilistic results. Others were instigated by deterministic considerations, but with subsequent confirmation by probabilistic analyses.

PSA informed decisions involve the level 1 as well as level 2 domain. In this summary, decisions concerning the level 2 domain are described. in section 7.6. The examples include risk informed decisions made by regulators and licensees.

7.2 Recent activities and publications related to risk informed decision making

Aspects of risk informed decision making are discussed, for example, in

- Compilation of Selected Modifications and Backfits in German, Swedish and US Nuclear Power Plants, SKI Report 95:25, December 1995
- PSA Based Plant Modifications and Backfits, CSNI-PWG5, Task 94-3 (Draft Report), 1996
- Proceedings of Executive Meeting on Risk-Based Regulations and Inspections, 12-14 August 1996, Stockholm, Sweden, SKI/HSK
- Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, NUREG-1560, Vol. 1, Chapter 8, Vol. 2, Chapter 14, Draft, April 1997.
- 1997 CNRA Special Issue Report (Draft) on Review Procedures and Criteria for Regulatory Applications of PSA, May 1997
- OECD/NEA/CSNI/R(97)1, Regulatory inspection activities related to inspection planning, plant maintenance and assessment of safety, Proceedings of an international workshop, Chester, UK, May 1996.
- USNCR Standard Review Plan, Use of probabilistic risk assessment in plant-specific, risk informed decision making: (Draft) March 1997
- Draft regulatory guide DG-1061, An approach for using probabilistic risk assessment in risk informed decisions on plant-specific changes to the current licensing basis, June 1997
- Draft regulatory guide DG-1062, An approach for plant-specific, risk-informed, decision making: in-service testing, June 1997.
- Draft regulatory guide DG-1064, An approach for plant-specific, risk-informed, decision making: graded quality assurance, June 1997.
- Draft regulatory guide DG-1065, An approach for plant-specific, risk-informed, decision making: technical specifications, June 1997.

7.3 Quality requirements for PSAs

The use of PSA results in the PSA informed decision making process requires an appropriate quality level of the involved PSAs. Such requirements are presently being prepared in

- Finland: Regulatory Guide YVL 2.8
- the Netherlands:
  - Dutch Procedures for Conducting Level 1 PSA (Final report)
  - Dutch Procedures Guide for Conducting Level 2 PSA (draft)
  - Level 3 PSA Guidelines, SVS 1995/25
- Hungary: IAEA 50-P-40, Procedures for Conducting PSA
- USA:
  - NUREG/CR-4550, Vol 1, Methodology Guidelines for NUREG-1150
  - NUREG-1602, The Use of PRA in Risk Informed Applications (draft report for comment)
  - USNCR Standard Review Plan, Use of probabilistic risk assessment in plant-specific, risk informed decision making: (Draft) March 1997
  - Draft regulatory guide DG-1061, An approach for using probabilistic risk assessment in risk informed decisions on plant-specific changes to the current licensing basis, June 1997
  - Draft regulatory guide DG-1062, An approach for plant-specific, risk-informed, decision making: in-service testing, June 1997.
  - Draft regulatory guide DG-1064, An approach for plant-specific, risk-informed, decision making: graded quality assurance, June 1997.
  - Draft regulatory guide DG-1065, An approach for plant-specific, risk-informed, decision making: technical specifications, June 1997.

7.4 National Positions on risk informed decision making.

In the 1997 CNRA SPECIAL ISSUE REPORT "Review Procedures and Criteria for Regulatory Applications of PSA, the position on risk informed decision making in OECD member countries is described. Below, excerpts from this report are presented:

"Although many countries are considering information from plant specific PSAs regularly in the decision process and some are doing this per their requirements, formal regulatory guidelines and review procedures for specific applications (e.g., in-service inspection program) are not in widespread use. Some countries indicate that formal written guidance is desirable; others take a contrary view based on their laws and policies and the nature of the relationship between the licensee and the regulatory authority (e.g., the relationship is sometimes less formal in countries with a small number of plants, such as Switzerland.).

In Spain, the regulatory process is basically deterministic; nevertheless, PSA insights are increasingly being taken into account. PSA arguments are developed and provide an additional input to the decision analysis; but no formal criteria have been established to define the relative weight of such arguments in the decision process. Decisions are made on a case by case basis. The relative weight of probabilistic arguments is case specific, and there have been cases where PSA arguments were used to reject utility proposals that otherwise would be acceptable from the deterministic point of view. On the other hand, only negligible risk increases are accepted by the regulatory authority, or compensating alternate measures are required. In the case of compliance with fire protection regulations, PSA insights have played an important role in determining what plant modifications were really necessary.

In Finland, the use of PSA is required in parallel with deterministic rules in certain situations. This requirement was established after a number of experiences in which it was found that deterministic considerations alone would not provide all the necessary information needed to determine the appropriate regulatory action in response to events at operating reactors. The principal guidance for implementing this requirement is contained in a regulatory guide.

In the Czech Republic, the PSA has been used in conjunction with traditional engineering assessments to improve the quality of the technical specifications at the Dukovany nuclear power station.

In the Netherlands, amendments to the licences of operating reactors will require the licensees to have an operational living PSA. However, the exact content of the living PSA has not yet been defined. Both the licensees and the regulatory authorities are in the process of defining the boundary conditions for the possible applications. The use of PSA for configuration control, optimisation of technical specifications, or event analysis might be objectives to be pursued. It is expected that there will be some reluctance in accepting the final numerical outcomes because of uncertainties, incompleteness, too simplistic models, etc, of PSAs. However, final numerical outcomes will unavoidably play a role in the decision-making process.

In several European countries, PSA and deterministic assessments are integrated within the context of a formal periodic safety review (PSR), which is conducted about every ten years. The deterministic part of the review includes a comparison of the design and the operational state of the plant with safety requirements in place at the time of the assessment.

In the UK, the decision making process is informed by the insights gained from an assessment made against both the deterministic and probabilistic principles. Where these insights are different, there are no formal procedures to resolve this and it has been done on a case-by-case basis. The usual approach has been that: if either of the approaches has identified that improvements could be made to reduce this risk, this would be required unless the licensee could demonstrate to the satisfaction of the regulatory authority that there was an overwhelming case for not doing so. A similar approach is taken in Switzerland.

In the USA, regulatory guides (RG) and standard review plans (SRP) are normally used to articulate NRC staff positions and guide licensees and applicants in meeting the Commission's requirements. The USNRC is currently developing new regulatory guides and inspection guidance, and updating their SRPs to address the application of PSA in developing programs for in-service testing, in-service inspection, graded quality assurance, and in modifying technical specifications.

7.5 Treatment of Uncertainties

Typically, the issues in risk informed decision making are accompanied by substantial uncertainties. This is a strength of PSA that allows to explicitly and tracebly account for the inherent uncertainties in order to arrive at decisions that are robust against uncertainties.

The analysis of the uncertainties should include the following attributes:

- The uncertainties associated with each decision relevant information should be clearly exhibited. Their determination should account for
  - parameter uncertainties, expressed by probability distribution,
  - modelling uncertainties, expressed by one of the several available approaches (see section 3.1.3).

It should be attempted to address the uncertainty resulting from potential incompleteness of the PRA model. This uncertainty is different from the others, as it reflects unanalysed portions of the risk spectrum. Its importance is related to the margin of calculated results to probabilistic targets. If this margin is large for a certain issue, qualitative arguments may suffice for discarding the issue. Otherwise, additional analyses may be required.

7.6 Examples of risk informed decisions in the level 2 domain

7.6.1 Implementation of strategies for fission product retention in a faulted unisolated steam generator at PWR plants

Unmitigated steam generator tube rupture events with failure in the open position of steam generator safety and/or relief valves have been identified in most PSAs for PWRs as potentially significant contributors to releases to the environment. Such releases can be drastically reduced if the fission products escaping through the ruptured tube(s) pass through a column of water in the faulted steam generator. Depending on the height of the water column above the break location, significant portions of the fission products can be retained inside the steam generator and the containment.

The most commonly used strategy for coping with steam generator tube rupture events involves shutting off and isolating auxiliary/emergency feedwater to the faulted steam generator to prevent the introduction of unborated water slugs from the steam generator to the reactor core, if the pressure in the primary systems falls below the secondary side pressure. Consequentially, the steam generator boils dry, opening up a release path to the environment with very little retention capability.

To mitigate the release, severe accident management actions to fill up the faulted steam generator with fire water have been implemented, for example, at Beznau, Borssele, Sizewell-B and the N4 plants.

A different strategy involves not to isolate: auxiliary feedwater to the faulted steam generator, but to control feedwater flow such that the water level in the steam generator is kept at a high level. This strategy is presently being studied for the Swedish PWRs at the Ringhals site.

The reduction of fission product release is due to scrubbing by the water pool and to deposition on the primary side of the steam generator tubes. The range of calculated decontamination factors is from 10 to 100.

The benefit from fission product scrubbing by a water column in the faulted steam generator is demonstrated by PSA results calculated for the Beznau and Ringhals plants. They show substantial reductions of conditional probabilities for exceeding 10% caesium release, given a LRCF mode. In the latest PSA für the Borssele plant, this is not visible because the frequency of SGTR events has been reduced to such an extent that they only insignificantly contribute to releases.

7.6.2 Modifications and backfits to containment systems at PWR plants

In many earlier PSAs, failures of containment systems leading to containment bypass have been identified as major contributors to risk. This finding has led to numerous improvements to containment systems. As consequence of such improvements, the frequencies of large releases due to failure of containment systems are assessed to be in the 10^{-7}/a range in all recent PSAs.

7.6.2.1 Improvements to existing containment systems

- Improved redundancy of containment isolation by the installation of 2 isolation valves for each penetration (some German PWRs)
- Provisions for securing the closing function of isolation valves in the event of loss of emergency power supply (some German PWRs)
- Design of the electrical equipment for accident environment (some German PWRs)
- Sump drainage line is normally closed (Biblis-B)
- Improved redundancy for the actuation of building isolation (Biblis-B)
- Isolation of the ventilation system in the event of increased pressure and activity (Biblis-B)
- Improved systems for measurement of leakage in operation (Ringhals 3/4)
- Provision of TV monitors in the containment (Ringhals 2/3/4)
- Increased capacity of the drainage system Ringhals 3/4)
- Putting in place of procedures for keeping the steam generators filled in interfacing systems LOCA situations (to enhance fission product deposition).
- Enhanced power supply to hydrogen igniters at US PWR plants with ice condenser containments.

7.6.2.2 Improvements by the addition of new systems.

- Provision of hydrogen recombiners/igniters (German PWRs, Borssele, Ringhals PWRs)
- Provision of water injection to the containment sump, serving as water source for
  - Containment spray (preventive AM, Swedish PWRs, Beznau)
  - Filling up of a water pool for quenching the molten core after RPV failure (mitigative AM, Swedish PWRs: external water injection, Beznau: fire water or external water injection, Sizewell-B: fire water). By preventing basemat attack by molten corium, this also eliminates the massive hydrogen production that would result from core melt-concrete interaction.
- Provision of alternate cooling capability for containment fan coolers by using river water (Beznau)
- Installation of filtered containment venting (Swedish and German PWRs, Beznau). For example, the filtered venting system at Swedish PWR plants is designed for a decontamination of 1500 for aerosols and iodine. At Beznau the venting system is also used for hydrogen control by removing hydrogen and oxygen from the containment atmosphere.

For plants with the "zirconium in the core/containment volume" ratio in the high range, the provision of devices for removal of hydrogen from the containment atmosphere are effective for reducing the likelihood of early and late containment failure due to loads from hydrogen combustion. Such devices are also helpful to successful containment venting because they can reduce the likelihood of hydrogen detonations in the venting line.

Filtered containment venting and water injection to the containment sump are very effective for reducing the likelihood of late containment overpressure failure due the build-up of steam and non condensable gases.

Improvements to the high pressure/low pressure interface of the ECCS have greatly reduced the importance of the V-sequence bypass scenario

Modifications and backfits to the systems for containment isolation, including the actuation and surveillance of such systems have practically eliminated the contribution to significant offsite consequences from failure of containment isolation

As SGTR events with unisolated SG are identified as a dominant contributor to massive offsite consequence, attempts to use hydrogen control, containment venting and water injection to the containment for reducing the likelihood of overpressure failure should be accompanied by intense efforts to prevent unisolatable SGTR situations or to mitigate their consequences

7.6.3 Provisions against LOCA outside containment at BWR plants

Unisolated LOCAs outside the containment could result in severe offsite consequences. Therefore, significant efforts have been made at many plants to reduce the likelihood of such events, for example:

- Replacement of actuator motors and gearboxes in the RHR system by a stronger design. (Ringhals 1)
- Replacement of internal isolation valve and actuator in the RHR system by a valve of new design with new actuating equipment capable of higher closing force (Ringhals 1)
- Installation of a leak monitoring system in the emergency core cooling system and auxiliary feedwater system steam lines. (Ringhals 1)
- Installation an additional motor-operated isolation valve in the RHR system (Oskarshamn 1/2, Ringhals 1, Barsebäck 1/2)
- Improved main steam isolation valves (Oskarshamn 3, Forsmark 1/2)
- Improvements to the spring-loaded valve actuators of isolation valves in the RHR system by supporting the closure manoeuvre by auxiliary nitrogen gas pressure (Forsmark 1/2)
- Replacement of the gearboxes for the auxiliary feedwater system's external isolation valves by a more robust design. (Forsmark 1/2)
- Improved calibration of the torque settings during shutdown to assure that the valves can be manoeuvred under maximum design loads. (all Swedish BWRs)

- Installation of an additional shut-down cooling line exiting the RPV at the level of the feedwater line exit nozzles to enable cooldown in the event of steam line break inside the reactor building and failed containment isolation (Gundremmingen)

For Swedish BWRs with external recirculation pumps (all generation 1/2 plants), the frequency of core damage scenarios with unisolated containment is reduced by a factor 10. For other plants, a specific probabilistic quantification of the effectiveness of the modifications and backfits is not available.

7.6.4 Protection of Suction Strainers against Clogging

Clogging of suction strainers is a general concern for all systems taking suction from water sources that are susceptible to the accumulation of impurities large enough to restrict the flow through the strainers. Of particular concern are strainers inside the containment, where they are inaccessible during accident conditions and where physical interactions attending accidents may be responsible for the accumulation of material apt to clog suction strainers:

In the 1983 PSA for the Ringhals-1 plant (and in the Barsebäck 1 PSA), this problem was considered to be important, leading at "first generation" Swedish BWRs to the provision of equipment for backflushing of the strainers, given the pressure drop across the strainers exceeds some predetermined value. It was assumed that substantial clogging of the strainers could occur for large LOCAs with probability 1 and for intermediate LOCAs with probability 0.1, but not for small LOCAs. The event on July 7, 1992 at the Barsebäck 2 plant has qualitatively, but not quantitatively, confirmed that assessment: In the start up phase with pressure at ~30 bar, a leaking valve led to a small LOCA condition. Against the earlier assumptions, sufficient insulation material was washed down to the suppression pool to clog the strainers and make necessary the backflush operation.

In response to the event, numerous modifications were implemented at all Swedish BWR plants, for example:

- Major portions of the mineral wool insulation of the piping have been replaced with metallic insulation material and glass fibre insulation. (Ringhals-1)
- The insulation material on the reactor vessel, a composite of calcium silicate and asbestos (Caposil), was replaced with metallic insulation on parts of the reactor vessel. (Ringhals-1)
- Removal of insulation material from
  - "cold" components
  - many pipes
- Replacement of the strainers in the ECCS and the CVSS with larger strainers. The strainers in the ECCS and the CVSS now have a surface area of approximately 20 m^2 each. Approximately 1 m^2 can be backflushed. (all generation 1,2 Swedish BWRs).
- Installation of new instrumentation for measuring the differential pressure across the strainers. (all generation 1,2,3 Swedish BWRs)
- Implementation of an extensive simulator training program for the control room personnel to perform the backflushing and to get used to the new configuration of controls and indicators (all generation 1,2,3 Swedish BWRs).
- Installation of equipment for backflushing of the strainers in the emergency core cooling system. The system consists of one pressure tank per sub which contains nitrogen gas at a

pressure of 5 bar. The pressure can be relieved to the water filled system, forcing a water slug through the strainers against the normal flow direction (Forsmark 1/2).

- At other Swedish BWRs the CS system can be used for backflushing of the suction strainers.

The modifications and backfits directed at ensuring sufficient coolant flow through the suction strainers in the condensation pool have significantly improved the safety of generation 1,2 and 3 Swedish BWRs. The likelihood of suction strainer clogging is estimated to be reduced by at least a factor 300, which reduces the unavailability of the ECCS function due to clogging of suction strainers to ~ 1/10 of its total unavailability.

7.6.5 Modifications and backfits to containment systems at BWR plants

In many earlier PSAs, failures of containment systems leading to containment bypass have been identified as major contributors to risk. This finding has led to numerous improvements to containment systems. As consequence of such improvements, the frequencies of large releases due to failure of containment systems are assessed to be in the 10^{-7} /a range in all recent PSAs.

7.6.5.1 Improvements to existing containment systems

- Protection against loads due to pipe breaks in the wetwell by
  - reinforcement of blowdown pipes, penetrations and cables in the wetwell
  - installation of missile protection for equipment in the wetwell (Ringhals-1)
- new level measurement in the condensation pool:
  - level measurement in the wetwell
  - level measurement for the entire containment from the outside of the drywall wall
  - differential pressure measurement across the containment spray system strainers
  - differential pressure measurement across the emergency core cooling system strainers (Ringhals-1)
- Installation of new instrumentation for measuring temperature, pressure, water level and activity in the containment (Forsmark (1/2)
- Enhanced power supply to hydrogen igniters at US BWR plants with Mark III containments

7.6.5.2 Improvements by the addition of new system

- Provisions for using the fresh water reservoir for flooding of the containment in a severe accident. (Ringhals 1)
- Containment pressure relief system

  Use of the high capacity containment pressure relief system without filtering for alternate heat removal in sequences that include failure of the pressure suppression by system, but operability of the normal ECCS systems. The pressure relief system is provided with a rupture disk, and backup actuation from the control room. As the request for pressure relief will occur when the core is not yet damaged, filtering is not required (Ringhals-1, Forsmark

1/2/3, Oskarshamn 1/2/3 At the 2 units of the Barsebäck plant, the relief line is connected to the FILTRA system)

- Primary Containment Venting (PCV)

  PCV is used to remove the residual heat to the atmosphere and to prevent the primary containment pressure limit from being exceeded when the CS and SPC systems have failed to perform successfully. As the request for PCV will occur when the core is not yet damaged, filtering is not required. At the same time, the suction source for injection to the core can be switched from the suppression pool to alternate water sources to avoid damage to pumps due to high temperature. In the US all utilities with Mark I containments committed to install a hardened wetwell vent system, if one was not already in place. The assessment of the benefit gained from a hardened vent capability varies between a few percent to a factor 14 reduction of core damage frequency among licensees. Among plants with Mark II containments, LaSalle and Nine Mile Point 2 have hardened vents are in place, while Limerick and WNP 2 use the existing, not-hardened vents. The Limerick IPE submittal acknowledges that use of the existing vent system will lead to a duct work failure and therefore should only be used when there is adequate core cooling. The WNP 2 analysis indicates very limited benefit from a hardened vent path.

  None of licensees with Mark III containments find that a hardened vent would have a significant impact on their CDF or containment results. One plant, Perry, evaluates the effect of a passive vent design featuring a rupture disk and an alternate vent line which would open automatically upon containment overpressure. A substantial decrease in the probability of RPV failure and containment failure is observed. This alternate vent path has not been designed, however.

- Containment Water Injection System (CWIS).

- Provision of diverse external water supply for safety systems and accident mitigation by the CWIS. It can be used to provide additional suction sources for

  - high pressure auxiliary feedwater (Ringhals 1)
  - addition of water via the containment spray system (all Swedish BWRs, Mühleberg)
  - accident mitigation by ultimate flooding of the containment to the upper core level, ensuring stable terminal cooling of core material (all Swedish BWRs, Mühleberg)

- Containment Spraying and Flooding

  Modification to the interconnection of the containment spray system and the fire protection water supply system to use fire water for flooding of the containment after an accident (Forsmark 1/2, Oskarshamn 1/2/3, Mühleberg). In the US, most of the IPE analyses for plants with Mark I containments discuss external sources for drywell spray or vessel injection. In many cases the submittal states that alternate water sources exist at the plant and have been credited in the IPE. Usually the external water source is provided via the service water system using river or pond water or the fire protection system. Among the plants with Mark II containments, a connection to the fire protection system for the drywell sprays exists at Limerick. Nine Mile Point 2 has implemented a raw water cross-tie as an alternate injection source to the RPV or the containment spray. A fire water connection is under consideration at LaSalle. At plants with Mark III containments, fire water is the principal alternate water source. The Clinton IPE notes that such a connection is under consideration. The Grand Gulf IPE finds that a fire water cross-tie to vessel injection has a significant impact on CDF. The River Bend IPE also states that a cross-tie of the fire protection water to the RPV injection has been made subsequent to the submittal of the IPE analysis.

- Provision for flooding of the lower drywall with water in the event of a vessel meltthrough by connecting the condensation pool and the lower drywell (Forsmark 1/2).
- Provision for flooding of the lower drywall with water in the event of a vessel meltthrough by using additional water sources for drywell spray or through RPV injection systems (US BWRs, compare the paragraph on "containment spraying and flooding")
- Provisions for hooking up the mobile fire pumps to the fresh water reservoir in the service building for flooding of the containment when the fire protection system cannot be used (Ringhals 1).
- Water pool underneath the core without supply from external source. To avoid concrete-core melt reaction which could ultimately lead to the penetration of the containment barrier, a sufficiently deep water pool under the RPV should is made available in the event of RPV failure:
  - At Ringhals-1, Oskarshamn 1/2 and Barsebäck 1/2, core melt would drop into water in the cylindrical suppression pool underneath the RPV, where the melt is assumed to be quenched
  - At all Forsmark plants and at Oskarshamn 3 with annular suppression pools, provisions are made to flood, by automatic opening of valves, the lower drywell with suppression pool water on indications of core melt accidents. Special provisions have been made to protect containment penetrations in the pedestal area
- To avoid breach of containment integrity in the event of a large LOCA with the beamwork between the drywell and wetwell not intact, a rupture disc and two stop valves have been installed for passive unfiltered venting to the atmosphere . The stop valve closes 10 minutes after containment isolation has been activated (Forsmark 1/2)
- Filtered Containment Venting

  To avoid a breach of containment integrity due to a slow pressure increase following an accident, systems for filtered containment venting have been installed. (FILTRA at the Barsebäck plant at which the system is also used for alternate heat removal, Multi Venturi Scrubber Systems (MVSS) at all other Swedish BWRs). At all German BWRs the filtered containment venting system is used for alternate heat removal (preventive AM) and for overpressure protection of the containment (mitigative AM)

  For example, the filtered venting system at Swedish BWR plants is designed for a decontamination of 1000 for aerosols and iodine.

  In combination with external water injection the filtered containment venting system can also be used for ultimate heat removal if all other systems have failed.
- Inertisation of the wetwell (Gundremmingen)
- Inertisation of the primary containment (German BWRs, except Gundremmingen)

The provision of large quantities of water for flooding the drywell and the pedestal area in case of impending RPV meltthrough is very effective for preventing attack by molten corium of containment structures. By combining containment flooding with filtered containment venting, the likelihood of significant releases to the environment can be substantially reduced. This is demonstrated by the results for Mühleberg, Barsebäck and, most notably, Forsmark 3.

7.7 Conclusions

The listed examples demonstrate substantial use of PSA in safety relevant decisions by regulators and licensees. With the application of PSA becoming more formal through the introduction of the concept of risk informed regulation, a further increase is to be expected to the extent supported by state of the art of PSA technology.

For operating plants, risk relevant information from level 2 PSAs is mainly used for the identification and implementation of accident management measures. The aim is to develop robust SAM guidelines, addressing the vulnerabilities and uncertainties identified in the PSA.

PSAs performed in this context should avoid any unnecessary conservatism and be as realistic as possible. In parallel, the inclusion of more realistic boundary conditions and assumptions should take place in deterministic analyses in order to make the complementary interaction of probabilistic and deterministic methods more effective and beneficial.

APPENDIX A: SEVERE ACCIDENT COMPUTER CODES

A.1 Fully Integrated Plant Simulation Codes

A.1.1 Modular Accident Analysis Program (MAAP)

MAAP 3.0B

General

The EPRI Modular Accident Analysis Program (MAAP 3.0B) [3.5], developed by Fauske and Associates, Inc., as a PSA tool is a fully integrated code that couples thermal-hydraulics with fission product release and transport. It has been used for many PSAs, especially for most of the U. S. Individual Plant Examinations (IPEs). It simulates the accident progression from a set of initiating events to either a safe, stable, and coolable state, or containment structural failure and radioactive release to the environment.

Accidents analysed include a variety of transients, including bypass, mid-loop operation and shutdown sequences.

The design intent for this code for PSA application results in major differences in modelling assumptions, when compared to the separate phenomena codes. An example is the simplifications introduced in the MAAP momentum equation that neglects the acceleration terms which are considered in SCDAP/RELAP5. The code has been subjected to independent design review and it has also been reviewed by the USNRC. MAAP 3.0B has been compared with other codes on some aspects of severe accident phenomena (e.g., core melt progression, source term estimates for plant applications using MELCOR [A.1]).

Over a period of time, the code has also been modified to be used as a tool to evaluate accident management actions. Separate versions for PWRs and BWRs are available.

Thermal-hydraulic Modelling

MAAP uses a control volume and flow path approach in which the geometry of the control volumes (called regions) is pre-specified and different for a PWR and a BWR. MAAP3.0B BWR has 8 control volumes for primary system gas flow and the PWR version has 14 plus the pressurizer and the quench tank. The reactor building has an arbitrary user-defined nodalisation. The primary system is divided into the regions: upper and lower plenum, reactor core, downcomer, and for PWRs, (un-)broken cold and hot legs, and (un-) broken steam generator loops. Separate mass and energy conservation equations are solved for each of the regions.

The PWR containment is divided into the following regions: upper and lower compartment, cavity, annular compartment, pressurizer relief tank, pressurizer, possibly two extra compartments for an ice-condenser, and primary system. The BWR containment is divided into two regions: reactor pedestal cavity, drywell, wetwell, possibly an upper and a lower containment compartment, and primary system. Flows consist of steam, water, hydrogen and other non-condensible gases, and corium. Flow paths can be e.g., pipes, surge lines, penetrations, and relief valves. Separate mass and energy conservation equations are solved for each region. The equations are lumped parameter, non-linear, first-order, coupled, ordinary differential equations.

Core Geometry and Core Melt Modelling

The core is divided into concentric radial rings (up to 7) and axial segments (up to 10). MAAP uses a single core relocation model. Features are included in the code such that limited sensitivity studies can be performed on the core melt behaviour and hydrogen generation. MAAP assumes a decrease in steam supply, and hence in hydrogen generation, due to channel blockage in the relocated core.

Other Physical Processes

MAAP3B has a model for flammability which depends upon composition and temperature, a model for combustion completeness in case of incomplete combustion, and a model for burn time. Flame propagation between compartments is also treated. MAAP also considers “jet-burning” (i.e., ignition of a hot jet containing flammable gases that enter a compartment with oxygen available; MAAP also considers auto-ignition of gases at high temperature, which leads to recombination in some cases.

MAAP also uses the SUPRA model for calculation of retention in water pools.

Radionuclide behaviour

MAAP models the transport and retention of fission products. The materials released from the core are divided into 12 fission product groups, divided according to chemical characteristics. The fission product states modelled are: vapour, aerosol, deposited and contained in core or corium. There is no separate model for retention/deposition and agglomeration. Succinctly stated, the MAAP aerosol model considers the combined effects of agglomeration and removal mechanisms including gravitational sedimentation, condensation removal, inter-compartmental transport, thermophoresis, and impaction. Revaporization is included as transfer between the states. The MAAP aerosol model is a correlation of exact solutions to the polydisperse integro-differential aerosol equation, and it is extensively validated.

MAAP 4

Recently, MAAP 4 [3.17] has been released. Apart from general modelling enhancement, it is designed to evaluate potential accident management actions and also for application to ALWRs (with passive designs), and VVERs. Three major differences compared with MAAP 3.0B are:

(1) *Core melt progression model and natural-circulation induced creep rupture failure of the RCS:* Prior to core uncovery, the RCS response is not substantially different from that calculated by the MAAP 3.0B code. Once the core is uncovered and overheated sufficiently to result in rapid oxidation of the Zircalloy cladding, the first major difference is apparent. In MAAP 4 when the melting point of the control rod material is calculated, it can relocate away from the fuel. In

addition, the MAAP 4 models include the process of dissolving the uranium dioxide fuel with molten zirconium and the relocation of the lower melting point material. This is substantially different than the lumped fuel behaviour in the MAAP 3.0B code. As the core melt progression continues, the potential for natural circulation flows, particularly for the open lattice PWR core design, is evaluated along with the potential for creep rupture of the hot leg piping, the pressurizer surge line and the steam generator tubes.

(2) *Modelling of reflood process, external vessel cooling, and vessel creep rupture:* If the accident sequence being considered results in reflooding of the reactor core once core degradation has occurred, the MAAP 4 models address this reflooding process and the potential for quenching the core debris, both within the original core boundaries and in the reactor pressure vessel lower plenum. The first of these is different from the MAAP 3.0B code, while the second is a set of phenomena which could be represented in the MAAP 3.0B codes. However, if water is available on the exterior of the RPV, the influence of external cooling in removing energy from the vessel wall and in preventing the potential creep rupture of the vessel due to core debris thermal attack on the vessel lower head is modelled. Both external cooling and creep rupture of the vessel are phenomena not included in previous MAAP versions.

(3) *Containment model:* For the containment analyses, the containment model has been enhanced to provide a generalised description of the containment such that the nodilisation can be specified by the user. In addition, the containment model considers counter-current flows and plume behaviour which are influential in containment mixing and fission product transport. The containment model for many of the advanced plants has been set up to include those features typical of the ALWR designs.

The additional models in MAAP 4 include the RPV failure model, the molten debris heat transfer model, a jet entrainment model for the debris fragmentation in the RPV lower plenum, an optional debris dispersal model, a two-dimensional core-concrete interaction model, the RPV external cooling model, a new model for hydrogen combustion, and the in-vessel debris cooling model. MAAP4 also calculates the pool pH which is useful for long-term iodine behaviour.

A.1.2 MELCOR

General

The MELCOR code [3.6] developed by Sandia National Laboratories (SNL) under the sponsorship of the United States Nuclear Regulatory Commission (NRC), is a fully integrated, full plant severe accident simulation code. MELCOR is designed to be relatively fast-running with a flexibility to model a large spectrum of severe accident progression phenomena. It includes many modelling features and concepts of the other NRC codes, such as for example SCDAP/RELAP5 and CONTAIN.

The use of parametric models is, in general, limited to areas with great uncertainties where there is no consensus concerning an acceptable mechanistic approach.

MELCOR is intended to be applied by the NRC for:

(1) PSA studies for existing and advanced LWRs.

(2) Best-estimate accident sequence studies to develop insights into (a) physical phenomena, and (b) hardware performance.

(3) Audit reviews of PSAs.

(4) Accident management studies that analyse the progression of accidents and evaluate the detrimental and beneficial effects of various strategies.

The code is based on specially developed models for thermal hydraulics, core melt, fission product release and transport processes. In several instances, a number of existing codes have been directly integrated into MELCOR architecture, these include CORSOR/CORSOR-M/CORSOR-BOOTH [A.2], VANESA [A.3], CORCON/MOD3 [3.12], MAEROS [A.4], TRAP-MELT2 [A.5], and SPARC-90 physics [A.6].

The MELCOR peer review [A.7] has identified many code limitations that are being addressed by USNRC-sponsored programs at SNL. Other peer review concerns associated with model sensitivities, time-step dependencies, and experimental benchmarking studies are being addressed. Reference [3.21] list some examples of recently published studies, based on the NRC-sponsored MELCOR Code Assessment Program (MCAP) activities.

A summary description of various MELCOR models follows.

Thermal-hydraulic Modelling

In MELCOR, the thermal-hydraulic processes are modelled by the Control Volume Hydrodynamics (CVH) and Flow Path (FL) packages, while the thermodynamic calculations are performed within the Control Volume Thermodynamics (CVT) package. The CVH/FL packages are based on a highly versatile architecture and a general control volume hydrodynamic network concept which provide thermal-hydraulic boundary conditions to other MELCOR phenomenological packages.

A general "volume/altitude" and "virtual volume" approach is employed to define, through user input, the control volume geometry. Hence, component and subsystem models must be built-in through user input. This provides a valuable tool for application of MELCOR to variety of nuclear reactor designs.

Control volumes are interconnected via "flow paths" through which hydrodynamic material may pass without any residence time (assumption of negligible volume). Flow path area can be modified by input to model valves, obstructions, etc. The material and energy contents of both coolant and non-condensible gases are assumed to reside within control volumes. Mass and energy sources and sinks are treated as boundary conditions to CVH/FL. This includes decay heat; heat from structures; water from condensation and evaporation on structures; non-condensible gas sources from core-concrete-interactions (CAV), oxidation, and other sources.

In CVH/FL, hydrodynamic materials are assumed to separate by gravity into a lower pool region (which may contain steam bubbles, but not non-condensible gases), and an overlying atmosphere (which may contain liquid droplets, gases, vapour). The pool and atmosphere velocities and directions may be different. The mass exchange models include options for (1) a thermal and mechanical equilibrium model which assumes the same pressure and temperature for both pool and atmosphere, and (2) a thermal non-equilibrium model which assumes the same pressure, but different temperatures for pool and atmosphere (vapour superheat and liquid subcooling).

The basic hydrodynamics methods embodied within CVH/FL package are closer in principle to those of RELAP4 than TRAC or RELAP5 computer codes. The CVH/FL models are an adaptation of the approach employed by HECTR [A.8] and CONTAIN [3.12] containment analysis codes.

Core Geometry and Core Melt Modelling

The core and the lower plenum in MELCOR are divided into a number of user specified concentric radial rings and axial segments. A number of component types and materials are modelled. Heat transfer during heat up is modelled using correlations published in the literature for conduction, convection, radiation and oxidation. A model for the COR package calculates the downward flow and subsequent resolidification of the materials.

Oxidation and heat transfer by radiation, conduction, and convection are calculated separately for each component. A simple, candling model treats the downward flow and refreezing of molten core materials, thereby forming layers of solidified debris on lower cell components, which may lead to flow blockages and molten pools.

Failure of the core structures such as the core plate, as well as lower head heat up and failure followed by debris ejection, are treated by simple parametric models.

Other Physical Processes

Besides the processes already mentioned, MELCOR includes models for: the forming of non-condensible gases, combustion of gases (using the HECTR models [A.8]), the thermal-hydraulic part of core-concrete interactions (using the CORCON/MOD3), and direct containment heating (using a parametric model).

The interaction of the debris released from the vessel with the concrete basemat in the cavity is modelled using the CORCON/MOD3 code. The molten debris may contain large amounts of un-oxidised metals such as zirconium and chromium as well as oxides such as ZrO_2 and UO_2. Provisions are included that allow for various configurations of the metallic and oxidic constituents of the ejecting debris. These include, instantaneous stratification of layers, mixing of layers, etc. CORCON/MOD3 calculates the rate of erosion in the concrete basemat, the temperature and composition of the molten layers, the temperature, flow rate and composition of gases such as CO_2, CO, H_2, and water vapour evolving from the concrete. Heat is exchanged between (1) the melt and the concrete, (2) layers of the melt and (3) the top surface of the melt and the atmosphere, water (if any) and the structures above it. The melt concrete heat transfer includes options for a gas film model and an intermittent film model. The gas film model assumes the occurrence of Taylor instability bubbling on the pool bottom as long as the pool bottom remains horizontal, and the existence of flowing gas film along the steeper portions on the side of the pool. Inter-layer heat transfer in the presence of gas bubbling (due to gases produced by ablation) are modelled. If a coolant layer is present over the oxide layer, boiling heat transfer to the overlying coolant layer is also modelled. The actual boiloff is treated in the CVH/FL packages. The concrete ablation products (e.g., steam and CO_2) are modelled to react with the un-oxidised metals present in the melt. The heat generation in the molten pool is both due to decay heat and the heats of reactions. As mentioned, the CORCON/MOD3 also includes improved models for heat transfer processes at melt concrete interface, improved phase diagrams, and a full integration of CORCON and VANESA codes, and several other improvements.

The major phenomenological calculations by VANESA include:

(1) CO_2 and H_2O which arise from the concrete attack, react with the major constituents of the metallic layer (Fe, Cr, Ni and Zr). An equilibrium analysis is performed and the resulting oxides are transferred to the oxidic layer. This determines the oxidic composition of the melt.

(2) The steam, CO_2, CO and H_2 are assumed to rise as bubbles, with bubble velocity calculated using standard equations.

(3) Vaporisation of fission products and other melt constituents from the melt into gas bubbles. Many of these releases involve chemical reactions. For instance, La_2O_3 vaporises from the molten core debris primarily as LaO. These chemical reactions are modelled as equilibrium chemical reactions, characterised by a thermodynamic equilibrium constant. Vapour species will exist as bubbles in the approximate proportion of their vapour pressures. When the gas bubbles burst through the melt surface, many of the vapour species will immediately condense in the cooler environment forming aerosol particles. The distribution of aerosol particles is given by an empirical size (mean diameter) equation that is a function of concentration and density. In addition to the condensing vapours, the mechanical action of the bursting bubbles is also predicted to produce aerosols.

The CORCON/MOD3(VANESA submodel) also includes an option for treatment of non-ideal chemistry, which is not yet fully tested and operational.
The experimental bases for CORCON/MOD3 is consistent with the available data base as documented in References [A.9 and A.10].

Radionuclide Behaviour

The release of aerosols and vapours from the core materials is treated by the CORSOR correlations with a dynamic surface-to-volume multiplier. Releases from core-concrete interactions are treated by the VANESA submodels of CORCON/MOD3. Aerosol agglomeration and deposition are calculated by MAEROS models. Transport of aerosols and vapours between control volumes occurs with the bulk fluids, gases or water, with zero slip, and aerosols can be removed as they pass through water pools, based on models from the SPARC code. User-specified chemical reactions can be treated, which should be based on the results of more detailed codes or on experiments.

During the heat up phase of the accident, additional fission products are released by vaporisation or other thermally activated process. In addition, materials from structural cladding and control rods heat up, vaporise and leave the core. The release of fission products from the fuel is modelled using either CORSOR , CORSOR-M or CORSOR-BOOTH representations of radiological release data for irradiated fuels.

The only difference between CORSOR and CORSOR-M is in their functional fit to experimental data. In CORSOR, the release rate coefficient is formulated as follows:

K [fraction/min] = A Exp{BT}

where A and B are empirical group-dependent coefficients, and T is the temperature. On the other hand, the CORSOR-M uses an Arrhenius representation (of the same experimental data) of the form,

K [fraction/min] = K_o Exp{-Q/RT}

Here K_o and Q are empirical group-dependent coefficients, R is universal gas constant, and T is temperature. With the exception of Te releases which is dependent on Zr oxidation fraction, the rates of release for the remaining fission product groups are dependent, only on fuel temperature.

Refinements in CORSOR release model (i.e., CORSOR-BOOTH) have been formulated recently by Ramamurthi and Kulman [A.2] in light of the substantial database that currently exist of fission product release observations in experimental fuel rod tests conducted over the past decade. The new and refined release model is expected to be incorporated into the MELCOR architecture in the near future.

The transport behaviour of fission product vapours and aerosols in RCS and containment is modelled by adaptation of the MAEROS [A.4] aerosol physics into the RN package. Given an aerosol source rate as calculated from CORSOR/CORSOR-M and/or VANESA, RN aerosol transport calculations are performed to determine: (1) the suspended mass concentration as a function of time, (2) the size distribution of airborne particles as a function of time (mass concentration of water and particles in each size class), (3) the cumulative settled out quantity, (4) the cumulative plated out quantity and (5) the cumulative leaked out masses. The phenomena treated include: (1) agglomeration (random movement, gravity, turbulence), (2) removal (random movement, gravity, movement in a condensing steam, thermophoresis and sprays), (3) steam condensation onto aerosols, and (4) homogenous nucleation of water droplets. The modelling of aerosol transport in containment is supplemented by calculation of aerosol behaviour as it passes through suppression pools, using the SPARC physics.

A.1.3 ESCADRE

General

ESCADRE is a set of computer codes dedicated to the analysis of water reactor severe accidents. Each aspect of accident phenomenology is covered by specialised codes : core degradation, thermalhydraulics, direct containment heating, corium-concrete interaction, aerosols and fission products transport in the primary circuit and the containment, iodine behaviour and simulation of safety systems. The core inventory is provided by the code PEPIN.

The main codes composing ESCADRE were developed in the 80’s as stand-alone modules. They have been assembled in the code system ESCADRE Mod 0 in March 1992. More recently, new developments have been undertaken in order to couple the ESCADRE codes in a flexible way through a database and resulted in the release of ESCADRE mod 1.0 in November 1995. Now, the effort is focused on the development of a tool for the French level 2 PSA project and will resulted in a more complete ESCADRE mod 1.1 version, expected in June 1996. In this version, the different codes will be coupled.

In the future, ESCADRE will constitute the basis for the development of the common German-French integral code ASTEC, that will include the German RALOC and FIPLOC modules as regards the thermalhydraulics and aerosols transport in the containment. The first ASTEC V0 version is expected at the end of 1996.

ESCADRE is an integral code: it includes simplified models but cover almost the whole phenomenological domains. It is designed to be a fast running code. In parallel, detailed mechanistic codes, able to generate input to ESCADRE database, are developed in some physical domains: CATHARE for the primary circuit thermalhydraulics, ICARE for the core degradation, MC3D for the steam explosion, CASTEM for the primary circuit and containment mechanical behaviour.

The validation of ESCADRE modules is performed either against detailed mechanistic codes, or directly against experiments in the physical domains where mechanistic codes do not exist.

Core degradation

The VULCAIN code is devoted to the calculation of the core thermal behaviour, the thermalhydraulics of the primary circuit and the release of fission products and control rod materials. VULCAIN was designed as a reactor code in the sense that the topology of the circuit (set of control volumes) is fixed and corresponds to 900 or 1300 MW French PWRs. The scope of VULCAIN is limited to the simulation of the core uncovering phase : the initial phase of the accident must be calculated by CATHARE. VULCAIN includes a description of the different physical phenomena occurring in the vessel: clad deformation and rupture, clad oxydation, UO2-Zr interaction, radiative heat transfer, liquid material relocation, core slumping and core support plate rupture. The next version of the code will include a corium pool model, the modelisation of the natural convection between the core and the upper plenum and a simplified model of vessel lower head rupture.

Fission product transport in primary circuit

The SOPHAEROS module calculates the behaviour of an aerosols population injected into a circuit. The physical models are those of the AEROSOLS code. SOPHAEROS also deals with the vapour fission product transport and deposition for volatile species (I2, Cs, CsI, CsOH, Te) in the circuit. The main mechanisms accounted for are vapour sorption/desorption and vapour condensation/evaporation on aerosols and wall surfaces.

Corium-concrete interaction

WECHSL/CALTHER is devoted to the calculation of corium interaction with the concrete of the cavity. The main aspects dealt with are the calculation of corium mass and energy balances, of concrete erosion and subsequent gas release. The aerosol release during corium-concrete interaction is not modelled in WECHSL.

Thermalhydraulics in containment

The JERICHO code is devoted to the calculation of thermalhydraulics in the containment. It is a one-compartment model with two phases in thermal desequilibrium. It is a comprehensive reactor code whose capability includes the simulation of safety systems, such as spray and filtered venting, the calculation of H2 and CO burning and of fission product behaviour as a heat source, partitioned between air, sump water and walls.

Fission product transport in containment

AEROSOLS-B2 handles the calculation of aerosols transport and deposition in the containment. The mechanisms taken into account are : settling, deposition on surfaces (thermophoresis, diffusiophoresis, brownian diffusion), coagulation (gravitational, brownian, turbulent) and aerosol growth by steam condensation (soluble and non-soluble species).
IODE aims at describing iodine physics and chemistry in the gaseous and aqueous systems of the containment. Elemental iodine can be found in different inorganic (I^-, I_2, IO_3^-, HOI) or organic (CH_3I) forms. Chemical reactions (with radiolysis effects), mass transfers and interaction with wall surfaces are modelled.

A.1.4 THALES/ART and THALES-2

The THALES/ART code system [A.15-A.17] was developed at the Japan Atomic Energy Research Institute and was composed of THALES for thermal hydraulic analysis and ART [A.18] for fission product (FP) transport analysis. These codes were combined into THALES-2 to make a fast running, fully integrated severe accident analysis code. These codes were used in JAERI for accident progression analysis, sensitivity analyses on source terms [A.19], a level 2 PSA [A.20], and examination of the effectiveness of accident management measures [A.21].

The thermal hydraulics model of THALES and THALES-2 uses a control volume and flow path approach. The geometry of the control volumes are pre-specified and are different for BWR and PWR. In each volume, a mixture level is considered which separates the volume into a gas region and a liquid region with void. For junctions a counter-current flow model can be applied. The containment can be divided into several volumes as a user option.

Models are provided for metal/water reaction, molten fuel relocation, debris relocation to selected containment volumes at the reactor vessel failure, hydrogen burning, core concrete interaction at each locations to which debris dispersed. Actuation logics of various plant systems and operator actions can be simulated. Considering the different configuration of BWRs and PWRs, two versions THALES-P2 and B2 have been developed. In THALES, the model for core melt progression assumes that, when the temperature of a fuel node reaches a user-specified melting temperature, the material in the node relocates at once according to the relocation model chosen by the user. In THALES-2, this model has been replaced with a relocation model with user-specified film velocity.

FP transport calculation in THALES-2 is made by the ART module [A.17] which has a set of mechanistic aerosol model described in the section for ART-mod2. Thermal-hydraulic conditions in each volume are directly used for FP transport calculations in the particular volume and the decay heat of FPs is transferred as a heat source to the gas, liquid or structures depending on the form of FPs.

Computational speed were increased by several special features: the use of different time step sizes for thermal-hydraulics and FP transport calculations, the use of relatively large control volumes, the integrated form of loop flow calculation in the PWR version, and an improved numerical method in the ART module. The CPU time requirement depends on the accident sequence and is about 1/10 - 1/20 of real time for BWR and 1/4 - 1/10 for PWR on the main frame computer at JAERI, FACOM-M780.

A.1.5 ART Mod2

ART Mod2 [A.22, A.23] has been developed at JAERI for the analysis of radionuclide behaviour in primary system and in containment under severe accident conditions. The code considers removal of radionuclides by natural deposition and by the engineered safety features (ESF) such as spray systems. The natural deposition mechanisms considered by the code are gravitational settling, thermophoresis, diffusiophoresis, Brownian diffusion, diffusion under laminar or turbulent flows and resuspension for aerosol, and condensation, adsorption and revapourisation for gaseous radionuclide. Aerosol growth by agglomeration and condensation/evaporation of volatile material at the aerosol surface are considered. The code also models the iodine chemistry in water such as radiolysis or hydrolysis. The code is capable of treating up to 60 materials including chemical compounds and of representing the systems by an arbitrary number of volumes. In each volume, materials can take the forms of gas, aerosol, deposition onto structure, and solution in water.

The code solves the governing equations for multi-component aerosol and gaseous radionuclides. The "sectional method" adopted by TRAPMELT-2 is used to describe the size distribution of aerosols. The phase change of chemical species can be considered while the chemical reactions among FP elements in gas phase is not taken into account. As a special feature, the code has a fast running capability for the use of Probabilistic Safety Assessment (PSA). Since ART Mod 2 is a module of the JAERI's integrated severe accident analysis code, THALES-2 [A.24], the improvement of ART Mod2 can be directly reflected on the source term analysis.

A.1.6 REMOVAL

The REMOVAL code has been developed at JAERI since 1983 to analyse fission products behaviour in the containment during a severe accident. The code analyses the behaviour of multi-component aerosol particles, inorganic iodine, organic iodides in the multi-compartmented containment. Mass concentration and size distribution of the airborne particles, cumulative deposited mass, leaked mass from the containment are calculated for aerosol as a function of time. Deposition of particles due to gravitational settling, Brownian diffusion, diffusiophoresis and thermophoresis is modelled in the code. Models of the Brownian, the gravitational and the turbulent agglomerations are included in the code [A.25]. Particle size change due to steam condensation/evaporation and hygroscopicity is also taken into account. Particles transport between compartments by the gas flow and the gravity are modelled in the code [A.26]. Concentrations of iodine in gaseous phase and aqueous phase such as water pool and water film are calculated by assuming equilibrium condition between the two phases and the absorption by the paint of the containment wall. Engineered safety features such as containment spray and filters are also modelled to analyse their effects on the behaviour of the gaseous and particulate FPs. For the calculation of concentration of noble gases and organic iodides, leakage from the containment is solely taken into account.

A.1.7 JASMINE

JASMINE (JAERI Simulator for Multiphase Interaction and Explosion) is a multi-dimensional multi-field steam explosion simulation code [A.27,A.28]. It consists of the premixng and propagation modules and they are presently being developed separately.

Premixing module is based on MISTRAL code originally developed for multiphase flow simulation by Fuji Research Institute Co. The conservation law of mass, momentum and energy for three fields, i.e. molten fuel, coolant liquid and coolant vapour, are employed as the fundamental equations. Constitutive models which describe the exchange terms between phases as well as the surface area transport equation for the melt phase are incorporated to close the problem. Numerical solution of the system is given by finite difference method with temporally semi-implicit scheme. TVD scheme is applied for spatial discretisation of mass and energy equations to improve the resolution of phase distribution.

Propagation module is under development. Basic frame is similar to the premixing module, but the numerical solution is designed more explicitly and phase handling is modified considering the difference of the time scale and controlling physics in the phenomena.

A.1.8 MACRES

The MACRES code, developed by the Nuclear Power Engineering Corporation (NUPEC) of Japan, analyses the time-dependent behaviour of fission products(FPs) and aerosol within the primary coolant system and the containment during a postulated LWR severe accident. This code includes models for fundamental phenomena governing aerosol formation by homogeneous and heterogeneous nucleation and prediction of chemical speciation by free energy calculations. Models for aerosol and FP transport such as aerosol agglomeration and deposition, vapour condensation and adsorption on structure surfaces and removal by engineered safety features are included in the code.

A.1.9 MAPLE (A code for analysis of DCH)

The computer code MAPLE (Multi-compartment model for Analyses of Pressure Load in the Event of direct containment heating) [A.30], developed by JAERI, simulates the thermal-hydraulic process of Direct containment heating(DCH) using a lumped-parameter dynamic model taking into account the heat and mass transfer between the debris particles suspended in the containment, the atmosphere, the suppression pool, and the structures as well as the heat generation by metal/water reaction. The user has to supply the following parameters as input: size of debris particles dispersed in the atmosphere, fraction of debris dispersed into the atmosphere, rate of debris ejection from the pressure vessel, temperature, and components of debris ejected from the pressure vessel. A unique feature of this code is that it has a multi-compartment framework to take into account the effect of suppression pool in a BWR containment.

A.2 Separate Phenomena Codes

The CORCON/MOD3 and MAEROS codes are stand-alone versions of the modules which are included in the MELCOR code. Other codes, such as the RELAP5, RAMONA, (CATHARE), and TRAC codes only contain thermal-hydraulic models, therefore they are not discussed here, since they have limited application to severe accidents.

A.2.1 SCDAP/RELAP5 Code

The SCDAP/RELAP5 code is designed to provide the overall RCS thermal-hydraulic response, core damage progression, and fission product release and transport during severe accidents [3.7]. The code is made up of three separate computer codes: SCDAP, RELAP5/MOD3, and TRAP-MELT.

The SCDAP code models melt progression, core heat up, debris formation, heat up and melting. The RELAP5 module models the RCS thermal-hydraulics, reactor kinetics, and the transport of steam and non-condensible gases. Fuel rod pressure build-up is calculated using correlations based on the FRAPCON-2 code. The PARAGRASS code is used to calculate fission product releases for volatile species, while, CORSOR-type correlations are used for more refractory aerosols.

The TRAP-MELT code calculates the transport and deposition of fission product vapours and aerosols on the RCS structural surfaces. This includes models for formation, growth and deposition of aerosols, and the revolatilisation of previously deposited species from structural surfaces. All these processes are fully coupled. TRAP-MELT handles ten species of materials including noble gases. The noble gases are not retained in the RCS and are included only for calculation of the decay heat. The fission products are treated as vapour when they leave the core. They can condense onto aerosols and walls, evaporate from walls and can become chemisorbed onto walls. The rest of the less volatile fission products along with the constituents of Zircalloy, stainless steel and control rods are assumed to be in condensed form and are treated as aerosols. These aerosols can agglomerate and deposit on surfaces, but they cannot evaporate. Random (Brownian) motion, gravity and turbulence are treated as the processes that cause agglomeration. Brownian motion, gravity, turbulence and movement in a temperature gradient (thermophoresis) are treated as the forces that cause wall deposition. The TRAP-MELT deposition mechanism is based on similarity of heat and mass transfer, and uses a parametric deposition velocity. A parametric treatment of chemical reactions is also included.

Other features of SCDAP/RELAP5 code include models for (a) the formation of debris beds either through cohesive mass of molten or solidified material, or through the accumulation of loose debris, (b) heat up and melting of debris beds, (c) formation of a molten pool supported by a crust of frozen material, (d) crust failure, (e) potential fragmentation of a molten jet, (f) interaction of debris with the vessel wall, and (g) failure of the vessel wall. In addition, BWR specific control blade models are also included to model the special features of B_4C. Models for thermal and mechanical failure of RCS structures exist for prediction of potential creep rupture failures within the RCS pressure boundary. A two fluid, non-equilibrium, six equation model for hydrodynamics and heat transfer and cross wall junction modelling for treatment of multi-dimensional effects are all included.

Models are being developed at Idaho National Engineering Laboratory (INEL) to track axial and radial movement of melt in the debris, melt ejection through the crust failure location, and movement of debris in the RCS. An independent peer review of SCDAP/RELAP5 has been completed and has identified the various modelling features and deficiencies, that can be found in Reference [3.3].

A.2.2 CONTAIN

The specific goal of the CONTAIN code is to enable an integrated, best estimate prediction of thermal, chemical and radiological conditions inside the reactor containment and auxiliary buildings, following release of coolant and fission products during design basis and severe accident conditions [3.12]. The major processes modelled in CONTAIN include: inter-cell flow, hydrogen combustion phenomena, heat and mass transfer processes, aerosol behaviour (agglomeration, deposition and condensation), fission product behaviour (decay heating and transport), ESFs such as fan coolers, sprays and ice condenser, and MCCI processes. The basis of CONTAIN physics is the inter-cell flow and atmosphere thermal hydraulic models. Gas flow; atmosphere temperature and pressure response; coolant pool thermal response and boiling; primary system sources; condensation and evaporation; structure heat conduction and hydrogen burning are some of the phenomena modelled by CONTAIN. The code is a control volume/flow path approach that allows for an arbitrary arrangement of control volumes and flow path connections.

A number of mechanistic models are available for representing the behaviour of aerosols. The aerosol model is based on the MAEROS [A.4] code. Three agglomeration processes, Brownian, gravitational and turbulent are treated. Also four deposition processes are modelled: gravitational settling, diffusiophoresis, thermophoresis and particle diffusion. An assessment of the modelling of diffusive gravitational aerosol deposition in CONTAIN is provided in Reference [A.11]. The code includes models for the decay and transport of fission products in the containment. A few models of chemical interaction of certain radionuclide species are also available. To determine the location of the radionuclides being studied, CONTAIN tracks them as if they were physically combined with other materials or hosts such as the gas molecules, aerosol particles and structure walls. The predicted fission particle transport is based on the mechanistic movement of airborne host materials. Fission products can be transferred from one host to other at user-specified rates which depend on host temperatures.

The cavity phenomena in CONTAIN are modelled using CORCON-MOD3 computer code. This code was already described in the earlier subsections. The pressure suppression pool models include pool flow, vent clearance, and fission product scrubbing.

An independent peer review of CONTAIN has been completed and has identified the various modelling features and deficiencies [3.4].

Of late, the code developers have concentrated on advanced model development including models for direct containment heating, and Advanced LWR Passive Containment Cooling Systems (PCCS). The direct containment heating models have been benchmarked to Integral Effects Test (IET) results in the Surtsey test vessel at Sandia National Laboratories. Extensive experimental benchmark studies have been performed by CONTAIN as summarised in Reference [3.4].

A.2.3 VICTORIA

The NRC's detailed in-vessel fission product release and transport models are incorporated into the VICTORIA [3.8] computer code. VICTORIA provides detailed mathematical models for prediction of physical and chemical behaviour of radioactive and non-radioactive material release and transport within the reactor coolant system. The mechanisms treated by the fuel fission product release model include diffusive release of volatiles from the fuel grains, surface interaction, condensation, diffusion within the fuel porosity and transgression through cladding material. It should be pointed out that the release from the fuel is based on a simple model which does not treat temperature dependent diffusivities, bubble formation within the grains and grain-boundary sweeping. In addition the effect of an oxidising environment on the fuel itself must also be taken into consideration.

The code also includes a detailed model for fission product chemistry. It treats a set of 167 chemical species and 25 chemical elements. Chemical reactions are modelled in the fuel grains, in the porosity of the fuel, within the fuel-cladding gap, and in the bulk gas of the RPV. The aerosol model in VICTORIA takes account of (1) condensation or evaporation from aerosol surfaces, (2) deposition on structural surfaces, (3) agglomeration of aerosol particles, and (4) convective transport of aerosols from one cell to another. The deposition mechanisms include gravitational settling, laminar or turbulent deposition, Brownian motion, thermophoresis, diffusiophoresis and inertial impaction. While, the agglomeration mechanisms include models for Brownian motion, gravitational motion and inertia in a turbulent field.

Additional model development areas include re-suspension of deposited aerosols (at the time of RCS depressurisation) and revapourisation of condensed species due to decay heat. The VICTORIA code is also not applicable in situations where the thermal hydraulics and fission product release and transport processes are strongly coupled together. Further development needs include models for Radionuclide release in the late phase of the core melt progression. VICTORIA is currently undergoing a peer review under NRC sponsorship.

A.2.4 Hydrogen Mixing Studies (HMS)

HMS is a best-estimate, transient, three-dimensional computer code designed to analyse the transport, mixing and burning of hydrogen and oxygen in containment and associated reactor buildings [3.14]. The code can model geometrically complex facilities with multiple compartments, and can simulate the effect of condensation, heat transfer to walls and structures, chemical kinetics and flow turbulence. HMS is a finite volume code that solves the Navier-Stokes equations for 3D volumes in Cartesian and cylindrical coordinates. It has transport equations for several gas species and one equation for internal energy. Three turbulence models, namely, algebraic, subgrid scale, and - model are provided in the code to determine turbulent velocity and length scales needed to compute the turbulent viscosity. Heat and mass transfer to walls and structures are calculated based on a modified Reynolds analogy. Heat conduction within the structures are calculated based on an one-dimensional conduction approximation. The computer code manual and the code itself has not been released.

A.2.5 TEXAS

The TEXAS computer code is based on a one-dimensional transient model for hydrodynamic calculations developed at Sandia National Laboratories and modified for fuel-coolant interactions. TEXAS solves the one-dimensional, three-field equations describing the fuel-coolant interaction and its hydrodynamics. Two fields represent the coolant as liquid and vapour; one field represents the discrete fuel particles. The liquid and vapour fields are solved using the Eulerian technique and the particle phase is treated using the Lagrangian formulation. In this model, the governing conservation equations for each phase (i.e., liquid, vapour, and particle) are written separately, which allows thermal and mechanical non-equilibrium between the phases to exist. The effects of condensation, evaporation, and interfacial momentum transport are included as source terms in the partial differential equations. A hydrodynamic particle breakup model based on the Rayleigh-Taylor instability mechanism is implemented in TEXAS. During the propagation phase of the FCI, the fuel fragmentation is due to vapour film collapse and coolant liquid jet impingement and entrapment below the fuel surface. This process results in rapid liquid coolant vaporisation leading to the fragmentation of the fuel particle.

A.2.6 IFCI

IFCI is a two-dimensional, Eulerian, four-field computer code that is intended to be used in the prediction of fuel-coolant interactions for nuclear reactors and other industrial applications. The four fields consist of water vapour (steam), liquid water, solid fuel and liquid fuel. A set of conservation equations (mass, momentum, and energy) are solved for each field which allows for non-equilibrium between different fields to exist. The phenomenological models in IFCI include: (1) a dynamic particle breakup, (2) melt surface area convection model (the convected quantity is the melt surface area per unit volume), (3) melt surface area tracking model (this algorithm is used in IFCI for cases where the size scale of the melt is greater than the finite difference length scale),(4) trigger model to initiate the explosion in the mixture, (5) particle fragmentation model to calculate the rate of particle breakup during propagation of the explosion in the mixture, and (6) constitutive relations for heat and momentum transfer between the fluids (different flow regime maps based on the local volume fraction of the mixture components).

A.2.7 PM-ALPHA/ESPROSE

The PM-ALPHA computer code simulates the premixing phase of the FCI. The code is based on a multi field Eulerian formulation. The fields consist of fuel melt, liquid coolant and vapour, and a number of constitutive relationships are provided to describe the interaction between the various fields. These constitutive laws provide interfacial heat and mass transfer, phase change and fuel breakup and fragmentation through a number of correlations. The governing equations are the partial differential equations describing the conservation of mass, momentum and energy. The finite-difference formulation for the solution of the partial differential equations is based on the algorithm developed for the KFIX computer code. PM-ALPHA is a two-dimensional computer code and can simulate the premixing phase of the interaction in cylindrical or Cartesian coordinate systems. A fuel breakup and fragmentation mechanism is also introduced into PM-ALPHA.

The ESPROSE computer code is intended for the simulation of the propagation phase of the explosion once the explosion has been triggered. The code is based on an Eulerian formulation with different fields representing the liquid coolant, fuel melt, and the debris. The debris field represents the fragmented debris. In the recent version of the code (ESPROSE.m) a field consisting of the debris and a homogenous equilibrium mixture of steam and water is introduced (m-field). A set of constitutive laws are provided in ESPROSE. The discretisation of the governing conservation equations and the numerical approach are based on KFIX computer code. The fuel fragmentation is the principal mechanism that drives the propagation phase of the steam explosion (breakup is not considered important during the propagation phase).

A 2.8 ATHLET-CD

The computer code ATHLET (Analysis of THermohydraulics for LEaks and Transients) is developed at the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH. It simulates the whole spectrum of leaks and transients in pressurised (PWR) and boiling water reactors (BWR). The physical models applied are based on a five-equation system, which is presently extended to a six-equation two-fluid model including the effect of non-condensible gases. The reactor coolant system is simulated by a network of one-dimensional components, allowing for crossflow between parallel channels.

For the analysis of severe accidents with core degradation, the extended ATHLET-CD is available. It comprises the basic ATHLET modules supplemented by the following codes [3.94, 3.95, A.31, A.32]:

- Core Heatup and Core Degradation code (ECORE) containing the basic modules to simulate:
    - Core Heatup and Melting (EHEAT)
    - Radiation to Core Surroundings (ERAD)
    - Cladding Oxidation (EDIFF)
    - Mechanical Rod Behaviour (ERODBH)
    - Fuel Rod Power Generation (EPOWER)
- Fission Product Core Release code (EFIPRE), in which the fission product release and rate equation is solved in the module EFPRAT.
- Fission Product and Aerosol Transport code TRAPG containing the module TRAP.

An analogous modular structure as outlined in the previous section is also maintained for the ATHLET-CD version.

The general modules ATHLET, ECORE, EFIPRE and TRAPG serve mainly as control and interface routines. The basic modules are coupled without structural changes, thus facilitating future updates.

The module HECU simulates structures of the Reactor Coolant System (RCS) and the surroundings of the core or test bundle. Axial conduction and melting is not simulated. For the Zircaloy oxidation the same approach is taken as in the module EHEAT. This module simulates the fuel rods in the core or test bundle. Radiative heat transfer is taken into account in radial and axial directions within the core and its surroundings. Cladding oxidation is described by empirical parabolic rate equations for the oxide layer thickness of the oxide mass increase.

Oxidation is limited by the amount of available steam and the remaining Zircaloy. The melting liquefaction of the rod materials is described in a simplify way by user input or by an interpolated Zr-U-O phase diagram. The relocation of materials is simulated by a stop-and-go or candling model with fixed relocation velocity and continuous heat transfer to the remaining cladding and to the fluid. The module ERAD calculates the radiation heat transfer between the fuel rods (i.e. EHEAT and the surrounding structures (HECU). EOXDIF provides the option to simulate metal water reaction by means of oxygen diffusion in the cladding. The module ERODBH simulates the thermomechanical rod behaviour, like thermal expansion of fuel and cladding, ballooning and cladding rupture. The module EPOWER offers different modes of power generation and distribution. Total power may be given by a time function or by integration of neutron kinetics. Axial power is either constant in time or, in case of an electrical heater rod, calculated from a temperature dependent heater resistance. The module EFPRAT calculates the fission product and aerosol release base on rate equations with the coefficients given by experimental data. Burst release, reduction of release due to relocation and unoxidised cladding is taken into consideration. The interface module EFIPRE feeds back the reduction of decay power in the fuel rods due to the released fission products to the module ECORE. The module TRAP simulates the vapour and aerosol retention in the RCS. Brownian, turbulent, gravitational depletion and agglomeration modes are considered for the particles. Vapour sorption is possible on walls. The particles are either transported with the fluid or may fall back to volumes located below. The interface module TRAPG feeds back the distribution of heat sources due to the transported and deposited fission products to the TFD and HECU modules.

A.2.9 RALOC

The original objective of the development of the RALOC-code [3.98] was to calculate the RAdiolysis and LOcal Gas Concentrations within the structures of a containment. Further developments [A.33], [A.34], [A.35] have been pursued to use the code extensively. In the course of developments and for intended coupling with other codes both the tasks and the structure of the code have been changed fundamentally since 1991.

The computer code RALOC is able to evaluate:

- pressure- and temperature build-up and history
- local temperature- and pressure distributions
- energy distribution and local heat transfer to and heat conduction in structures
- local gas distributions (steam and different non condensable gases)
- hydrogen combustion and catalytic recombination
- water distribution
- mass- and volume flow for the release of fluids via opening and leakage
- heat- and combustion gas distribution during fires

Calculations can be performed for simple and multi-compartmented containments and closed buildings of nuclear power plants, as well as for compartmentalised systems (buildings, tunnels, pit system) with more or less large openings to the environment. Mainly the consequences of design basis accidents and severe accidents were analysed with the code in containments of LWRs i.e. for PWR and BWR, but also in containments of VVER powerplants. Some fire events have been investigated, too.

For the description of the physical processes during an accident propagation arbitrary compartment systems and -geometries can be simulated by specified volumes. The conditional changes related to location and time are reduced to a purely time dependent behaviour within the control volumes (nodes). These volumes are connected by 'junctions'. For the simulation of heat transfer and heat conduction via walls and internal components specified structures can be coupled to the nodes. The heat conduction is described in one dimension, for the simulation of heat transfer processes (heat- and mass transfer) different models and correlations are available.

The mass transfer between nodes is described separately for gas and liquid flow by different momentum equations (unsteady, incompressible) taking into account geodetic height differences of the node centres. The mass flow rates of the different components are calculated without slip according to the composition of the source node. Furthermore mass transfer by diffusion is considered. The diffusion flow rate is calculated in a quasi-steady way separately for all gas components.

For the simulation of heat and mass transfer between the zone atmosphere and the structures heat transfer is described by the different physical phenomena of free and forced convection, radiation (wall to gas, gas to wall, wall - gas -wall, wall to wall) and condensation depending on the thermal status of the zone and structures.

Heat conduction is described 1-dimensionally by Fourier's equation. Walls and other internals consisting of different materials can be represented in cylinder type geometries. Such a wall, being denoted as structure, can consist of several materials, arranged one after another. The different materials can be separated by air-filled gaps. Each material can be subdivided into an arbitrary number of layers with different thickness for the calculation of the heat conduction. The arbitrary materials are defined by the values for heat conductivity, specific heat and density.

The combustion of gases (preferably hydrogen) runs in general very fast with large local pressure- and temperature gradients. Due to this nature of the combustion older models for combustion with simply averaged energy release rates over large volumes were not very successful in describing such processes in a realistic way. The now available combustion model DECOR in RALOC uses a new model approach. A 1-dimensional flamefront is assumed with flexible separation of the unburned and burnt parts of the volume. For the simulation of the flame acceleration correlations for the relative movement of the two parts are used, which are based on experimental data. To these correlations functions for the growth of the flame front are added, which are deduced from representative experiments. At the end of the validation process of this model these functions will be put into classes depending on geometry and gas composition.

For a realistic description of accident sequences the simulation of engineered systems is possible like pumps, heat exchangers, ventilation systems, weir, doors and flaps of different kinds with inertia effect, spray systems, catalytic and thermal recombiners and pressure suppression systems.

A.2.10 FIPLOC-MI

The mechanistic Computer Code FIPLOC-MI (**Fi**ssion **P**roduct **Loc**alisation **- M**AEROS, **I**MPAIR) [3.99] has been developed at the Gesellschaft für Reaktorsicherheit (GRS) mbH, Germany, for the integrated analysis of thermal hydraulics and aerosol behaviour in multi compartment geometries. The main purpose of FIPLOC-MI is to calculate the distribution and retention of airborne fission products in a LWR-Containment during a severe accident and to predict the radioactive source term to the environment.

FIPLOC-MI uses a lumped parameter technique. The Containment is represented by a number of control volumes which are interconnected by atmospheric flow paths. The code includes

1. a thermal hydraulic model, which bases on the RALOC code,
2. two independent aerosol models: the monodisperse aerosol model MONAM and the polydisperse, multicomponent aerosol model MAEROS with the moving-grid-condensation model MGA (the letter two were developed by Sandia, USA),
3. the iodine model IMPAIR (has been developed by PSI, Switzerland, and
4. a fission product decay heat model.

These models are numerically tightly coupled so as to include also important interrelation phenomena: fission product transport by natural and forced convection, atmospheric stratification phenomena, local fog formation, condensation on insoluble and hygroscopic aerosols, radioactive heating of gas and walls, etc. Important separate models and their couplings have been examined by several uncertainty and sensitivity analyses.

FIPLOC-MI has been verified by a large number of integral experiments of the projects CSE, DEMONA, LACE, FIPLOC-Verification Experiments, VANAM, PHEBUS and KAEVER. The German VANAM series of 5 experiments, carried out in the 640 m^3 model containment at Battelle, Frankfurt, were especially designed to validate FIPLOC and comparable codes.

FIPLOC has already been applied to:

design and analyses of integral containment experiments, severe accident analyses for German LWRs, Russian type VVERs, and the French/German EPR and analysis of gas and aerosol distribution in ventilation systems of nuclear fuel factories and reprocessing plants.

A.2.11 WECHSL

The WECHSL-Mod3 code, developed by KfK [3.97], is a mechanistic computer code developed for the analysis of the thermal and chemical interactions of initially molten reactor materials with concrete in an axisymmetrical concrete cavity. The code performs calculation of concrete erosion right from the time of bottom head failure of the reactor pressure vessel when a hot molten pool comes long term basemat erosion possibly penetrates the basemat.

It is assumed that a metallic melt layer with an overlying oxidic melt layer exists or alternatively that only oxidic melt layer is present which can contain a homogeneously dispersed metallic phase. Heat generation in the melt is due to decay heat and chemical reactions from metal oxidation. Energy is lost to the melting concrete and to the upper containment by radiation or evaporation of sump water possibly flooding the surface of the melt.

Thermodynamic and transport properties as well as criteria for heat transfer and solidification processes are internally calculated for each time step. Heat transfer is modelled taking into account the high gas flux from the decomposing concrete and the heat conduction in the crusts possibly forming in the long term at the melt/concrete interface.

A.2.12 SageProc

The specific goal of the SageProc code [3.100], developed by GRS, is detailed modelling of release processes during molten-core concrete interactions (MCCI). SageProc was designed to simulate time dependent processes applying the chemical equilibrium computer program ChemSage, which is based upon the SOLGASMIX Gibbs energy minimiser.

The SageProc code thus includes the full abilities of ChemSage to perform thermochemical calculations in complex systems involving phases exhibiting non-ideal mixing properties. This is completed with the flexibility to simulate heterogeneous phase equilibria of an open system, with time dependent input data for the flow of mass into the system, temperatures of input material, and the temperature of reaction.

A driver module WXLSIM is included in order to simulate the environment of the WECHSL code. It provides all input data like mass and composition of corium and concrete, and the required process parameters like melt temperature and depth of concrete ablation.

A database of Gibbs energies assuming ideal-associate mixing properties for applications in MCCI simulations has been developed.

A.2.13 IVA4

The development of the code IVA4 started in the early 80's at the Nuclear Research Centre Karlsruhe.

The code models transient multiphase flows consisting of water, steam, non-condensable gases, microscopic and macroscopic solid particles and/or molten materials. The code implies three velocity fields in 3D space in a thermal and mechanical non-equilibrium. Each of the velocity fields consists inert and non-inert components. The geometry is defined by surface permeabilities and volumetric porosities as functions of time in such a manner that arbitrary technical facilities can be represented. IVA4 is the first multiphase code that consequently exploiting the concept of dynamic fragmentation and coalescence for all velocity fields. IVA4 is an Eulerian general purpose computer code solving 21 partial differential equations for conservation of mass (6), momentum (6), energy (3 in entropy form) and particle number density (3). The numerical solution method reduces the residuals simultaneously to zero (or prescribed accuracy). The use of a novel concept, known as the entropy concept, allows analytical reduction of the discretised conservation equations to single pressure equations per cell. The solution method largely based on a analytical reductions and backwards substitution-method. Important elements of the code are published in [3.96], and [A.36] through [A.39].

Constitutive models:

1. The new unified theory for wall-bubble departure dynamics, taking into account the mutual interaction of the bubble growth and departure, is developed, verified and implemented into the code for the nucleate boiling at heated surfaces, and flashing of superheated liquid at adiabatic solid surfaces. The decrease of the bubble departure diameter with the wall-superheating is in the code for the first time successfully (theoretically) modelled. The exciting feature of this theory was its consequence for prediction of the critical heat flux for boiling and flashing.
2. Film flashing model at adiabatic surfaces.
3. IVA4 is the first multiphase code that consequently uses the concept of dynamic fragmentation and coalescence for all velocity fields.
4. The mechanical interaction models for three-phase flows with or without heat and mass transfer are validated with the QUEOS experimental and Japanese data.
5. The film boiling model is a very important element of the modelling of steam explosion and takes into account that the subcooling and the superheating of the liquid will be dominated by the radiation term.
6. The optical properties of the resulting flow pattern will be used for the radiative interaction-model.
7. The dynamic fragmentation model is analysed by using the non-explosive FARO (JRC - Ispra) system experiments with materials which are interesting for severe accident analysis in the nuclear industry. The KROTOS - experiments are being now used for analysis of the explosive melt - water interaction.
8. IVA4 can predict the pressure wave propagation for single-phase gas, liquid, in 1D and 2D geometry and in two-phase flashing flow. The prediction of explosive wave propagation of gas into 2D space with flow obstacles initially filled with gas and liquid is also possible. The demonstration of the capability of the code for prediction of the pressure wave propagation in single- and two- phase Systems is a prerequisite for prediction of pressure wave propagation in three phase flows.
9. An uncertainty and sensitivity analysis of a melt water interaction simulation is performed and gives an important information about the limitation of the accuracy in the predictions due to uncertainties of some of the important constitutive models and initial conditions in IVA4.

A.3 Parametric Codes

A collection and comparison of all the results obtained from codes such as MELCOR and MAAP shows that releases to the environment can be approximated parametrically with some accuracy. This simplification is based on two assumptions. First, that the fission product species are grouped according to their respective chemical forms and release characteristics (an assumption already present in deterministic codes). And second, that accident states for an individual plant may be categorised according to a set of attributes (e.g, concrete composition, pressure at vessel breach, etc.).

This reduction led to the definition of "source terms issues" and to the development of simplified codes to be used for uncertainty analyses, such as the XXSOR series used in the NUREG-1150 study [A.13]. These codes allow for the interpolation of the results of limited deterministic calculations to a more complete spectrum of accident conditions for a given nuclear power plant. However, it must be emphasised that the XXSOR codes are not time dependent, and therefore their application is somewhat limited to a preordained set of accident types (e.g., "early" and "late" containment failures). Moreover, in the XXSOR treatment, several aerosol transport phenomena are oversimplified.

A.3.1 XXSOR Codes

A separate and specific code was developed for each of the five plants studied in the NUREG-1150 [A.13]. Thus, XXSOR is comprised of SURSOR, ZISOR, GGSOR, PBSOR, and SEQSOR. Differences in the codes are due to different nodalisation of the containment (one compartment for PWRs, two compartments for BWRs), and to intercompartmental flow paths (depending on containment configuration).
The secondary containment and/or reactor building are not modelled, since in general no credit can be given to the integrity of these structures during a severe accident.

Nine radionuclide groups are considered in the models: noble gases, iodine, cesium, tellurium, strontium, ruthenium (and molybdenum), lanthanum (trivalents), cerium (tetravalents), and barium. For each radionuclide group, releases into the containment (sources) are in general calculated using the following mass conservation principle:

$$S(i) = FCOR(i)\ FVES(i) + FCCI(i) + FDCH(i) + FREV(i)\ FCOR(i)\ [1 - FVES(i)] \qquad (A.1)$$

where

S (i) =	fraction of the initial inventory of radionuclide group i that is released into the containment,
FCOR (i) =	fraction of the initial inventory of radionuclide group i which is released from the fuel (in-vessel) prior to vessel breach,
FVES (i) =	fraction of FCOR (i) which is transported through the RCS into the containment,
FCCI (i) =	fraction of the initial core inventory of radionuclide group i released from the fuel during core-concrete interactions (ex-vessel),
FDCH (i) =	fraction of the initial core inventory of radionuclide group i dispersed into the containment atmosphere during pressurised melt ejection, and
FREV (i) =	fraction of radionuclides of radionuclide group i deposited on hot surfaces (mainly in the RCS) which revapourise following vessel breach.

Additional sources may be included, depending on the configuration of the containment (e.g., a source due to wetwell-to-drywell vacuum breakers in BWRs). Revaporization is included only for the I, Cs and Te groups.

In addition, aerosol retention in deep (including pressure suppression pools) and shallow water pools is accounted for through separate parametric decontamination factors (DFs). Other decontamination factors included in the models take into consideration the ESFs (sprays and/or fan coolers).

Releases to the environment are estimated on the basis of containment deposition (decontamination) factors, which are accident- and plant- specific. Therefore, the source terms ST(i) are calculated as:

$$ST(i) = S(i) / DFC(i,j) \quad (A.2)$$

where:

S(i) = source for group i into the containment
DFC(i,j) = containment decontamination factor for group i, accident type j.

A.3.2 ERPRA Codes

For the analysis of the Swiss nuclear power plants, a set of codes was developed which retain the essential features of the XXSOR codes in the representation of gross phenomena, with large associated uncertainties. At the same time an attempt is made to refine the aerosol transport and release mechanisms, with the introduction of time-dependent models similar to the ones used in the existing PSA codes.

The parametric characterisation is limited to the estimation of the source in containment, as in relation A.1. The source, however, is treated in a continuous, time-dependent fashion (spread over a period of time, which depends on accident progression analysis), that is, each release parameter (FCOR (i), FVES (i), FDCH (i), and FCCI(i)) is provided for with appropriate time windows. The time windows for the sources due to core-concrete interactions are also specie-dependent (due to exothermic chemical reaction, some radionuclide species such as Te and Ru may evolve from the core debris for prolonged periods, even though the debris is cooling). All time windows are in general obtained from plant-specific MELCOR calculations [A.14].

Flow paths are defined to account for different accident conditions. For instance, in the PWR model, flow paths are included for gross containment failure, for leak (large and small) from the containment to the environment, and for releases from the stack through the venting system. In addition, a flow path is defined for direct transport of releases from the RCS into the environment (primary containment bypass). Basemat melt-through accidents are treated as being equivalent to very small atmospheric leaks to the environment (the ground and the liquid pathways are not explicitly modelled).

Flow of aerosols and gases in this model is controlled by the definition of time dependent volumetric flow rates, which are also obtained from plant-specific MELCOR calculations. In particular, the impact on radiological releases due to hydrogen combustion can be indirectly simulated, at appropriate times.

Natural deposition phenomena in the primary containment are modelled explicitly, using a formulation similar to that used in the NAUA code. That is:

$$D(i; t) = S(i; t)\, e^{-L\,t} \qquad \text{(A.3)}$$

where:

$D(i; t) =$ Total depletion of aerosol i from atmosphere,
$S(i; t) =$ aerosol i present in atmosphere at time t and
$L =$ the removal rate(s) or depletion parameter(s).

The removal rate L is a function of compartment geometry (total deposition surface and volume) and several deposition velocities. Gravitational settling, thermophoresis, diffusiophoresis and Brownian deposition are included in the estimation of the removal constants. Other removal processes which are not presently included are of secondary importance.

For aerosol, a global estimate of the deposition velocity based on an assumed particle size distribution is used. Large modelling uncertainties are, however, inherent in any deterministic code calculation of particle size distributions; hence, the present estimate of L is considered to be uncertain.

In addition, the deposition constant in Equation (A.3) can account for aerosol depletion if the containment spray system is activated. In this case, instead of L, a removal rate L_{Total} is defined as follows:

$$L_{Total} = L + E_{Spray}\, G \qquad \text{(A.4)}$$

where

$L_{Total} =$ global deposition rate,
$E_{Spray} =$ spray removal efficiency and
$G =$ spray geometric and flow data.

Note that spray removal efficiency is also a very uncertain parameter. In addition, the time of activation of the spray system is controlled by the accident progression characteristics.

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APPENDIX B

B.1 EVNTRE

A generalised event tree processor, EVNTRE /1/, has been developed at Sandia National Laboratories for use in probabilistic risk analyses of severe accident progressions for nuclear power plants. The event tree processor itself is useful for a large class of applications because many reactor safety and risk assessments, as well as decision analyses in general, involve the analysis of the progression of events leading to a large number of conditions or scenarios. Such progressions can depend upon both continuous and discontinuous processes, and the outcomes of particular events can affect subsequent events. Typically, the large uncertainties in the outcomes of particular events can lead to many possible progressions for any given scenario. Major goals of an analysis often include the identification of important factors, such as details of scenario definition or assumptions regarding individual processes that strongly influence the outputs of interest and the associated uncertainty.

EVNTRE was developed to process complex event trees that systematically follow the progression of severe accidents in nuclear power plants. This type of assessment is a particularly difficult type of analysis to accomplish.

Event trees can handle very complicated structures, as (1) multiple branches are allowed at each question o node, (2) branch probabilities can depend upon the path through the tree (indicated by Boolean expressions of the branches taken at previous nodes), (3) continuous processes can be accommodated by branch probabilities that are functions of path dependent parameters (e.g., containment loads can be compared against capacity), and (4) each path through the tree can be classified according to several (approximately 10) characteristics (e. g., mode of vessel breach, time, size and location of containment failure) that are determined by Boolean expressions of the branches.

The capacity of EVNTRE to handle large problems allows time dependence to be treated by identifying several time regimes and constructing sets of questions (e.g., system operation, containment loads and status) appropriate for each time regime.

A flexible facility in EVNTRE for processing multiple sets of inputs to the tree permits Monte Carlo sampling to be used to generate approximate mapping from input to output. A postprocessor can sort or reclassify the output and generate summary tables. These results and statistical analyses of the mappings are the basis of sensitivity analyses that identify the questions, branches, input parameters, or dependencies in the tree that contribute to the outcomes of interest and the associated uncertainty.

In previous event tree analyses, the inability to process large and complex trees led to oversimplifications of severe accidents. EVNTRE removes many of these restrictions on event analysis by processing large trees with detailed descriptions of the dependence of an event on the previously determined events.

B.2 SOLOMON

The Solomon program has been developed by AEA Technology /2/ as the state-of-the-art tool for working with large event trees in a Level 2 PSA context. It is written in ANSI C for UNIX workstation.

The Solomon event tree is defined using a keyword-driven input file. The basic working unit is the event or node which represents a binary question - a probability of the outcome of the node being YES or NO can be set. The nodes of the event tree are defined in the order that they occur in the accident sequence.

A list of nodes can be defined, each one with a fixed probability or a probability that depends on the result of previous events. This feature is called prior path dependency.

Many paths through the event tree can be allocated to a smaller number of End Categories. This allows similar accident sequences to be grouped together. Solomon presents the total probability of each End Category.

For some paths, it is often necessary to jump over nodes. This can occur, for instance, when containment failure is predicted early in the accident sequence. Solomon supports jumping, or straightlining, with prior path dependencies.

Solomon allows calculation of any cornplexity to be carried out within the event tree. For this feature, Solomon provides a new type of node called a *straight node.* These nodes can be placed at any point in the tree and are treated as non-branching events in the accident sequence. This allows the calculations to include prior path dependencies in the same way as the conventional, branching nodes.

User-defined variables can be defined to be *real* or *enumerated.* Real variables are treated as double precision floating point, Enumerated variables can take values from a user-defined list of names, e.g. the variable PRESSURE could be defined as LOW, MEDIUM or HIGH. This is useful for setting switches in the event tree without using meaningless index numbers. Once defined, variables can be set to constants or the results of calculations and used in further calculations as needed.

A variable can also be defined as being *randomly distributed*, in which case its value is defined by a probability density function. This is for use with uncertainty analysis.

It is possible to define numeric functions for repeated use within the event tree. This can be used to calculate a physical model.

A function of one variable can be supplied to the event tree in the form of a set of x,y values, and used to interpolate values within the event tree. This is useful for using experimental data to set probability values. Solomon has a unique way of producing graphical output from the event tree which allows the analyst to condense the entire event tree or to focus in on selected nodes.

Using the nodes of the existing event tree, a new set of nodes can be defined which are logical combinations of some or all of the original nodes. These new nodes are called *supernodes.* Solomon produces a new event tree (the supernode tree) which shows the individual supernode probabilities as well as the path probabilities.

This is a very flexible system which can be used in a number of ways:

- Showing all the paths leading from a single point in the tree. This is useful for checking that the tree has been set up correctly, as individual node probabilities can be compared with the event tree definition.
- Showing the results from one part of the tree. In this case only the nodes of interest are included in the supernodes. This is useful for diagnosis and illustrating the operation of part of the tree.
- Displaying End Category logic. A supernode tree can be constructed which mirrors the logic used to group the event tree paths into end categories. This is useful in discussing the choice of End Categories.
- Reporting results in a simplified containment event tree (SCET). With the correct choice of supernodes, a large tree can be condensed into a SCET which displays important features of the original event tree. A number of SCETs may be needed to illustrate all the main features.

Solomon provides a number of features for performing uncertainty und sensitivity analysis.

A set of importance measures is automatically calculated which give an estimate of the sensitivity of each End Category to each node probability.

Variables used in calculating node probabilities can be defined as being randomly distributed. A number of scaleable distributions are available, including normal, lognormal, uniform, loguniform, triangular and beta functions. Solomon uses Latin hypercube sampling (an enhanced Monte Carlo technique) to calculate the uncertainty in each end category resulting from uncertainty in the input. The result of this is an uncertainty distribution for each End Category.

Using the results from the uncertainty analysis, Solomon can perform a sensitivity analysis using multiple linear regression. This indicates the sensitivity of each End Category to each of the randomly distributed parameters.

B.3 RISKMAN

The event tree module of the RISKMAN program has been developed by PLG Inc. (USA) /3/ and is used for Level 2 Containment Event Tree (CET) evaluation in 11 PSAs performed in the U.S. and Europe. The Code based of a PC software (under DOS) and has following program features:

Because of its user friendly rule based split fraction and end state logic for automatic assignment to sequences during frequency quantification, a large number of top events can be included in the CETs.

Typically, 20 to 30 top events have been utilised for PWRs.

- Multistate top events can be modelled. For containment performance analysis in the PSAs the multistate top event option has been used to represent the RCS pressure at the time of vessel breach. The spectrum of pressure at the time of vessel breach has been represented by as many as four discrete ranges of pressure.

- The RISKMAN event tree module makes no approximations relative to the probability of the success branch of a top event. This is especially important for Level 2 quantification since many of the split fractions associated with phenomenological top events are greater than 0.1, making the success branch probabilities less than 0.9. Many event tree codes assume that the probability of the success branch is 1.0.

- The CET can be linked directly to the RISKMAN Level 1 event trees and quantified for each initiating event. Up to 10 trees and 300 top events can be linked together. The number of sequences in a single tree is unlimited. The event tree quantification time is limited by the complexity of the trees and the sequence frequency truncation value assigned by the user.

- Alternatively, the CET can be quantified separately from the Level 1 model; i.e., one for each key plant damage state identified in the Level 1 analysis. Although a large number of plant damage states were identified as end states for the Level 1 model, the concept of combining plant damage states on the basis of relative frequency and potential risk reduces the complexity of the Level 2 analysis. RISKMAN has been used for a number of Level 2 studies in which the fault tree linking approach was used for Level 1 quantification.

- Split fractions utilised in the RISKMAN Event Tree Module can be represented by probability distributions as well as point estimates. Using Monte-Carlo techniques, the split fraction uncertainties are propogated through the Level 2 model resulting in uncertainty distributions for the frequencies of various types of releases.

The event tree analysis module of RISKMAN can produce a number of different reports for the display of results, including the following:

- Listings of the sequences ranked by frequency.
- The sequence bins (release categories for Level 2 analysis) sorted by frequency.
- Contributions of sequenccs sorted by initiator. For a Level 2 analysis, the key plant damage states resulting from the Level 1 analysis are the initiators.
- The importance of each tree event tree top event, split fraction, and basic event used in the logic models of the CET.
- A detailed listing, of all aspects of a single sequence can be displayed. Included for an individual sequence are, the initiating event, the state of each top event and the split fraction used; the descriptions of the top events and split fractions utilised for the sequence, the sequence frequency, and the event tree end state to which the sequence is assigned.

For the above reports, RISKMAN can generate the information for all of the sequences saved to the database during event tree quantification. The user decides which sequences to save to the database by virtue of a frequency cutoff. The user can impose additional restrictions to select those sequences in the database which are to be included in the reports. The subset of sequences defined by these restrictions are called a sequence group. This sequence group feature provides a virtually unlimited number of ways to interpret the sequences stored in the database.

The Systems Analysis Module of RISKMAN allows a separate fault tree model to be developed for each top event in the event tree. The fault tree limits are 512 gates and 512 basic events for each system. However, numerical results developed from any other fault tree program can be used directly by the Event Tree Module of RISKMAN.

B.4 SPSA

SPSA is a Risk Spectrum PSA-Code and has been developed by the Finnish Centre for Radiation and Nuclear Safety /4/. It consists of models for the event trees, fault trees, cut sets and a data base and uses the small event tree / large fault tree methodology. The code runs on a PC. The fault trees are linked to event trees so that SPSA produces cut set files related to an event tree sequence, and taking into account success sequences. The sequences can be classified with consequences. The sequences leading to the same consequence can be matched (combined) inside the event tree, producing additional cut set files related to an event tree and consequence. Finally, these combined cut set files with same consequence can be mismatched across all event trees to produce new cut set files classified by consequence only. A hierarchical classification of consequences can be created for combining different consequences into higher level groups.

The CCF basic events are automatically created during the minimal cut search. The creation of these events is controlled by data fields.

The interface from the Level 1 to Level 2 part is the binner, which is a rule-based system that creates PDS bins from Level 1 accident sequence cut sets. Each PDS bin contains the 3000 most important cut sets, descriptors as included level 1 sequences and related frequency information. Binning of 500 level-1 accident sequences to 20 PDS bins (millions of cut sets) takes about 20 minutes. The level 2 model can utilise information on level 1 accident sequence and provide information in terms of level 1 building blocks.

For each branch point in the level 2 model where a conditional branching probability is needed, a function is added. This function returns the probability and makes any additional computation that is desired. This computation can include probabilistic computation or computation of the accident propagation or source terms. Time-dependent models can be developed with loops that are executed until a condition triggers change. The maximum number of outcomes from a branch point is 8.

For the creation of these CET functions, a modelling language (CETL, CET Language) has been developed. The code contains an editor and a compiler for CETL in order to make it user-friendly. The basic functions can manipulate reals, integers, boolean variables, strings, distributions, DPDs (Discrete Probability Density variables), vectors, tables, sequence descriptors and cut set lists. In addition to a "standard“ programming language, there are a number of high-level tainties in phenomenological issues were propagated in the event trees (and in the evaluation of accident frequencies and source terms) with a stratified Monte Carlo Code.

SPSA automatically keeps track of edited parts of the PSA model. Updating of PSA results can be done either unconditionally or by automatically detecting changed parts of the model and updating only these and dependent results.

B.5 References

1. EVNTRE
2. SOLOMON
3. RISKMAN
4. SPSA